



HDI-IPEC-22-071

10 CFR 50 Appendix E

10 CFR 50.4

September 14, 2022

ATTN: Document Control Desk  
U.S. Nuclear Regulatory Commission  
Washington, DC 20555-0001

Indian Point Unit No. 2  
Facility License No., DPR-26  
NRC Docket No. 50-247

Subject: Biennial Defueled Safety Analysis Report Update, and Regulatory  
Commitment Change Summary – September 2020 to September 2022

In accordance with Title 10 of the Code of Federal Regulations (CFR) 50.71(e), Holtec Decommissioning International, LLC (HDI) is submitting the Indian Point 2 (IP2) biennial Defueled Safety Analysis Report (DSAR) updates. Enclosure 1 is a complete copy of the entire DSAR Revision 1.

The enclosed DSAR Revision 1 reflects the current plant status which includes permanent cessation of operations and permanent removal of fuel from the reactor vessel. IP2 is in the final spent fuel pool off-load campaign and is preparing for the Independent Spent Fuel Storage Installation Only (ISFSI-only) phase of decommissioning.

There were no Regulatory Commitment Changes nor Technical Specification Bases Changes required to be reported with this biennial update.


The entire DSAR submittal is publicly available, containing no proprietary, personal or safeguards information.

This letter contains no new regulatory commitments.

If you have any questions or need further information, please contact Mr. Walter Wittich, IPEC Licensing at 914-254-7212.

Sincerely,

**Jean A. Fleming**

 Digitally signed by Jean A. Fleming  
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Jean A. Fleming  
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Enclosure 1: Indian Point 2 Defueled Safety Analysis Report



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## *ENCLOSURE TO HDI-IPEC-22-071*

### *INDIAN POINT 2*

## Defueled Safety Analysis Report



# **Indian Point 2**

## **Defueled Safety Analysis Report**



## IP2 DEFUELED SAFETY ANALYSIS REPORT

### Instructions and Key to Unit 2 DSAR Revision 1

#### Current Changes - Revision 1:

The 2022 revision to the Unit 2 Defueled Safety Analysis Report (DSAR), issued as Revision 1 contains revision bars to annotate the following changes:

Change	Description	Affected Sections
1	Clarify classification of a portion of Spent Fuel Pool Cooling system piping to Augmented Quality (QP) / Seismic II to address a Seismic II/I concern	1.7
2	Fuel Storage Building Air Filtration System operations	3.10
3	Transfer operating authority for IP2 from Entergy (ENOI) to Holtec Decommissioning International, LLC (HDI). Adopt HDI's Decommissioning Quality Assurance Plan (DQAP)	1.1, 1.2, 1.6, 2.1, 2.2, 5.1, 5.5 and TRM 5.3.B
4	Retire the Waste Gas Analyzer and Vent Header	1.3, 3.2, 3.3, 4.1, and 6.3
5	Post shutdown system abandonments and long-term system and component tag-outs	2.2, 3.2, 3.3, 3.4, 3.6, 3.11, 3.15, and 4.2.3
6	Use of HI-TRAC Version MS and Multipurpose Canister (MPC-32M)	3.5 & TRM 3.9.E
7	Clarify Iodine (I-131) monitoring because iodines are no longer a source term after permanent shutdown	4.2.3
8	Incipient Fire Brigade	5.1 & 5.2
9	Revised FHA & HIC Analyses	6.2 & 6.5
10	Rescind Fukushima Orders for FLEX	TRM 3.10
11	Rescind Fukushima Order for SFP Level	TRM 3.10
12	Explosive Gas Monitoring	TRM 3.7.B & 5.5.J
13	App R DG fuel oil sampling	TRM 3.8.B

#### Historical Information:

The DSAR has some parts marked as "Historical Information" as defined in NEI 98-03, "Guidelines for Updating Final Safety Analysis Reports". Historical Information is not updated.

The definition of Historical Information means that the information meets the following criteria:

- Information relating to initial plant licensing and start-up that was included in the original FSAR to meet the requirements of 10CFR50.34(b).
- Information that was accurate at the time the plant was originally built, but is not intended or expected to be updated for the life of the plant (unless required by the Commission).
- Information that is not affected by changes to the plant or its operation.
- Information that does not change with time.

### **DSAR Figures and Plant Drawings:**

The figures from the UFSAR that were retained in the DSAR have not been renumbered - following the Instructions is a Table titled "CROSS REFERENCE, DSAR Figures to UFSAR Figures" to be used to determine which UFSAR figure corresponds to the DSAR figure number in the text. Referenced plant drawings are not included in the DSAR. Refer to the most current revision of the official plant drawing.

# IP2 DEFUELED SAFETY ANALYSIS REPORT

## CHAPTER 1 INTRODUCTION AND SUMMARY

### 1.1 Introduction

On February 8, 2017, Entergy Nuclear Operations, Inc. (Entergy) notified the U.S. Nuclear Regulatory Commission (NRC) that it would permanently cease power operations at Indian Point Nuclear Generating Station Unit No. 2 (IP2) no later than April 30, 2020. On May 12, 2020, Entergy submitted certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel to the NRC in accordance with 10 CFR 50.82(a)(1)(i) and (ii). Following the NRC docketing those certifications, the 10 CFR Part 50 license no longer permits operation of the reactor or placement of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). USNRC Letter to Holtec International, dated May 28, 2021 (Reference 1.1-1) transferred control of Indian Point 2's Renewed Operating License DPR-26 to Holtec International subsidiary "Holtec Indian Point 2, LLC," and transferred Entergy's (ENOI) operating authority to Holtec Decommissioning International, LLC (HDI). With the sale and license transfer, IPEC site has adopted the HDI Decommissioning Quality Assurance Program (DQAP).

This Defueled Safety Analysis Report (DSAR) was derived from Revision 27 of the IP2 Updated Final Safety Analysis Report (UFSAR). The DSAR is a licensing basis document that reflects the permanently defueled condition of IP2 and supersedes the UFSAR. The DSAR is intended to serve the same function during SAFSTOR and decommissioning that the UFSAR served during operation of the facility. An evaluation of the systems, structures and components (SSCs) described in the UFSAR was performed to determine the function, if any, these SSCs would perform in a defueled condition. The criteria used to evaluate the major SSCs and the conclusions of the evaluations are provided in appropriate facility documents.

For the purposes of 10 CFR 50.59 screenings or other activities that reference the UFSAR, the DSAR constitutes the safety analysis report reflective of the permanently shut down and defueled facility following the docketing of the certifications required in 10 CFR 50.82(a)(1) in accordance with 10 CFR 50.82(a)(2). The term DSAR is utilized in lieu of the term UFSAR. The DSAR is updated consistent with the requirements of 10 CFR 50.71(e).

Con Edison's ownership/operation of Indian Point 1 and 2 was transferred to Entergy Nuclear in August 2001. Entergy's ownership/operation of Indian Point 1 and 2 was transferred to Holtec International and HDI in May 2021. Consequently, references to Con Edison or Entergy (or derivatives thereof) in this document remain only when used in historical context.

The remainder of the sections of Chapter 1 summarize the principal design features and parameters of the facility. A general description of the facility is included as well as a statement and summary of the General Design Criteria that remain applicable.

### REFERENCES FOR SECTION 1.1

1. USNRC letter to Holtec International, transfer of control of Indian Point 2's Renewed Operating License DPR-26 to Holtec International subsidiary "Holtec Indian Point 2, LLC," and transfer of Entergy's (ENOI) operating authority to Holtec Decommissioning International, LLC (HDI) and Approving Conforming Amendments, dated May 28, 2021.

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### 1.2 Summary Facility Description

#### 1.2.1 Site

Indian Point Unit 2 is adjacent to and north of Unit 1 on a site of approximately 239 acres of land on the east bank of the Hudson River at Indian Point, Village of Buchanan in upper Westchester County, New York. Indian Point Unit 3 (owned and operated by Holtec Indian Point 3, LLC and HDI) is adjacent to and south of Unit 1. The site is about 24 miles north of the New York City boundary line. The nearest city is Peekskill, 2.5 miles northeast of Indian Point. An aerial photograph, [Historical] Figure 2.2-1, shows the site and about 58 miles<sup>2</sup> of the surrounding area.

##### 1.2.1.1 Meteorology

Meteorological conditions in the area of the site were determined during a 2-year test program. The site meteorology provides adequate diffusion and dilution of any released gases as established in the analyses of the postulated fuel handling accident (FHA) and release of gaseous wastes or radioactive liquids provided in Chapter 6.

##### 1.2.1.2 Geology and Hydrology

Geologically, the site consists of a hard limestone in a jointed condition, which provides a solid bed for the facility foundation. The bedrock is sufficiently sound to support any loads, which could be anticipated up to 50 tons per ft<sup>2</sup>, which is far in excess of any load, which may be imposed by the facility. Although it is hard, the jointed limestone formation is permeable to water. Thus, water from the facility that enters the ground would percolate to the river rather than enter any ground water supply. Additional studies by geology consultant, Thomas W. Fluhr, and examination of soil borings confirmed the above conclusions.

In the Hudson River, about 80,000,000 gallons of water flow past the facility each minute during the average tidal flow. This flow provides additional mixing and dilution for liquid discharges from the facility. In fact, however, this aspect is superfluous since the assumption in the facility design is to treat the river water as if it were used for drinking and thus to reduce radioactive discharges, by dilution with ordinary facility effluent, to concentrations that would be tolerable for drinking water. There is minimal danger of flooding at the site as discussed in Section 2.5.

##### 1.2.1.3 Seismology

Seismic activity in the Indian Point area is limited to low-level microseismicity. Detailed field investigations (e.g., Ratcliffe, 1976, 1980; Dames and Moore, 1977) have been conducted in the immediate vicinity of Indian Point and along the major faults in the region. To date, no evidence has been found in the rocks exposed at the surface or sediments overlying fault traces or in cores obtained in the vicinity of Indian Point, that might support a conclusion that displacement has occurred along major fault systems within the New York Highlands, the Ramapo or its associated branches during Quaternary time (the last 1.5 million years). In the vicinity of Indian Point, evidence that no displacement has occurred in the last 65 million years (since the Mesozoic) along specific major structures has been observed.

The facility is designed to withstand an earthquake of Modified Mercalli Intensity VII. The validity of the selection of an Intensity VII earthquake was adjudicated before the Atomic Safety and



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Licensing Appeal Board. The Appeal Board's decision (ALAB-436) verified Intensity VII as the design basis earthquake for the plant.

### 1.2.1.4 Environmental Radiation Monitoring

Environmental radioactivity has been measured at the site and surrounding area in association with the past operation of the three Indian Point Units. These measurements will be continued and reported. The radiation measurements of fallout, water samples, vegetation, marine life, etc., have shown no significant postoperative increase in activity. Noticeable increases in fallout have coincided with weapons-testing programs and appear to be related almost entirely to those programs. The New York State Department of Health in an independent 2-year postoperative study found that environmental radioactivity in the vicinity of the site is no higher than anywhere else in the State of New York.

### 1.2.1.5 Conclusions

Consideration of all the items mentioned above, plus the inherent safety features included in the facility design lead to the conclusion of appropriate suitability of the site for the safe storage and handling of spent fuel at IP2. Accident analyses presented in Chapter 6 verify that the maximum expected doses at or beyond the site boundary are within applicable limits.

### 1.2.2 Facility Description

The facility incorporates a radioactive waste disposal system, fuel handling system and all auxiliaries, structures, and other onsite facilities required for the safe storage and handling of spent fuel.

The general arrangement of the facility is shown on historical Figures 1.2-1 and 1.2-2, and facility drawing 504688 (Formerly Figure 2.2-2). Other general plant arrangement drawings have been removed due to security reasons following September 11, 2001 and can be viewed as plant drawings 9321-2510, 9321-2511, 9321-2514, 9321-2517, 9321-3052, and 209812.

#### 1.2.2.1 Spent Fuel Storage

Auxiliary systems are provided to perform the following functions:

1. Cool system components.
2. Cool the spent fuel storage pool.
3. Dispose of liquid, gaseous and solid wastes.

#### 1.2.2.2 Electrical System

Facility power is provided by a 13.8-kV/6.9-kV autotransformer. Standby power (diesels) is included to ensure further continuity of electrical power for critical loads.

The function of the auxiliary electrical system is to provide reliable power to those auxiliaries required during any normal facility conditions.

The system design provides sufficient independence, isolation capability, and redundancy between the different power sources to avoid complete loss of auxiliary power.

## IP2 DEFUELED SAFETY ANALYSIS REPORT

### 1.2.2.3 Control Room

The facility is provided with a control room containing all necessary instrumentation to ensure safe wet storage of spent fuel and management of radioactive waste processing systems.

### 1.2.2.4 Diesel Generators

The SBO / Appendix R diesel generator is manually available and a standby diesel-generator set can be made available to supply standby power for facility loads in the event of a loss of all other alternating current auxiliary power.

### 1.2.2.5 Waste Disposal System

The waste disposal system collects and processes liquids, gaseous, and solid waste from facility activities for removal from the site. All removals are made in accordance with government guidelines for the process.

### 1.2.2.6 Fuel Handling System

The fuel handling system provides the ability to handle the spent fuel in the spent fuel pit.

The system also includes the following features:

1. Safe accessibility for facility personnel.
2. Provisions to prevent fuel storage criticality.
3. Visual monitoring of the fuel handling procedures at all times.

### 1.2.2.7 Structures

The major structures are the reactor containment building, the primary auxiliary building, the control building, the fuel storage building, the turbine building, and the maintenance and operations building. General layouts and interior components arrangement of the primary auxiliary building, control building, fuel storage building, and holdup tank building were removed due to security reasons following September 11, 2001 and can be viewed on facility drawings.

### 1.2.2.8 Containment

The reactor containment is a steel-lined reinforced concrete cylinder with a hemispherical dome and a flat base.

Ground accelerations for the operational basis earthquake used for containment design purposes and all seismic Class I structures (Section 1.7) are 0.10g applied horizontally and 0.05g applied vertically. In addition, ground accelerations for the design basis earthquake of 0.15g horizontal and 0.10g vertical are used to analyze the no loss-of-function concept. In the permanently shut down and defueled condition, the containment must retain its structural integrity during natural phenomenon events to ensure that it does not impact the safe storage of spent fuel in the spent fuel pit.

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### 1.2 FIGURES

Figure No.	Title
Figure 1.2-1	Indian Point Nuclear Generating Units 1 & 2 [Historical]
Figure 1.2-2	Cross Section of Plant [Historical]

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### 1.3 General Design Criteria (GDC)

The General Design Criteria define or describe safety objectives and approaches incorporated in the design of this plant. These General Design Criteria, tabulated explicitly in the pertinent systems sections in this report, comprised the proposed Atomic Industrial Forum versions of the criteria issued for comment by the AEC on July 11, 1967. Also included in this section, are brief descriptions of related plant features, which are provided to meet the design objectives reflected in the criteria at the time of the initial license application. The descriptions are more fully developed in those succeeding sections of the report indicated by the references.

Con Edison completed a study of compliance with 10 CFR Parts 20 and 50 in accordance with the Commission's Confirmatory Order of February 11, 1980. The detailed results of the evaluation of Indian Point Unit 2 compliance with the then current General Design Criteria established by the Nuclear Regulatory Commission (NRC) in 10 CFR 50 Appendix A, were submitted to the NRC by Con Edison on August 11, 1980 (Reference 1.3-1). Commission concurrence was received on January 19, 1982.

The parenthetical numbers following the section headings indicate the numbers of their related proposed Atomic Industrial Forum versions of the General Design Criteria as described in the first paragraph of this section.

#### 1.3.1 Overall Facility Requirements (GDC 1, 2, 3, and 5)

All systems and components of the facility are classified according to their importance. Those items whose failure or malfunction might cause or increase the severity of an accident that could endanger the public health and safety are designated Class I. Those items important to safely store and handle irradiated fuel but not essential to preventing an accident that would endanger the public health and safety and are not essential for the mitigation of the consequences of these items are designated Class II. Those items that are not directly related to safe storage and handling of spent fuel and are not essential for preventing an accident that would endanger the public health and safety and are not essential for the mitigation of the consequences of those accidents are designated Class III.

Class I systems and components are essential to the protection of the health and safety of the public. Consequently, they are designed, fabricated, inspected, erected, and use materials selected to the applicable provisions of recognized codes, good nuclear practice and to quality standards that reflect their importance.

All systems and components designated Class I are designed so that there is no loss of function in the event of the maximum potential ground acceleration acting in the horizontal and vertical directions simultaneously. The working stresses of both Class I and Class II items are kept within code allowable values for the operational basis earthquake. Similarly, measures are taken in the plant design to protect against high winds, sudden barometric pressure changes, flooding, and other natural phenomena.

Reference sections:

<u>Section Title</u>	<u>Section</u>
Site & Environment;	
Meteorology	2.6
Geology and Seismology	2.7

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Introduction & Summary;	
Design Criteria for Structures and Equipment	1.7

Fire prevention in all areas of the facility is provided by structure and component design, which maximizes the use of fire-resistant materials, optimizes the containment of combustible materials and maintains exposed combustible materials below their ignition temperature in the design atmosphere. Fixed and portable firefighting equipment is provided with capacities proportional to the energy that might credibly be released by fire.

Reference sections:

<u>Section Title</u>	<u>Section</u>
Instrumentation & Control;	
Information Display and Recording	3.13
Auxiliary Systems;	
Facility Service Systems	3.6

A complete set of facility and system diagrams including arrangements, plans, and structural plans and records of initial tests and operation are maintained throughout the life of the reactor. A set of all the quality assurance data generated during fabrication and erection of the essential components of the facility, as defined by the quality assurance program, is retained.

Reference sections:

<u>Section Title</u>	<u>Section</u>
Conduct of Operations;	
Records	5.4
Introduction & Summary;	
Quality Assurance Program	1.6

### 1.3.2 Nuclear and Radiation Controls (GDC 11, 17, and 18)

The facility is equipped with a control room.

The non-nuclear process instrumentation measures temperatures, pressures, flows, and levels in auxiliary systems.

The quantity and types of process instrumentation provided ensures safe storage and handling of spent fuel and radioactive wastes.

The plant vent, the waste disposal system liquid effluent, and the component cooling loop are monitored for radioactivity concentration during all normal conditions.

Monitoring and alarm instrumentation are provided for fuel and waste storage and handling areas to detect excessive radiation levels. The permanent record of activity releases is provided by radiochemical analysis of known quantities of waste.

A controlled ventilation system removes gaseous radioactivity from various areas of the plant and discharges it to the atmosphere via the plant vent. Radiation monitors are in continuous service in these areas to actuate high activity alarms on the control board annunciator.

## IP2 DEFUELED SAFETY ANALYSIS REPORT

Reference sections:

<u>Section Title</u>	<u>Section</u>
Auxiliary Systems; Auxiliary Coolant System	3.3
Waste Disposal & Radiation Protections System; Radiation Protection	4.2

### 1.3.3 Fuel and Waste Storage Systems (GDC 66 - GDC 69)

The spent fuel storage racks are designed so that it is impossible to insert assemblies in other than the prescribed locations. The spent fuel storage pit is filled with borated water. The fuel is stored vertically in an array with sufficient neutron absorbers and distance between assemblies to assure  $k_{\text{eff}} < 1.0$  even if unborated water were used to fill the pit and  $\leq 0.95$  when filled with water borated  $\geq 2000$  ppm boron.

The design of the fuel handling equipment incorporating built-in interlocks and safety features, the use of detailed fuel handling instructions and observance of minimum operating conditions provide assurance that no incident could occur during fuel handling activities that would result in a hazard to public health and safety.

Adequate shielding for radiation protection is provided during fuel handling activities by conducting all spent fuel transfer and storage operations underwater. This permits visual control of the operation at all times while maintaining low radiation levels for periodic occupancy of the area by facility personnel. Pit water level is alarmed and water to be removed from the pit must be pumped out as there are no gravity drains. Shielding is provided for waste handling and storage facilities to permit operation within requirements of 10 CFR 20.

Gamma radiation is continuously monitored in the fuel storage building. A high-level signal is alarmed locally and is annunciated in the control room.

Auxiliary shielding for the waste disposal system and its storage components was also designed to limit the dose rate.

All fuel and waste storage facilities are contained and equipment designed so that accidental releases of radioactivity directly to the atmosphere are monitored and will not exceed the applicable limits.

The spent fuel storage pit is a reinforced concrete structure with a seam-welded stainless-steel plate liner. This structure is designed to withstand any anticipated earthquake loadings as seismic Class I structure so that the liner should prevent leakage even in the event the reinforced concrete develops cracks.

Reference sections:

<u>Section Title</u>	<u>Section</u>
Auxiliary Systems; Sampling System	3.4
Waste Disposal & Radiation Protection System; Waste Disposal System	4.1
Radiation Protection Systems	4.2

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### 1.3.4 Plant Effluents (GDC 70)

Liquid, gaseous, and solid waste disposal facilities are designed so that discharge of effluents and offsite shipments are in accordance with applicable governmental regulations.

Radioactive fluids entering the waste disposal system are collected in sumps and tanks until determination of subsequent treatment can be made. They are sampled and analyzed to determine the quantity of radioactivity, with an isotopic breakdown if necessary. Before any attempt is made to discharge, they are processed as required and then released under controlled conditions. The system design and operation are characteristically directed toward minimizing releases to unrestricted areas. Discharge streams are appropriately monitored and safety features are incorporated to preclude excessive releases.

Liquid wastes are processed to remove most of the radioactive materials. The spent resins from the demineralizers and the filter cartridges are packaged and stored onsite until shipment offsite for disposal.

Reference section:

<u>Section Title</u>	<u>Section</u>
Waste Disposal & Radiation Protection System; Waste Disposal System	4.1

### REFERENCES FOR SECTION 1.3

1. Letter from P. Zarakas, Con Edison, to H. Denton, NRC, Subject: Actions Taken to Comply with NRC Confirmatory Order of February 11, 1980, dated August 11, 1980.

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### 1.4 Design Parameters

#### 1.4.1 Fuel Cladding

The fuel rod design for the facility employs zircaloy as a cladding material.

#### 1.4.2 Fuel Assembly Design

The fuel assembly incorporates the rod cluster control concept in a canless 15 x 15 fuel rod assembly using a spring clip grid to provide support for the fuel rods.



## IP2 DEFUELED SAFETY ANALYSIS REPORT

### 1.5 Supplements and Revisions to Original FSAR

#### 1.5.1 Supplements

Supplement 1 to the Indian Point Unit 2 Final Safety Analysis Report consisted of responses to questions from the Atomic Energy Commission as contained in two letters. The first letter from Peter A. Morris, Director of the Division of Reactor Licensing, on March 5, 1969, to Mr. Donham Crawford of Con Edison, requested additional information on the medical plans and facilities at Indian Point. The questions and responses are found following Tab I of Volume 5 of the original FSAR. These responses were incorporated into Section 11.2.5 of the original FSAR as page changes. The responses to the questions in Volume 5 indicate where the specific answer may be found in the page change.

The second letter to Arthur N. Anderson of Con Edison from Peter A. Morris, dated August 4, 1969, requested additional information on Chapters 1, 2, 3, 4, 5, 6, 7, 8, 11, 12, and 14 of the original FSAR. Supplement 1 responded to several of the questions in the second letter found behind Tab II of Volume 5 of the original FSAR. The responses consisted of questions and answers given in Volume 5 of the original FSAR and also of page changes to the original text of the FSAR in some instances.

Supplement 2 to the Indian Point Unit 2 Final Safety Analysis Report consisted of responses to questions from the Atomic Energy Commission and page changes to the report. The questions were contained in a letter to Arthur N. Anderson of Con Edison from Peter A. Morris, Director of the Division of Reactor Licensing, dated August 4, 1969. The responses consisted of questions and answers added to Volume 5 of the original FSAR in the proper order behind Tab II. Page changes for the FSAR were included with Supplement No. 2.

Supplement 3 to the Indian Point Unit 2 Final Safety Analysis Report consisted of responses to questions from the Atomic Energy Commission and page changes to the report. The questions were contained in a letter to Arthur N. Anderson of Con Edison from Peter A. Morris, Director of the Division of Reactor Licensing, dated August 4, 1969. The responses consisted of questions and answers added to Volume 5 of the original FSAR in the proper order behind Tab II. This supplement responded to several questions concerning Chapters 1, 4, 5, 7, 8, and 11 of the report.

Supplement 4 to the Indian Point Unit 2 Final Safety Analysis Report consisted of responses to questions from the Atomic Energy Commission and page changes to the report. The questions were contained in a letter to Arthur N. Anderson of Con Edison from Peter A. Morris, Director of the Division of Reactor Licensing, dated August 4, 1969. The responses consisted of questions and answers added to Volume 5 of the original FSAR in the proper order behind Tab II. Also included with this supplement was a description of the project reorganization within Westinghouse. This supplement also responded to several questions concerning Chapters 4, 5, 7, 11, and 14 of the report.

Supplement 5 to the Indian Point Unit 2 Final Safety Analysis Report consisted of responses to questions from the Atomic Energy Commission. The questions were contained in a letter to Arthur N. Anderson of Con Edison from Peter A. Morris, Director of the Division of Reactor Licensing, dated August 4, 1969, and a letter to William J. Cahill, Jr., of Con Edison from Peter A. Morris dated November 13, 1969. The responses consisted of questions and answers added to Volume

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5 of the original FSAR in the proper order behind Tab II. The supplement responded to several questions concerning Chapters 1, 4, 6, 11, 12, and 14 of the report.

Supplement 6 to the Indian Point Unit 2 Final Safety Analysis Report consisted of responses to questions from the Atomic Energy Commission. The questions were contained in a letter to Arthur N. Anderson of Con Edison from Peter A. Morris, Director of the Division of Reactor Licensing, dated August 4, 1969, and a letter to William J. Cahill, Jr., of Con Edison from Peter A. Morris dated November 13, 1969. The responses consisted of questions and answers added to Volume 5 of the original FSAR in the proper order behind Tab II. The supplement responded to several questions concerning Chapters 1, 3, 4, 6, 9, and 14 of the report. Also included with this supplement was the Indian Point Unit 2 Containment Design Report.

Supplement 7 to the Indian Point Unit 2 Final Safety Analysis Report consisted of responses to questions from the Atomic Energy Commission and page changes to the report. The questions were contained in a letter to Arthur N. Anderson of Con Edison, from Peter A. Morris, Director of the Division of Reactor Licensing, dated August 4, 1969, and a letter to William J. Cahill, Jr., of Con Edison, from Peter A. Morris, dated November 13, 1969. This supplement responded to several questions concerning Chapters 4, 5, 6, 9, 13, and 14 of the report.

Supplement 8 to the Indian Point Unit 2 Final Safety Analysis Report consisted of responses to questions from the Atomic Energy Commission and page changes to the report. The questions were contained in a letter to Arthur N. Anderson of Con Edison, from Peter A. Morris, Director of the Division of Reactor Licensing, dated August 4, 1969, and a letter to William J. Cahill, Jr., of Con Edison, from Peter A. Morris, dated November 13, 1969. The responses consisted of questions and answers added to Volume 5 of the original FSAR. The supplement responded to questions concerning Chapters 4, 6, 7, and 13 of the report.

Supplement 9, 10, 12, 14, 20 and 21 to the Indian Point Unit 2 Final Safety Analysis Report consisted of corrections and additional information for the original FSAR in the form of page changes.

Supplement No. 11 to the Indian Point Unit 2 Final Safety Analysis Report provided the proposed Technical Specifications for operation of the facility in accordance with the rules of practice, 10 CFR 50.36.

Supplement 13 to the Indian Point Unit 2 Final Safety Analysis Report consisted of responses to questions from the Atomic Energy Commission contained in a letter from Peter A. Morris, Director of the Division of Reactor Licensing, on July 24, 1970, to William J. Cahill, Jr., of Con Edison. The letter requested additional information on Chapters 1, 4, 7, 8, 12, and 14 of the original FSAR. The responses consisted of questions and answers given in Volume 5 of the FSAR and also of page changes to the original text of the FSAR in some instances.

Supplement 15 to the original Final Safety Analysis Report consisted of correction pages that updated certain areas where final design parameters were available and where design modifications had resulted from AEC review. In addition, a cross-reference index was submitted for each chapter of the FSAR where required. The index referenced the responses to questions in Volumes 5 and 6 where additional information could be found concerning specific sections. The proposed Technical Specifications were reissued in their entirety with this supplement. This issue superseded the specifications submitted in Supplement 11.

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Supplement 18 to the original Final Safety Analysis Report consisted of the relocation of information from the site Custom Technical Specifications into the UFSAR for items and topics that were no longer found in the Improved Technical Specifications. It also updated references to the new Technical Specification sections, to information relocated from the Technical Specifications into the Off-Site Dose Calculation Manual (ODCM) and added cross references to the new Technical Requirements Manual (TRM).

Supplement 19 to the original Final Safety Analysis Report consisted of corrections and additional information for the original FSAR in the form of changes to reflect several plant modifications, changes to reflect 10 CFR 100.11, the new fuel design and new core design for Cycle 17 and Cycle 16 Core Reload Design, the permanent increase in  $T_{ave}$  to 565°F, and the approved alternate source term fuel handling accidents (FHB & VC) which take no credit for charcoal filtration. Changes were also included from NRC approved projects, including Appendix "K" Power Uprate [1.4% Power Uprate] with the re-analysis of some of the Chapter 14 accidents to account for the 1.4% power uprate, re-analysis of the Loss of Electrical Load transients and LONE/LOOP transients, and the re-analysis of the Feedwater System Malfunction with a step increase of 120% of nominal feedwater flow to one steam generator, and to reflect the approved Stretch Power Uprate to 3216 MWt.

### 1.5.2 Revisions

Pursuant to 10 CFR 50.71(e), Con Edison submitted an updated Final Safety Analysis Report for Indian Point Unit 2 on July 22, 1982, reflecting changes made up to a maximum of 6 months prior to the submittal date. In addition, the following revisions to the updated Final Safety Analysis Report have been submitted to date:

Revision 1,	July 1983
Revision 2,	July 1984
Revision 3,	July 1985
Revision 4	July 1986
Revision 5,	June 1987
Revision 6,	June 1988
Revision 7,	June 1989
Revision 8,	June 1990
Revision 9,	June 1991
Revision 10,	June 1992
Revision 11,	June 1993
Revision 12,	June 1994
Revision 13,	December 1995
Revision 14,	December 1997
Revision 15,	December 1999
Revision 16,	July 2001
Revision 17,	May 2003
Revision 18,	October 2003
Revision 19,	May 2005
Revision 20,	November 2006
Revision 21,	October 2008
Revision 22,	October 2010
Revision 23,	October 2012
Revision 24,	September 2013

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Revision 25, September 2014

Revision 26, September 2016

Revision 27, September 2018

Revision 0 established the DSAR effective June 2020. For the purposes of 10 CFR 50.59 screenings or other activities that reference the UFSAR, the DSAR constitutes the safety analysis report reflective of the permanently shut down and defueled facility following the docketing of the certifications required in 10 CFR 50.82(a)(1) in accordance with 10 CFR 50.82(a)(2). The term DSAR is utilized in lieu of the term UFSAR. The DSAR is updated consistent with the requirements of 10 CFR 50.71(e).

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### 1.6 QUALITY ASSURANCE PROGRAM

#### 1.6.1 General

The Quality Assurance Program (QAP) for Indian Point Unit 2 is described in the HDI Decommissioning Quality Assurance Program (DQAP) and associated implementing documents provide for control of activities that affect the quality of safety-related nuclear plant structures, systems, and components. The QAP is also applied to certain quality-related equipment and activities that are not safety-related, and where other regulatory or industry guidance establishes program requirements. Changes to the program description are submitted to the NRC in accordance with the provisions of 10 CFR 50.54(a)(3).

#### 1.6.2 Scope

The DQAP applies to all activities associated with structures, systems, and components that are safety related or controlled by 10 CFR 72. The DQAP also applies to transportation packages controlled by 10 CFR 71. The methods of implementation of the requirements of the DQAP are commensurate with the item's or activity's importance to safety. The applicability of the requirements of the DQAP to other items and activities is determined on a case-by-case basis. The DQAP implements 10 CFR 50 Appendix B, 10 CFR 71 Subpart H, and 10 CFR 72 Subpart G. All items and activities affecting safety addressed in Regulatory Guide 1.29 "Seismic Design Classification" revision 3, September 1978, are also governed by the Quality Assurance Program. A list of safety related items is maintained. Elements of the Quality Assurance Program are also applicable to activities and items affecting safety as defined in Licensing commitments. (Reference 1.6-1)

#### 1.6.3 Organization and Responsibilities

The organizational structure responsible for implementation of the Quality Assurance Program is described in the Decommissioning Quality Assurance Program (DQAP). The organizational structure consists of corporate functions and the nuclear facility. The specific organization titles for the quality assurance functions described in the DQAP are identified in procedures. The authority to accomplish the quality assurance functions described in the DQAP is delegated to the incumbent's staff as necessary to fulfill the identified responsibility.

### REFERENCES FOR SECTION 1.6

1. Letter from John D. O'Toole, Con Edison, to Director of Nuclear Reactor Regulation, NRC, Subject: Response to NRC letter of September 23, 1980 to Mr. Zarakas requesting information on the Quality Assurance Program for Indian Point Unit 2 dated March 11, 1981.

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### 1.7 DESIGN CRITERIA FOR STRUCTURES AND COMPONENTS

#### 1.7.1 Definition of Seismic Design Classifications

All structures and components are classified as seismic Class I, Class II, or Class III as recommended in:

1. TID-7024, "Nuclear Reactors and Earthquakes," August 1963 and,
2. G. W. Housner, "Design of Nuclear Power Reactors Against Earth-quakes," Proceedings of the Second World Conference on Earthquake Engineering, Volume I, Japan, 1960, Pages 133, 134 and 137.

#### Class I

Seismic Class I is defined as those structures, equipment, and components whose failure or malfunction might cause or increase the severity of an accident that could endanger the public health and safety.

#### Class II

Class II is defined as those structures, equipment and components that are important to safely store and handle irradiated fuel, but are not essential for preventing an accident that would endanger the public health and safety and are not essential for the mitigation of the consequences of these accidents. A Class II designated item shall not degrade the integrity of any item designated as Class I.

#### Class III

Class III is defined as those structures, equipment, and components, which are not directly related to safe storage and handling of spent fuel. Also, structures, equipment and components that are within this Class are not essential for preventing an accident that would endanger the public health and safety and are not essential for the mitigation of the consequences of those accidents.

The analysis showing that the rupture of a gas decay tank does not exceed the special dose limits selected for Indian Point Unit 2 is found in Section 6.3.

All components, systems, and structures classified as seismic Class I are designed in accordance with the following criteria:

1. Primary steady state stresses, when combined with the seismic stress resulting from the response to a ground acceleration of 0.05g acting in the vertical and 0.10g acting in the horizontal planes simultaneously, are maintained within the allowable stress limits accepted as good practice and, where applicable set forth in the appropriate design standards, e.g., ASME Boiler and Pressure Vessel Code, USAS B31.1 Code for Pressure Piping, ACI 318 Building Code Requirements for Reinforced Concrete, and AISC Specifications for the Design and Erection of Structural Steel for Buildings.
2. Primary steady state stresses when combined with the seismic stress resulting from the response to a ground acceleration of 0.10g acting in the vertical and 0.15g acting in the

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horizontal planes simultaneously, are limited so that the function of the component, system or structure shall not be impaired as to prevent a safe and orderly shutdown of the plant.

All Class II structures and components are designed on the basis of a static analysis for a ground acceleration of 0.05g acting in the vertical and 0.10g acting in the horizontal directions simultaneously.

The structural design of all Class III structures meets the requirements of the applicable building code, which is the "State Building Construction Code" State of New York, 1961. This code does not reference the Uniform Building Code.

The Original Steam Generator Storage Facility (OSGSF) has been constructed for the storage of the original steam generators. The OSGSF is a seismic Class III structure, designed in accordance with the requirements of the State of New York Official Compilation of Codes, Rules and Regulations, Title 9, Subtitle S, 1995 edition, copyright 1999, and the American Concrete Institute (ACI) 318, Building Code Requirements for Structural Concrete, 1999.

Table 1.7-1 gives the damping factors used in the design of components and structures.

The design of seismic Class I structures and components utilizes the "response spectrum" approach in the analysis of the dynamic loads imparted by the earthquake. The analysis is based upon the response spectra shown on Figures 1.7-1 and 1.7-2.

The following method of analysis is applied to seismic Class I structures and components, including instrumentation:

1. The natural period of vibration of the structure or component is determined.
2. The response acceleration of the component to the seismic motion is taken from the response spectrum curve at the appropriate period.
3. Stresses and deflections resulting from the combined influence of normal loads and the seismic load due to the design earthquake (0.05g acting in the vertical and 0.10g acting in the horizontal planes simultaneously) are calculated and checked against the limits imposed by the design standard.
4. Stresses and deflections resulting from the combined influence of normal loads and the seismic loads due to the maximum potential earthquake (0.10g acting in the vertical and 0.15g acting in the horizontal planes simultaneously) are calculated and checked to verify that deflections do not cause loss of function and that stresses do not produce rupture.

Where the vibrator system is of a highly complex geometric shape, such as piping systems, the maximum response from the response curve with the appropriate damping factor is selected. By using this conservative value and demonstrating that the stresses are satisfactory, it becomes unnecessary to perform any further analysis to determine the natural periods of the system.

For a further discussion of the models and methods used for the seismic Class I design of structures, equipment, piping, instrumentation and controls, see Section 1.7.4.



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### 1.7.2 Classification of Particular Structures and Equipment

Examples of particular structure and equipment classifications are given below. These classifications are not intended to be all-inclusive.

<u>Item</u>	<u>Class</u>
<u>Buildings and Structures</u>	
Containment	III
Spent fuel pit	I
Control Building	III
Diesel Generator Building	III
Intake structure (to the extent that water is always available to the service water pumps)	III
Service water screenwell	III
Primary Auxiliary Building	III
Turbine Building	III
Buildings containing conventional facilities Such as the Maintenance and Outage Building Original Steam Generator Storage Facility	III III III
<u>Equipment, Piping, and Supports</u>	
<i>[Note - Class I components (equipment, piping, instrumentation, etc.) located in or supported on a Class II structure are protected from earthquake damage or are backed up by other Class I components located in or supported by a Class I structure.]</i>	
Radiation monitoring system	III
Process instrumentation and controls	III
Fuel assemblies	I
Refueling Water Storage Tank	I
Auxiliary building ventilation system	III
Component cooling loop	III
Instrument air system	III



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Sampling system	III	
Spent fuel pit cooling loop		
Line 328 between outlet and support FSB-2	II	
Line 329 between inlet and support FSB-16	II	
All other lines and portions of lines 328 & 329	III	
Standby power supply system	III	
Diesel generator and fuel oil storage tank	III	
DC power supply system		
Power distribution lines to equipment		
Control panel board		
Motor control centers		
Waste disposal system	III	
Containment crane	III	
Manipulator and other cranes	III	
Conventional equipment, tanks and piping, other than Classes I and II	III	
Auxiliary boiler feed and service water pumps and piping	III	
Chemical and volume control system	III	

### 1.7.3 Design Criteria for Seismic Class I Structures and Equipment

The criteria for functional adequacy of structures, equipment, piping, instrumentation, and controls follow.

No loss of function implies that rotating equipment will not freeze, pressure vessels will not rupture, supports will not collapse under the load, and systems required to be leak tight will remain leak tight.

The criteria for functional adequacy of the structures state stresses will not exceed yield when subjected to a 0.15g ground acceleration. The manner in which these criteria have been met is by limiting stresses in seismic Class I structures to meet the above criteria.

For all seismic Class I piping and their supports, the criteria for functional adequacy and the manner in which the criteria are met are the following:

With a ground acceleration of 0.15g horizontal, the spectral acceleration corresponding to the maximum point on the 0.5-percent critical damping response curve was used to calculate an equivalent static force imparted to the pipe at its support points. This resulted in a seismic design load approximately equal to 0.6W horizontally and 0.4W vertically taken simultaneously, where W is the weight of the pipe including static forces. The sum of the resulting additional stress plus

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the normal stresses was limited to 1.2 times the B31.1 code allowable. The stresses in the pipe supports and hangers were likewise limited to 1.2 times the B31.1 code allowable.

Since all the buildings containing seismic Class I piping are essentially rigid structures, no amplification is expected.

For seismic Class I equipment and tanks the same method was used to arrive at an equivalent static force. In each case, the total of seismic and normal stresses was limited to the applicable code allowable. The refueling water storage tank was designed in accordance with the stress limitations of American Water Works Association Station D100. The loading combinations, which are employed in the design of seismic Class I components of these systems, i.e., vessels, piping, supports, vessel internals and other applicable components, are given in Table 1.7-2.

Table 1.7-2 also indicates the stress limits, which are used in the design of the listed equipment for the various loading combinations. The original design criteria given above and in Table 1.7-2 have been modified in certain instances in accordance with NRC guidance given in References 1.7-1 and 1.7-2.

### 1.7.3.1 Piping, Vessels, and Supports

The reasoning for selection of the load combinations and stress limits given in Table 1.7-2 is as follows.

In the case of the maximum potential earthquake, it is only necessary to ensure that critical components do not lose their capability to perform their safety function, i.e., maintain the capability to safely store and handle spent fuel. This capability is ensured by maintaining the stress limits as shown in line 3 of Table 1.7-2.

### 1.7.4 Models and Methods for Seismic Class I Design

The variety of design problems associated with the seismic analysis of all Class I structures, systems and equipment were approached by various methods. For the design of Class I piping an amplification factor of 4.0 was used with respect to ground motion of 0.15g. This amplification factor was based on the maximum for a one-half percent damping of the ground response spectrum. The fundamental frequency of the reactor building internal structure is approximately 17 cycles/sec. As can be seen from Figure 1.7-2 for this frequency level, no significant building amplification of the ground response is encountered.

With the exception of the containment, primary auxiliary building, and electrical cable tunnel, no dynamic analyses were performed on Indian Point Unit 2 structures, hence no mathematical models were developed. The following methods were used in the seismic design of Class I structures.

#### 1.7.4.1 Containment Building

See Sections 2.0, 3.0, and 4.0 of the Containment Design Report for Indian Point Unit 2 containment building structures and components. In the permanently shut down and defueled condition, the containment building is declassified to seismic class III. The information in this section is retained as bounding information.

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### 1.7.4.1.1 Steel

In the design of the steel, 100-percent of the dead load and 50-percent of the live load were considered. The peak of the response curve for 0.15g ground acceleration and 1.0-percent critical damping was used to obtain the seismic forces, which were distributed by the method described in the Containment Design Report and resisted by the bracing. The 1.0-percent critical damping is conservative since the structure is shop welded and field bolted to the columns. The actual critical damping value would be between 1.0-percent (welded) and 2.5-percent (bolted). A one-third increase over working stress was allowed in the design of the bracing.

### 1.7.4.1.2 Concrete

In the design of the concrete, 100-percent of the dead load and 50-percent of the live load were considered. The Modified Rayleigh Method was used to calculate the natural period and the base shear was distributed by the same method described in the Containment Design Report. The forces determined from the response curve for a 0.15g ground acceleration with 5-percent critical damping were applied at the node points where the masses were lumped for the Rayleigh approach. These loads were resisted by the vertical walls, which acted as shear walls, and horizontal reinforcing, which resisted the moment. The Ultimate Strength Design method of ACI 318-63 was used for the design and construction of the containment building.

### 1.7.4.2 Control Building

The dead load and equipment loads were considered. The period was determined from the formula  $T = 0.1 n$ , where  $n$  = number of stories (Design of Multistory Reinforced Concrete Building for Earthquake Motions by N. M. Newmark, et. al.). The response curve for 0.15g ground acceleration with 2.5-percent critical damping was used to determine the base shear. This base shear was distributed at the floor levels by the same method described in the Containment Design Report and resisted by a rigid frame structure with a one-third increase on allowable working stresses. The design was controlled by a deflection limitation due to the adjacent Unit 1 control building.

Historically, the Control Building was classified as seismic Class I. However, its classification changed following the permanent shut down and defueling of IP2.

### 1.7.4.3 Diesel Generator Building

Due to the light weight of the structure, the wind load controlled the design. Historically, the Diesel Generator Building was classified as seismic Class I. However, its classification changed following the permanent shut down and defueling of IP2.

### 1.7.4.4 Intake Structure

One hundred percent of the live and dead load were considered. The peak of the response curve for 0.1g (OBE) ground acceleration with 5-percent critical damping was used to obtain the seismic loads. The effect of water sloshing was considered in the earthquake analysis (per TID-7024 "Nuclear Reactors and Earthquakes," Section 6.5). Although DBE was not explicitly considered in the calculation (the seismic forces used in the design shows that DBE is not governing), the controlling factor in the design of the intake structure was the service load with the worst combination being one chamber empty and the adjacent chamber filled with water.

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Historically, the Intake Structure was classified as seismic Class I. However, its classification changed following the permanent shut down and defueling of IP2.

### 1.7.4.5 Waste Holdup Tank Pit

One hundred percent of the dead load and 50-percent of the live load were considered (including the tank dead weight on the roof). The peak of the response curve for 0.15g ground acceleration with 5-percent damping was used to determine the base shear. Using working stress limits for the seismic design, service loads controlled the design of the top slab. The bottom slab and wall of the pit were designed for earthquake loads with stresses limited to yield multiplied by the  $\Phi$  factors recommended in Section IV-B of the ACI-318-63 "Building Code." Consideration was given to the tanks in the pit when designing the base slab.

### 1.7.4.6 Spent Fuel Pit

The seismic loads, as determined in TID-7024 "Nuclear Reactors and Earthquakes," Section 6.5, were resisted by the reinforced concrete walls and base slab. Working stresses were used except for the moment at the base of the walls where ultimate strength design was considered with stresses limited to  $\phi f_y$ . The effects of water in the pool are accounted for in this design approach. Ground acceleration of 0.15g was used. In 1990, new high density spent fuel storage racks were installed. Prior to their installation, the spent fuel pit was reanalyzed (Reference 1.7-3). The new racks were also analyzed (References 1.7-3 and 1.7-4).

### 1.7.4.7 Primary Water Storage Tank and Refueling Water Storage Tank Foundation

The seismic loads on the circular wall and center pier were those supplied by the tank manufacturer. The shear force from the earthquake on the water in the tank was applied at  $3/4 L$  above the top slab. The shear force from the earthquake on the tank was applied at  $L/2$  above the top slab, where  $L$  = the height of the tank. The horizontal shear force from the earthquake effect on the dead weight of the foundation was determined by using the peak of the response curve for 0.15g ground acceleration with 5-percent critical damping. A triangular distribution was used. The earthquake effect of the backfill was also considered. The load was applied to the walls as the resultant of a triangular pressure distribution. The stresses were limited to working stress design limits. The temperature steel considerations controlled the design of the walls and center pier.

### 1.7.4.8 Class I Piping Systems

Class I piping systems were designed and analyzed as described in the succeeding paragraphs. However, in an attempt to correlate the simplified method of analysis suggested by the AEC for the H. B. Robinson Nuclear Generating Station, the following discussion is presented:

If no dynamic analysis is performed on Class I piping systems, these systems for H. B. Robinson plant were to be checked to determine whether the results conform to the following formula:

$$1.3 * K S_s + S_n \leq 1.8 S_a$$

**[Note - \*The 1.3 factor was recommended by the AEC to represent the contributions of higher modes above the fundamental mode. Detailed dynamic analyses performed on Indian Point Unit**

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*2, and described later, indicate that where significant stresses exist in piping systems, a more realistic modal contribution factor would be 1.1. However, for the present discussion we will adhere to the 1.3 factor for additional conservatism.]*

where:

$S_s$ -	represents seismic stress including effects of valve motors, from design calculations
$S_n$ -	represents normal primary and bending stresses for loadings other than seismic, from design calculations
$1.8 S_a$ -	equals 1.8 times the allowable stress or yield stress, whichever is higher for code listed materials.
$K$ -	ratio of peak acceleration of floor response spectra to acceleration used in the piping design

The piping design criteria limited the deadweight and seismic stresses to  $0.2 S_a$ . The longitudinal pressure stress is  $0.5 S_a$ .

$$1.3 K (0.2 S_a) + 0.5 S_a \leq 1.8 S_a$$

Solving, the K-factor becomes:

$$K = 5$$

This factor combined with the 1.3 modal contribution factor gives a combined factor of 6.5, which is more than double the original suggested multiplier of 3.

Indian Point Unit 2 conservatively meets the criteria suggested for application on the H. B. Robinson Plant for seismic Class I piping.

However, a different and more detailed method of analysis was actually undertaken to illustrate the conservatism of design approach used for Indian Point Unit 2. This approach is described in detail below:

It is obviously necessary to use simplifying assumptions when performing initial design of piping systems, including restraints, rather than a dynamic analysis involving a trial and error procedure. Simplified design procedures are not uncommon and often suggested in codes, i.e., USAS B31.1 - Power Piping Code.

A complete flexibility analysis involving detailed modeling of Class I piping systems is unnecessary if the conservatism of the simplifying assumptions used in the initial design can be demonstrated. A "third party" review was conducted to establish the adequacy and conservatism of the original design criteria for Class I piping systems as performed by the architect/engineer (United Engineers and Constructors, Inc.) and the seismic restraint supplier (Bergen-Paterson Pipe Support Corp.). The review involved the following steps:

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1. Representatives from Westinghouse and United Engineers and Constructors, Inc., visited the Indian Point Unit 2 site and inspected the Class I piping systems.
2. Based upon their best engineering judgment, representative worst-case lines were selected for detailed dynamic analyses.
3. In exercising their engineering judgment, these representatives looked for the following characteristics, which would indicate possible sources of problems.
  - a. Amplification due to the location and elevation in building.
  - b. Large concentrated masses such as overhung motor-operated valves, particularly in what appear to be flexible sections of the pipe.
  - c. Complexity of configuration of the piping system itself such that application of the original design criteria would be difficult.
  - d. Manual excitation of the pipe by pushing or kicking indicated excessive flexibility either in the pipe excited or the piping attached to it.
4. The results of the dynamic analyses were compared with original design values to determine whether the design approach was conservative. Portions of the following systems were analyzed:
  - a. Service water (Historical Classification).
  - b. Component cooling (Historical Classification).

### 1.7.4.8.1 Design Approach

The design and placement of seismic restraints were predicated on the principle of containing the seismic stresses without restricting the free thermal expansion of the piping system. The systems were designed to have sufficient flexibility to prevent the movements from causing failure of piping or anchors from overstress.

Two fundamental principles underlie the design approach, namely:

1. The system be designed such that its fundamental natural frequency does not coincide with the exciting frequency.
2. The maximum seismic stresses in piping be less than the USAS B31.1 code allowable value. The seismic stresses were limited to 0.2 S allowable (3000 psi). This is extremely conservative since the longitudinal pressure stress accounts for approximately 0.5 S allowable leaving a margin of safety of 0.5 S allowable, which is unused. (Note-this is based on a maximum allowable of 1.2 S<sub>a</sub>)

These fundamental principles should ensure that stresses will be within code allowable stress limits, and that the piping will not go into resonance with the exciting frequency. Tables of recommended maximum spacing of supports, for straight runs of pipe, were developed. The recommended spacing of supports was modified near bends and concentrated masses (i.e. valves) to account for additional weight and flexibility.

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### 1.7.4.8.2 Analysis Approach

In order to determine whether the design procedure resulted in an acceptable system, selected worst case Class I piping systems were modeled and a dynamic flexibility analysis performed. A detailed description of the method of analysis is given below.

The analysis was performed using a proprietary computer code called WESTDYN. The code uses as input system geometry, inertia values, member sectional properties, elastic characteristics, support and restraint data characteristics, and the appropriate Indian Point seismic floor response spectrum for 0.5-percent critical damping. Both horizontal and vertical components of the seismic response spectrum are applied simultaneously.

With this input data, the overall stiffness matrix of the three-dimensional piping system is generated (including translational and rotational stiffness's). The modal participation factors are computed and combined with the mode shapes and the appropriate seismic response spectra to give the structural response for each mode.

Each piping run is modeled as a three-dimensional system, which consists of straight segments, curved segments, and restraints. Straight segments are distinguished from curved segments during data output.

The computer code requires that the piping be represented by a discrete mass model. Each mass includes the contribution of both the steel encasement and conveyed fluid. Where valves or other concentrated masses exist in the piping system, they were included in the model.

Restraints were included in the model at their proper location. The directionality of the restraints was also considered. The detailed dynamic analyses of selected worst-case Class I piping indicated that the method used to design the seismic restraints was conservative. Based on this critical review of the selected worst-case systems and the consistent application of the same design procedure to all completely engineered seismic Class I systems, the seismic design of other Class I systems, not analyzed, was deemed adequate.

The maximum stresses imposed by the normal loads plus loads associated with the design-basis earthquake (DBE) are below  $1.2S$ , where  $S$  is the allowable stress limit obtained from the Power Piping Code - USAS B31.1.0 - 1955.

Some of the items of conservatism employed in the seismic design of Class I piping systems for Indian Point Unit 2 were:

1. The maximum longitudinal stress due to seismic excitation was limited to  $0.2S$  rather than the usual  $0.7S$ .
2. The maximum allowable stress was limited to  $1.2S$ . If the combination of normal and DBE loads were considered as a faulted condition, the allowable membrane and bending stresses could be chosen as those corresponding to 20-percent to 40-percent of the material uniform strain at temperature, respectively. This would give more than a factor of 2 margin between the allowable and the maximum actual stresses.



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3. A low value of the fraction of critical damping was adopted (0.5-percent). Dr. N. M. Newmark recommended a value of 2-percent for vital piping at or just below the yield point. This would reduce the maximum amplification of the ground acceleration.
4. The maximum longitudinal stresses due to pressure, deadweight, and seismic loads were presumed to occur at the same cross-section and some point in the cross-section.

Some averaging of the response spectra was performed to smooth out the erratic response of the earthquake's random behavior. At the high frequency end of the spectra, the acceleration levels of the smoothed spectra converge to the values of the unsmoothed spectra.

It is therefore concluded that the design procedure used to design seismic Class I restraints for Indian Point Unit 2 is conservative.

NRC IE Bulletin (IEB) No. 79-07 was concerned with inadequacies identified in the seismic analysis of certain piping systems at several power reactors. The inadequate treatment of piping loads from earthquakes was attributed to the fact that some piping analysis codes used an algebraic summation of the loads predicted separately by computer code for both the horizontal components and the vertical component of seismic events. In accordance with the IEB, such co-directional loads should not be algebraically added unless certain more complex time-history analyses are performed. The IEB emphasized that to properly account for the effects of earthquakes on systems important to safety, such loads should be combined absolutely or by using techniques such as the sum of the squares.

In response to IE Bulletin No. 79-07, eight (8) Indian Point Unit No. 2 lines were reanalyzed using the UE&C-ADLPIPE-2 dynamic seismic computer code. This code utilizes the worst-case two-dimensional evaluation technique and uses the square root of the sum of the squares option for combining both intramodal and intermodal responses.

The difference between the newly calculated total pipe stress and the originally calculated total pipe stress is not significant. Even after applying a 1.3 "adjustment" factor to the calculated seismic stress component, the total pipe stress remains below the allowable stress limit.

Furthermore, the loads on the pipe supports and equipment nozzles were re-evaluated on the basis of the confirmatory reanalysis and found to be acceptable, as documented in Reference 1.7-5.

### 1.7.4.9 Service Water Lines

The service water lines consist of two 24-in. diameter carbon steel pipes. They run in a common trench, which is backfilled. Assuming that the ends of a pipe are free to displace vertically but not rotate and that the maximum permissible stress is restricted to 30,000 psi, a parametric study showed that the following maximum allowable relative displacements may occur during a seismic disturbance without overstressing the pipe:

Length, ft	1	10	25	50	75	100
Displacement, in.	0.002	0.20	1.25	5.01	11.27	20.04

It is therefore concluded that the service water lines could withstand, without being overstressed, relative bedrock displacements associated with the earthquakes defined for the Indian Point site.



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Historically, the service water lines were classified as seismic Class I. Following the permanent shut down and defueling of IP2, the service water lines are no longer classified as seismic Class I.

### 1.7.4.10 Masonry Walls

In response to IE Bulletin 80-11, safety related masonry walls were evaluated to demonstrate the ability to withstand the specified design load conditions without impairment of wall integrity or the performance of required safety functions. NRC acceptance of this evaluation is documented in Reference 1.7-6. As a result of this evaluation, certain walls in the control building, the Unit No. 1 Superheater building, and the fuel storage building were reinforced.

### 1.7.5 Wind Effects

The IP2 licensing basis does not include tornado protection for the design of the buildings, structures and components. Tornado protection is not a design criterion for IP2. However, the following structures were evaluated for tornado loads: containment building, primary auxiliary building, control building, fuel storage building (including the spent fuel pit), and the intake structure.

Detailed information on the containment structure is found in Appendix B of the Containment Design Report. The containment structure will not be penetrated by a 4-in. x 12-in. x 12-ft wood plank traveling at 300 mph, or by a 4000-pound auto traveling at 50 mph less than 25-ft above the ground.

With respect to the primary auxiliary building, control building, and fuel storage building, information from the siding manufacturer indicates that siding panels will blow out at 170 psf, which is equivalent to a 1.18 psi negative pressure. Panels fail at 60 psf external pressure, which is equivalent to a 162-mph external wind load (60 psf controls the external loading condition). The grits will fail at 90 psf, which is equivalent to a 0.62 psi negative pressure. The 3.25-in. thick siding panels are not capable of resisting any tornado-generated missiles.

Spent fuel pit tornado protection is discussed in proprietary WCAP-7313-L. The intake structure is capable of resisting any wind or missile loads generated by a tornado. This is true for the structure itself, but it does not necessarily include associated equipment.

### 1.7.6 Structural Effects

The potential for damage to Class I structures due to failure of nearby Class II or Class III structures, or due to failure of Class III cranes, has been considered.

The only Class III crane whose failure could endanger any Class I function is the 40-ton fuel storage building overhead crane. The wheels of the bridge and the trolley are shaped such that sliding perpendicular to the rail would not be possible. The lateral load from an earthquake on the trolley crane rail is about 50-percent greater than the lateral loads from impact specified by the AISC Code for design within working stress limits. The stresses on the crane rail are low due to the earthquake load. For this reason, no failure of the crane rail is anticipated.

The turbine building and the fuel handling building are functionally Class III structures. However, these structures have been analyzed using a multidegree of freedom modal dynamic analysis

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method to ensure that there is no potential for gross structural collapse of these structures as a result of the maximum hypothetical earthquake. The results of the analyses are given below. A value of 7-percent structural damping was assumed in the analysis. Total response of the structure was determined on the basis of the "square root sum of the squares" basis of each mode contribution. A similar dynamic analysis was also performed to ensure that no potential gross failure of the Indian Point Unit 1 stack or superheater building could occur for the maximum hypothetical earthquake, or for the design-basis tornado for Indian Point Unit 2. The resultant dead, live, and seismic design stresses in the basic building structure is limited to 0.9 yield of the steel.

The results of specific analyses are discussed in the following sections.

### 1.7.6.1 Seismic Analysis of the Indian Point Unit 2 Turbine Building

A spectrum response analysis was performed for the turbine building considering the design-basis earthquake (DBE), which has a peak horizontal ground acceleration of 0.15g. The associated earthquake response spectrum is shown in Figure 1.7-2.

The foundation was considered rigid since the footings for the structural frames of the building are underlaid by either rock or a lean concrete, which bears on rock. Also, in the analysis, interaction between the turbine and the structural frame for the building was neglected. The analysis, as performed, represents a linear elastic system.

The analysis of the turbine building was performed under the assumption that the north-south motions, east-west motions and vertical motions will be uncoupled. The dynamic analysis effort was limited only to horizontal motions in the east-west and north-south directions. However, vertical components of the earthquake were considered by adding a 0.13g component to dead loads. Each of the models was simulated for the computer program called STARDYNE. A description of the modeling capabilities of STARDYNE are contained in "STARDYNE Structural Analyses Systems Users' Manual" prepared by Mechanics Research, Inc., for Control Data Corporation.

The STARDYNE program was used in three ways. First, the portal frames were analyzed for a static unit force at each portal to determine their resistance to horizontal motions resulting from the turbine bay crane. This information was incorporated into the model for the analysis of the crane girder to determine the distribution of horizontal turbine bay crane loads to the various east-west portal frames. Secondly, the program was used to determine the forces induced in the frames as a result of gravity forces, and, thirdly, the STARDYNE program was used to determine the fundamental frequencies of each of the models and the characteristic shapes. In addition, the STARDYNE program is also capable of determining the modal member forces for each of the fundamental frequencies. This information for each model and mode was stored on tape along with the gravity forces for each model and later used in an earthquake analysis program to determine the maximum probable deflection, acceleration, member forces, member stresses, and the combined gravity plus earthquake member stress responses. Dynamic characteristics of the turbine building are shown in Table 1.7-3.

Results of the analysis indicated that the 0.9 F<sub>y</sub> combined load allowable stress was not violated except locally in the flange of columns where cross bracing framed in eccentric to other joint members. Reduction of stresses to allowable values is accomplished by the addition of flange cover plates.

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While allowable stresses in the cross bracing did not exceed the 0.9 yield stress allowable, it was determined that most of the "x" cross bracing would buckle at very low compressive stress due to high  $\ell/r$  ratios. In order to assure the lateral stiffness of the bents and load carrying capacity as determined in the analysis, cover plates were attached to the bracing equal to the original area of the "x" crossing bracing. This assures design adequacy with only "x" cross bracing in tension assumed to be active in carrying lateral load.

### 1.7.6.2 Seismic Evaluation of the Fuel Storage Building Structure Above the Spent Fuel Pit

The fuel storage building for Indian Point Unit 2 consists of the spent fuel pit constructed of reinforced concrete and founded on rock. The fundamental frequency of the pit is approximately 22 cps and therefore can be considered rigid. The steel superstructure above the pit encloses the pit and supports the fuel cask handling crane. This superstructure was designed as a Class III structure. The seismic loads used in the analysis of the steel superstructure were as follows:

1. Zero period ground acceleration: 0.15g horizontal, 0.10g vertical.
2. 7-percent damping.
3. Response spectrum curve as defined in Figure 1.7-2.
4. Inertial forces for each mass point are determined on the basis of the square root of the sum of the squares.

A dynamic multidegree of freedom, modal analysis of the structure was constructed as shown in Historical Figures 1.7-3 and 1.7-4. The stiffness properties of the elements were determined by the combined stiffness of the frame bents in the north-south and east-west directions taken separately. The stiffness of each bent was determined by the computer program STRUDL. The total inertial forces determined by the dynamic analysis were distributed to each individual bent and resultant member stresses were determined. The crane was assumed fully loaded. Evaluation of these seismic stresses show maximum stresses occurring in diagonal bracing. The maximum stress thus determined in the cross bracing was 18.5 ksi. The maximum combined dead and seismic column load stress determined by the analysis was 12.8 psi compression.

On the basis of these results it was determined that the fuel storage building superstructure was adequately designed to carry the seismic load defined for the site.

In addition to the analysis of the building structure, the fuel crane bridge was evaluated to determine the potential for the crane bridge to lift off its track support in the event of a seismic disturbance. The vertical mode fundamental frequency of the fuel storage building is approximately 9 cps.

The crane bridge has also been analyzed dynamically both loaded and unloaded and for various positions of the trolley. It was determined that the crane with the trolley at the end of the span and unloaded would have a fundamental frequency of approximately 9 cps. Considering potential resonance with the fundamental vertical mode of the building at 9 cps the resulting g-loading was 1.05g. The only potential for crane lift-off will be in the unloaded condition with the trolley parked near the support. Since the unloaded crane will not be parked over the pool no potential hazard exists and vertical restraints are not required.

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### 1.7.6.3 Seismic and Wind Analysis of the Superheater Stack of Indian Point Unit 1 (Historical)

The Indian Point Unit 1 superheater stack has been analyzed for seismic, tornado, and vortex-shedding wind load effects. The results of this analysis are summarized below. As a result of this analysis on the existing stack it is concluded:

1. The stack can withstand a tornado wind load of approximately 300mph prior to buckling failure of the stack steel shell.
2. The maximum stress in the stack at the critical vortex-shedding frequency wind velocity is 7660 psi, which provided a 3.64 factor of safety against stack failure by this mode.
3. The maximum combined dead and seismic stress for the earthquake parameters defined for the site is 19,140 psi, which provides a 1.46 factor of safety against stack failure by this mode.

#### 1.7.6.3.1 Load Case 1 - Tornado

##### I. Load Criteria

Wind = 300 mph

$L = D + W'$

where:

L = Total load

D = Dead load

W' = Tornado load

##### II. Method of Load Analysis

As prescribed in ASCE Paper 3269 for uniform wind velocity with height; no gust factor.

##### III. Allowable Stress Criteria

$$\sigma_a = \frac{0.72Et}{\pi(1 - \nu^2)r} = 27,900 \text{ psi}$$

where:

$\sigma_a$  = allowable stress (psi)

E = modulus of elasticity (psi)

t = shell thickness (in.)

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$\nu$  = Poisson's ratio

$r$  = radius of stack (in.)

### IV. Stress Determination

$$\sigma = \frac{D}{A} + \frac{W\bar{y}r}{I} = 1.54 + 25.75 = 27.29 \text{ ksi}$$

where:

$\bar{y}$  = centroidal height of stack (in.)

$I$  = moment of inertia of stack (in.<sup>4</sup>)

$A$  = cross sectional area of stack (in.<sup>2</sup>)

$$\text{Factor of Safety} = \frac{\sigma_a}{\sigma} = \frac{27.9}{27.29} = 1.02$$

#### 1.7.6.3.2 Load Case 2 - Seismic

##### I. Load Criteria

a) Zero period ground acceleration: 0.15 g horizontal; 0.10 g vertical.

b) Damping 7-percent.

c) Ground response curve - Figure 1.7-2.

$$L = D + E'_h = E'_v$$

where:

$E'_h$  = load resulting from horizontal earthquake component

$E'_v$  = load resulting from vertical earthquake component

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### II. Method of Load Analysis

Multidegree of freedom modal analysis of the superheater building and stack as shown in Figure 1.7-5. The square root of the sum of the squares of seismic inertia forces at mass points is used to determine resultant shears and moments in the stack.

### III. Allowable Stress Criteria

See Load Case 1, item III.

### IV. Stress Determination

$$\sigma = \frac{D}{A} + \frac{E'v}{A} + \frac{E_h \bar{X}r}{I}$$

$$\sigma = 1.54 + 0.20 + 17.4 = 19.14$$

$$\text{Factor of Safety} = \frac{\sigma_a}{\sigma} = \frac{27.9}{19.14}$$
$$= 1.46$$

where:

$\bar{X}$  = lever arm of node inertia force

#### 1.7.6.3.3 Load Case 3 - Vortex-Shedding

##### I. Expression for maximum uniformly distributed force due to vortex-shedding.

$$P = (MF) 1/2 \rho v^2 \times C_L \times D \times L \frac{\pi}{\delta}$$

$C_L$  = Lift coefficient for a stationary circular cylinder

MF = A multiplying factor applied to the lift coefficient to account for a vibrating cylinder

D = Average stack diameter (ft)

L = Length of stack (ft)

$\delta$  = Logarithmic decrement

$\rho$  = Air density (0.0023385 lb - sec<sup>2</sup>/ft<sup>4</sup>)

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$$v = F1 \times V_c$$

$V_c$  = Critical vortex-shedding velocity (fps)

$F1$  = A correction factor, which accounts for the fact that stack oscillations have occurred as high as 30-percent above shedding velocity

$$V_c = \frac{f \times D}{S}$$

$S$  = Stronhal number

$f$  = Fundamental frequency (cps)

### II. Pertinent parameters

$$CL = 0.1$$

$$MF = 4.0$$

$$D = 20\text{-ft}$$

$$L = 334.5\text{-ft}$$

$$\delta = 0.04\pi \text{ (2-percent critical damping)}$$

$$V_c = 42.7 \text{ fps}$$

$$F1 = 1.2$$

$$S = 0.27$$

$$f = 0.576 \text{ cps}$$

### III. Stress criteria

$$\sigma = \frac{D}{A} + \frac{Phr}{2I} = 1.54 + 6.12 = 7.66\text{ksi}$$

$$\text{Factor of Safety} = \frac{\sigma_a}{\sigma} = \frac{27.9}{7.66} = 3.64$$

In addition to the analysis performed for the existing stack it was determined that the stack with 80-ft removed from the top would have the capacity to resist a 360 mph wind for the criteria as defined in Load Case I; the seismic as defined in Load Case II; and the vortex-shedding as defined in Load Case III. The reduction in stack height from El. 400' to approximately El. 202' significantly reduces the wind and seismic stresses discussed above.

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### 1.7.6.4 Seismic and Tornado Evaluation of the Superheater Building at Indian Point Unit 1 (Historical)

A spectrum response analysis was performed for the superheater building considering the design basis earthquake, which has a maximum horizontal ground motion of 0.15g. A dampening coefficient equal to seven percent was assumed for all modes. The earthquake response spectra used is shown in Figure 1.7-2 normalized to 0.15g zero period ground acceleration. In the analysis no interaction with the foundation was considered since the footings for the structural frame for the building are underlaid by rock. Also, in the analysis, the stiffness interaction between the turbine building and the structural frame for the superheater building was neglected, but the mass of the turbine building was included in the dynamic analysis. The analysis, as performed, represents a linear elastic system.

The analysis of the superheater building was performed under the assumption that the north-south motions, east-west motions, and vertical motions were uncoupled. The analysis effort was limited only to horizontal motions in the east-west and north-south directions, and no attempt was made to model vertical motions or to combine vertical and horizontal motions. However, vertical seismic motions have been considered in the results by increasing the dead load stress in building members by a factor equal to two thirds of the combined mode horizontal inertial g-load as determined in either the east-west or north-south direction.

In each direction, north-south and east-west, the column lines were modeled in detail. These structural models were developed for elastic-static analyses obtained from the computer program STRUDL. They were used for two purposes: to develop the master stiffness matrices associated with the two directions, east-west and north-south, used in the dynamic analyses; and to determine resultant member stresses using the equivalent static seismic forces determined from the dynamic analyses.

The dynamic characteristics, frequencies, and mode shapes of the superheater building were determined using the Westinghouse computer program SAND. The equivalent static forces resulting from the dynamic response were developed using a response spectrum seismic analysis performed by the Westinghouse computer program SPECTA.

The equivalent static force associated with a particular mass resulting from a dynamic response is defined as the square root of the sum of the squares of the equivalent static forces associated with that mass for each mode. The equivalent static force associated with a mode and a mass point is defined as the value of the mass times the maximum acceleration associated with the mass point for that particular mode. The maximum acceleration associated with a mode and mass point is defined as follows:

$$(\ddot{U}_m)_{\text{Max}} = (\ddot{A}_n)_{\text{Max}} \phi_m$$

$$(\ddot{A}_n)_{\text{Max}} = \Gamma_n S_{a_n}$$

$$\Gamma_n = \frac{\sum_r M_r \phi_r'}{\sum_r M_r \phi_r'^2}$$



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Where:

$n$  = Refers to mode  $n$

$r$  = Refers to mass  $r$

$\phi'_m$  = Component of  $\phi_m$  in the direction of the earthquake

$\phi_m$  = Component of mode shape  $n$  for mass  $r$

$M_r$  = Mass lumped at point  $r$

$(\ddot{A}_n)_{\text{Max}}$  = Maximum modal acceleration for mode  $n$

$S_{a_n}$  = Spectral acceleration for mode  $n$  from response curve for 7-percent damping

$(\ddot{U}_m)_{\text{Max}}$  = Maximum acceleration in mode  $n$  for mass point  $r$

$\Gamma_n$  = Modal participation factor for mode  $n$

Sectional views in the north-south and east-west directions are shown in Figures 1.7-5 and 1.7-6. A typical column line modeled for STRUDL to determine overall column line stiffness and permit determination of resultant seismic stresses is shown in Figure 1.7-7. In Figure 1.7-8 is presented the dynamic model used to determine inertial forces.

Results of the analysis showed several column lines contained diagonal bracing with stresses, which exceeded the allowable stress value of  $0.9 f_y$ . In addition, several of the cross bracings showed compressive stress levels, which exceeded the expected buckling stress as determined by the  $\ell/r$  ratio for the member. Overstressed members can be strengthened by attaching cover plates to the angle bracing. In a few instances, columns were found to be locally overstressed due to eccentric positioning of cross bracing. These areas can be reinforced by flange cover plates. Approximately 30 tons of additional plate will strengthen the structure.

With respect to tornado resistance of the structure, total lateral load in the north-south direction is approximately 10-percent, and in the east-west direction 20-percent, less than the seismic-induced lateral load on the structure.

Tornado loads were based on a 360-mph wind using the shape factors for a rectangular building as defined in ASCE Paper 3269. It was assumed that 20-percent of the wall area of the building was still intact as a reaction surface for the wind in addition to the total surface area of major equipment and the stack at its existing height. On the basis of this analysis, the building has approximately the same resistance capacity to a 360-mph tornado wind as it does for the 0.15g earthquake.

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### 1.7.6.5 Evaluation of Structural Modifications

In the analysis of the superheater and turbine buildings under lateral loads, the following connections were examined:

1. Gusset plates.
2. Check of connections between beams and columns to determine their adequacy to transfer horizontal shear load.
3. Check of connections at column bases in the foundation to determine their ability to transfer the given horizontal shear load. For those column base connections subjected to a net uplift load, an analysis has been performed to ensure that they are adequate for these loads.

If it was found that a connection was inadequate to support the given load, it was redesigned.

It is not necessary to reanalyze the turbine building after the redesign because the building stiffness characteristics are essentially the same as those assumed in the initial analysis. This is because the significant fixes involved the cross-bracing system, which is made up of pairs of cross bracing members. In the initial analysis, both sets of cross bracing were assumed active. However, the bracing system was such that cross members would buckle under a very small compressive load. Therefore, lateral building load must be carried in tension by the bracing system.

The fix used in the redesign was to double the area of cross bracing. The bracing in compression, due to buckling, is not active in resisting lateral building load. Therefore, only half of the cross-bracing assumed in the initial analysis, which is in tension, resists this load. However, since the area of cross-bracing has been doubled, the resultant effective lateral resistance is the same as that assumed in the original analysis.

An initial analysis was made of the superheater building using the existing design parameters. After completion of the analysis, the overstressed members were strengthened and a dynamic reanalysis made.

Tables 1.7-4, 1.7-5, and 1.7-8 give the relative comparisons in stiffness, horizontal inertial load, and frequency between the initial analysis and the reanalysis.

Subsequently, retired Unit 1 superheater-associated equipment has been removed from certain areas of the superheater building and the areas refurbished to provide permanent administrative facilities. These areas do not contain any safety-related equipment. The total loading on the superheater building has been reduced from the original design loading due to the removal of superheater-associated equipment. Therefore, the administrative facilities will not adversely affect the response of the superheater building during a safe-shutdown earthquake.

### REFERENCES FOR SECTION 1.7

1. NRC Generic Letter, Relaxation in Arbitrary Intermediate Pipe Rupture Requirements, G.L. 87-11, dated June 19, 1987.

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2. NRC Branch Technical Position MEB 3-1, Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment.
3. Letter (Attachment I) from S. Bram, Con Edison, to NRC, Subject: Indian Point Unit No. 2 Spent Fuel Storage Capacity Increase, dated January 19, 1990.
4. Letter (Attachment B) from S. Bram, Con Edison, to NRC, Subject: Request for License Amendment to Technical Specification Modifying Spent Fuel Storage Requirements, dated June 20, 1989.
5. Letter from Cahill, Con Edison, to A. Schwencer, Director of Nuclear Reactor Regulation NRC, Subject: Supplemental Response to IE Bulletins 79-02 and 79-07, dated November 27, 1979.
6. Letter from Steven A. Varga, NRC to John D. O'Toole Con Edison, Subject: Completion of IE Bulletin 80-11, "Masonry Wall Design" for Indian Point Nuclear Generating Unit No. 2 (IP2), (Safety Evaluation Report included) dated October 19, 1983.

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**TABLE 1.7-1**  
**Damping Factors**

COMPONENT	PERCENT OF CRITICAL DAMPING
Containment structure	2.0
Steel assemblies:	
Bolted or riveted	2.5
welded	1.0
Vital piping systems	0.5
Concrete structures above ground	
Shear Wall	5.0
Rigid Frame	5.0

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**TABLE 1.7-2**  
**Loading Combinations and Stress Limits**

Loading Combinations	<u>Vessels</u> <sup>1</sup>	Piping	Supports
1. Normal loads	$P_m \leq S_m$ $P_L + P_B \leq 1.5 S_m$	$P_m \leq S$ $P_L + P_B \leq S$	Working stresses or applicable factored load design values
2. Normal + design earthquake loads	$P_m \leq S_m$ $P_L + P_B \leq 1.5 S_m$	$P_m \leq 1.2 S$ $P_L + P_B \leq 1.2 S$	1-1/3 working stresses or applicable factored load design values
3. Normal + maximum potential earthquake loads	$P_m \leq 1.2 S_m$ $P_L + P_B \leq 1.2 (1.5 S_m)$	$P_m \leq 1.2 S$ $P_L + P_B \leq 1.2 (1.5 S)$	Deflections and stresses of supports limited to maintain supported equipment within their stress limits
4. Normal + pipe rupture loads	$P_m \leq 1.2 S_m$ $P_L + P_B \leq 1.2 (1.5 S_m)$	$P_m \leq 1.2 S$ $P_L + P_B \leq 1.2 (1.5 S)$	Deflections and stresses of supports limited to maintain supported equipment within their stress limits

Where:

- $P_m$  = primary general membrane stress; or stress intensity
- $P_L$  = primary local membrane stress; or stress intensity
- $P_B$  = primary bending stress; or stress intensity
- $S_m$  = stress intensity value from ASME B and PV Code, Section III
- $S$  = allowable stress from USAS B31.1 Code for Pressure Piping

Note: 1. Limited to vessels designed to ASME, Section III, Class A (or Class 1) rules. Otherwise use piping for stress limits.

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**TABLE 1.7-3**  
**Dynamic Characteristics of the Turbine Building**

MODE No.	Frequency (cps)	Values
1	0.5042	0.08
2	1.6141	0.12
3	2.2849	0.19
4	4.3292	0.2
5	5.2813	0.2
6	8.2814	0.18
7	12.1704	0.15
9	15.1274	0.15
10	20.754	0.15
11	22.4809	0.15
12	23.8001	0.15
13	27.3040	0.15
14	33.9678	0.15

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**TABLE 1.7-4**  
**Relative Stiffness Percentages**

RELATIVE LOCATION IN SUPERHEATER BUILDING	Percentage Increase In Stiffness Between First And Second Analysis (Percent)	
	EAST-WEST DIRECTION	NORTH-SOUTH DIRECTION
BOTTOM	8	56.7
MIDDLE	18.3	41.4
TOP	19.9	10.4

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**TABLE 1.7-5**  
**Inertial Loads**

Relative Location in Superheater Building	Inertial Loads for First and Second Analysis (Units: Kips)			
	East-West Direction		North-South Direction	
	Original	Reanalysis	Original	Reanalysis
Bottom	908	908	1091	1102
Middle	1888	1914	1687	1803
Top	1242	1271	1082	1181



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**TABLE 1.7-6**  
**Frequencies**

Frequencies for First and Second Analysis  
(Units: cps)

<u>MODE</u>	EAST-WEST DIRECTION		NORTH-SOUTH DIRECTION	
	<u>ORIGINAL</u>	<u>REANALYSIS</u>	<u>ORIGINAL</u>	<u>REANALYSIS</u>
1	0.94	1.0	0.72	0.88
2	2.07	2.15	1.58	2.13
3	4.08	4.19	3.47	4.12

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### 1.7 FIGURES

Figure No.	Title
Figure 1.7-1	Ten Percent of Gravity Response Spectra
Figure 1.7-2	Fifteen Percent of Gravity Response Spectra
Figure 1.7-3	Fuel Storage Building North-South Model [Historical]
Figure 1.7-4	Fuel Storage Building East-West Model [Historical]
Figure 1.7-5	Indian Point Unit 1 Superheater Building North-South Section
Figure 1.7-6	Indian Point Unit 1 Superheater Building East-West Section
Figure 1.7-7	Column Line "G"
Figure 1.78	Representation of Lumped Mass Model of Superheater Building Used in Dynamic Analysis

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### **1.8 Control of Heavy Loads**

Control of heavy loads in the Fuel Storage Building is addressed in Section 3.5.5.

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## CHAPTER 2 SITE AND ENVIRONMENT

### 2.1 Summary and Conclusions

This section of the DSAR sets forth the site and environmental data, which together formed the basis for the criteria for designing the facility and for evaluating the routine and accidental release of radioactive liquids and gases to the environment. These data support the conclusion that there will be no undue risk to public health and safety with the facility as designed and the environmental characteristics as described. This conclusion rests not only upon the data, but upon the scientific documentation of several independent consultants in their particular area of expertise-health physics, demography, geology, seismology, hydrology, and meteorology.

Environmental characteristics of the area have been documented by field measurements and studies conducted since 1958. These studies quantified the effects on the environment of the operation of nuclear power plants.

Conservative projections have been made of the probable growth of population in the area, and these projections have been taken into account in facility design both as to control of accidents and as to assumptions about operation.

[Historical Information] According to 1980 population estimates, about 50 people reside within a 1100-m radius of Unit 2 (most of them to the east-southeast), and approximately 2600 live within 1-mile. Approximately 75,000 people reside within a 5-mile radius of the facility. The largest concentration of population is in the City of Peekskill, the center of which is about 2.5-miles northeast of the site. The most densely populated 15-degree sector, within 5-miles, is toward Peekskill to the northeast.

The 1960 population within a 15-mile radius of the site was approximately 352,000, whereas in the year 2000 the estimated population is 1,107,195. The projections do not indicate, and there is no reason to conclude otherwise, that the land usage within this radius will shift appreciably during the intervening period. (The land is now zoned principally for residential and state park use, although there is some industrial activity and minor or isolated agricultural and grazing activity.)

The outer boundary of the low-population zone has been set at 1100 m from Unit 2.

Geologically, the site consists of a hard limestone in a jointed condition that provides a solid bed for the facility foundation. The bedrock is sufficiently sound to support any loads that could be expected up to 50 tons/ft<sup>2</sup>, which is far in excess of any load that may be imposed by the facility. Although it is hard, the jointed limestone formation is permeable to water. Thus, if water from the facility should enter the ground, it would percolate to the river rather than enter any ground-water supply.

About 80 million gallons of Hudson River water flow past the facility each minute during the peak tidal flow. This flow will provide additional mixing and dilution for liquid discharges from the facility. The assumption in the facility design is to treat the river water as if it were used for drinking and thus to reduce radioactive discharge, by dilution with ordinary plant effluent, to concentrations that would be tolerable for drinking water. There is a very low probability of flooding at the site.

[Historical Information] Seismic activity in the Indian Point area is limited to low-level microseismicity. Detailed field investigations<sup>1-3</sup> have been conducted in the immediate vicinity of Indian Point and along the major faults in the region. To date, no evidence has been found in the

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rocks exposed at the surface or sediment overlying fault traces or in cores obtained in the vicinity of Indian Point that might support a conclusion that displacement has occurred along major fault systems within the New York Highlands, the Ramapo, or its associated branches during Quaternary time (the last 1.5 million years). In the vicinity of Indian Point, evidence that no displacement has occurred in the last 65 million years (since the Mesozoic) along specific major structures has been observed.

The facility is designed to withstand an earthquake of Modified Mercalli Intensity VII as required by Appendix A to 10 CFR 100 "Seismic and Geologic Siting Criteria for Nuclear Power Plants." The validity of the selection of an Intensity VII earthquake was adjudicated before the Atomic Safety Licensing Appeal Board. The Appeal Board's decision (ALAB-436) verified Intensity VII as the facility's design-basis earthquake.

Meteorological conditions in the area of the site were determined during a 2-year program (1955 to 1957). The validity of these conclusions has been verified by several programs, including that performed by the Atmospheric Services Department of York Services Corporation in completing a meteorological update for Consolidated Edison Company in 1981 (see Appendix 2A).

[Historical Information] These data have been used in evaluating the effects of gaseous discharges from the facility during normal operations and during the postulated accidents (as previously bounded by the historical hypothetical loss-of-coolant accident). The evaluations indicate that the site meteorology provides adequate diffusion and dilution of any released gases.

Environmental radioactivity has been measured at the site and surrounding area since 1958 in association with the operation of Indian Point Unit 1 and the construction and operation of Indian Point Units 2 and 3. Unit 3 is owned by Holtec Indian Point 3, LLC. These measurements will be continued and reported. The radiation measurements of fallout, water samples, vegetation, marine life, etc., have shown no perceptible post-operative increase in activity. Noticeable increases in fallout have coincided with weapons-testing programs and appear to be related almost entirely to those programs. The New York State Department of Health in an independent 2-year postoperative survey<sup>4,5</sup> found that environmental radioactivity in the vicinity of the site is no higher than anywhere else in the State of New York.

[Historical Information] Consultants who have participated in the preparation of the various reports, measurements, and conclusions appearing in this chapter include Dr. Merrill Eisenbud, director of Environmental Radiation Laboratory, Institute of Industrial Medicine, New York University; Dr. Benjamin Davidson (deceased), meteorologist and director, Geophysical Science Laboratory, New York University College of Engineering; Dr. James Halitsky, senior research scientist, Department of Meteorology and Oceanography, New York University, College of Engineering; Dr. Edgar M. Hoover, Regional Economic Development Institute, Inc.; Metcalf and Eddy Engineers, hydrology specialists; Quirk, Lawler, and Matusky Engineers, Environmental Science and Engineering Consultants; Mr. Karl R. Kennison, consulting civil and hydraulic engineer; and Woodward-Clyde Consultants, consulting engineers, geologists and environmental scientists.

## IP2 DEFUELED SAFETY ANALYSIS REPORT

### REFERENCES FOR SECTION 2.1

1. Ratcliffe 1976.
2. Ratcliffe 1980.
3. Dames & Moore 1977.
4. Hollis S. Ingraham, Consolidated Edison Indian Point Reactor Environmental and Post Operational Survey - August, 1965, Division of Environmental Health Services, New York State Department of Health.
5. Hollis S. Ingraham, Consolidated Edison Indian Point Reactor Environmental and Post Operational Survey - July, 1966, Division of Environmental Health Services, New York State Department of Health.

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### 2.2 Location

#### 2.2.1 General

[Historical Information] Indian Point is a multiunit site consisting of approximately 239 acres of land on the east bank of the Hudson River at Indian Point, village of Buchanan, in upper Westchester County, New York. Indian Point Units 2 and 3 (see Section 2.2.3) are located north and south, respectively, of Unit 1, which has been retired. The site is about 24-miles north of the New York City boundary line. The nearest city is Peekskill, located 2.5-miles northeast of Indian Point, with a population of about 20,000. An aerial photograph, Historical Figure 2.2-1, shows the site and about 58-miles<sup>2</sup> of the surrounding area.

#### 2.2.2 Access

The site is accessible by several roads in the village of Buchanan. A paved road links the eastern boundary of the site to the existing facility. The existing wharf is used to receive heavy equipment as needed. The site is not served by rail.

#### 2.2.3 Site Ownership And Control

Holtec Indian Point 2, LLC owns the Indian Point Units 1 and 2 Nuclear Power Plants. As shown in Figure 2.2-3, the Algonquin Gas Transmission Company has a 26-inch gas mainline and a 30-inch gas mainline on a 65-foot wide right-of-way running east to west through the property. Unit 2 is 1450-ft north of the 26-in. Algonquin gas mainline. One 30-inch main and 2-24-inch mains pass under the river to a pipeline facilities station on the easement near the river. One 24-inch main is available as a bypass alternative and ends in the pipeline facilities station while the other two continue as the 30-inch and 26-inch mains.

The Algonquin Gas Transmission Company has installed a 42-inch gas mainline that crosses the Hudson River south of the site property and turns north. It passes through the south-easternmost corner of the site and then crosses Broadway between the Buchanan Switchyard and the GT 2/3 fuel oil storage tank. The 42-inch main joins the 30-inch and 26-inch mains north of the switchyard and east of the site. Potential hazards posed by the mainlines on IPEC structures and equipment have been evaluated.

The Georgia-Pacific Corporation has an easement, 1610-ft long and 30-ft wide, through the southerly part of the Indian Point site. The Georgia-Pacific easement is used for overhead electrical power and telephone lines and underground gas, water, and sewer lines. These easements permit HDI to determine all activities within the right-of-way in order to ensure safe operation of the units.

Units 1, 2, and 3 have a security fence surrounding the "protected" areas. Access to the protected areas is controlled via security buildings that are manned on a 24-hr basis. In addition, spaces within the protected area designated as "vital areas" are provided with additional access control. All roads within the site are continuously patrolled by security personnel. A site plot plan is shown in Plant Drawing 327152.

#### 2.2.4 Activities On The Site

The principal activities on the site are the generation, transmission, and distribution of electrical energy; associated service activities; activities relating to the controlled conversion of the nuclear

## IP2 DEFUELED SAFETY ANALYSIS REPORT

energy of fuel to heat energy by the process of nuclear fission; and the storage, use, and production of special nuclear source and byproduct materials.

### 2.2 FIGURES

Figure No.	Title
Figure 2.2-1	Aerial Photo of Indian Point Site and Surrounding Area [Historical]
Figure 2.2-2	Indian Point Building Identification Replaced with Plant Dwg No. 504688
Figure 2.2-3	Algonquin Gas Transmission Pipeline Hudson River Crossing & Indian Point Nuclear Generation Facility



## IP2 DEFUELED SAFETY ANALYSIS REPORT

### 2.3 Topography

[Historical Information] The Indian Point Generating Station is on the east bank of the Hudson River. The river runs northeast to southwest at this point but turns sharply northwest approximately 2-miles northeast of the facility. The west bank of the Hudson is flanked by the steep, heavily-wooded slopes of the Dunderberg and West Mountains to the northwest (elevations 1086 and 1257-ft, respectively, above mean sea level) and Buckberg Mountain to the west-southwest (elevation 793-ft). These peaks extend to the west and gradually rise to slightly higher peaks.

The general orientation of this high ground is northeast to southwest. One-mile northwest of the site, Dunderberg bulges to the east. North of Dunderberg and the site, high grounds reaching 800-ft form the east bank of the Hudson River. At this location the Hudson River makes a sharp turn to the northwest. To the east of the site, peaks are generally lower than those to the north and west. Spitzenberg and Blue Mountains average about 600-ft in height, and there is a weak, poorly-defined series of ridges that run in a north-northeast direction. To the west of the site there are the Timp Mountains at an elevation of 846-ft. To the south of the site, elevations of 100-ft or less gradually slope towards Verplanck. The river south of the site makes another sharp bend to the southeast and then widens as it flows past Croton and Haverstraw.

Historical Figure 2.3-1 shows topographic features of the site and the surrounding areas.

#### 2.3 FIGURES

Figure No.	Title
Figure 2.3-1	Topographical Map of Indian Point and Surrounding Area [Historical]

## IP2 DEFUELED SAFETY ANALYSIS REPORT

### 2.4 Population and Land Use

#### 2.4.1 Overview

The population within a 50-mile radius of the Indian Point site has been estimated for 1990. These population estimates were taken from statistics recently released by the U.S. Census Bureau. The population within the 50-mile radius of Indian Point has increased from the 1980 estimates by approximately 68,000 people, less than half of one percent.

#### 2.4.2 Population And Land Use

According to 1990 estimates, approximately 15.465 million people live within a 50-mile radius of the Indian Point site. A major part of this number live in New York City, an area 25 to 50-miles south of the facility. Approximately 1650 persons, concentrated in sectors south to southeast of the station, live within 1-mile of the facility. Approximately 74,000 persons live within 5-miles of the facility.

The area surrounding the Indian Point site is generally residential with some large parks and military reservations. Some increased commercial development has occurred within a mile of the station since 1980. Most of the area to the east of the Hudson River within 15-miles of the site is zoned for residential uses. West of the Hudson within a 15-mile radius, the Palisades Interstate Park and residential areas are the dominant land uses. The only agricultural areas within 15-miles are south or northwest of the facility on the west side of the River.

Several maps and tables are included to illustrate the population distribution and land use of the area. Figure 2.4-1 and Figure 2.4-2 show the sector/zone approach to the population data and the area within a 50-mile radius of the Indian Point site. Historical Figures 2.4-3 through Figure 2.4-5 illustrate the 1980 population distribution radially by sectors out to 50-miles from the facility site. Historical Figure 2.4-6 through Figure 2.4-8 show, respectively, the land uses based on official zoning maps, areas served by public utilities, and areas served by sewage systems, all as of 1970. Table 2.4-1 explains the sector/zone designations for the population maps and tables that follow. [Historical] Table 2.4-2 through Table 2.4-18 give the 1990 estimated populations for all sector/zones within a 50-mile radius of the Indian Point site.

The New York State Department of Commerce projects no substantial increases in population from 1986 to the year 2013 in any of the four counties in the vicinity of Indian Point.

[Historical] Table 2.4-19 and Table 2.4-20 show the estimated and projected land uses by County for 1960 and 1980, respectively. These estimates were developed by the Regional Economic Development Institute, Inc., from Regional Planning Association data.

#### 2.4.3 Low-Population Zone

About 50 people reside within a 1100-m radius of Unit 2, most of them to the east-southeast. This distance was used as the outer boundary of the low population zone in the analysis of a postulated fission product release. The water boundary (Peekskill Bay) of the more densely populated area of Peekskill was used as the population center distance, which exceeds 1-1/3 times the distance from the reactor to the outer boundary of the low-population zone. A low-population zone outer boundary radius of 1100-m satisfies both 10 CFR 100.11(a)(3) and 10 CFR 50.67. The low-population zone population in the year 2010 is projected to be approximately 88.

## IP2 DEFUELED SAFETY ANALYSIS REPORT

### 2.4.4 Exclusion Area

The exclusion area for Indian Point Unit 2 includes facility property within a 520-m radius of the reactor containment. An exclusion radius of 520-m satisfies both 10 CFR 100.3(a) and 10 CFR 50.67.

### 2.4.5 Population Data Sources

The population data used in this section were developed from the following sources:

1. 1978 Official Population Projections for New York State Counties, prepared by the Economic Development Board, New York State Department of Commerce.
2. Population by Municipality 1970-2000, prepared by the Westchester County Department of Planning, October 1979.
3. Population of Rockland County, Capacity and Forecast, 1970-2000, prepared by the Rockland Planning Board, April 1978.
4. Population Estimate and Projections, Orange County, New York, prepared by the Orange County Planning Department, March 1980.
5. Putnam County Population Projections, prepared by the Putnam County Planning Board, 1977.
6. New Jersey Revised Total and Interim Age and Sex Population Projections, 1980-2000, prepared by the New Jersey Department of Labor and Industry, Division of Planning and Research, Office of Demographic and Economic Analysis, April 1979.
7. State of Connecticut Population Projections for Connecticut Municipalities and Regions to the Year 2000, prepared by the Office of Policy and Management, Comprehensive Planning Division, February 1980.
8. Pennsylvania Projection Series, Summary Report, Employment by Labor Market Area, and Population and Labor Force by County for 1980, 1985, 1990, 1995 and 2000, Report No. 78, PPS-1, prepared by the Office of State Planning and Development, State Economic and Social Research Data Center, June 1978.

## IP2 DEFUELED SAFETY ANALYSIS REPORT

**TABLE 2.4-1**  
**Sector and Zone Designators for Population Distribution Map<sub>1</sub>**

Sector Nomenclature		Zone Nomenclature	
Centerline of Sector in Degrees True North from Facility	22.5° Sector <sub>2</sub>	Miles from Facility	Zone
0 and 360	A	0-1	1
22.5	B	1-2	2
45	C	2-3	3
67.5	D	3-4	4
90	E	4-5	5
112.5	F	5-6	6
135	G	6-7	7
157.5	H	7-8	8
180	J	8-9	9
202.5	K	9-10	10
225	L	10-15	15
247.5	M	15-20	20
270	N	20-25	25
292.5	P	25-30	30
315	Q	30-35	35
337.5	R	35-40	40

**Notes:**

1. An area is identified by a sector and zone alphanumeric designator (refer to Figure 2.4-1). Thus, area A1 is that area, which lies between 348.75- and 11.25-degrees true north from the facility out to a radius of 1-mile. Area G4 would be that area between 123.75- to 146.25-degrees and the 3- and 4-mile arcs from the facility.
2. The letters I and O have been omitted from sector designators so as to eliminate possible confusion between letters and numbers.

## IP2 DEFUELED SAFETY ANALYSIS REPORT

**[Historical] TABLE 2.4-2**  
**Population Estimates, 1990, For All Sectors**

Zone	Population
1	1,644
2	15,130
3	18,428
4	14,225
5	24,508
6	25,922
7	28,096
8	25,967
9	36,930
10	46,488
15	342,852
20	488,652
25	920,850
30	2,171,399
35	2,276,172
40	3,451,123
45	3,416,140
50	2,199,601

## IP2 DEFUELED SAFETY ANALYSIS REPORT

**[Historical] TABLE 2.4-3**  
**Population Estimates, 1990, for Sector A (North)**

<u>Sector, Zone</u>	<u>Population</u>
A1	0
A2	70
A3	0
A4	0
A5	400
A6	390
A7	5,301
A8	5,898
A9	2,474
A10	874
A15	4,132
A20	36,987
A25	31,000
A30	57,873
A35	39,998
A40	20,100
A45	17,689
A50	40,853

## IP2 DEFUELED SAFETY ANALYSIS REPORT

[Historical] TABLE 2.4-4  
Population Estimates, 1990, for Sector B (North-Northeast)

<u>Sector, Zone</u>	<u>Population</u>
B1	0
B2	54
B3	139
B4	143
B5	1,721
B6	1,553
B7	867
B8	246
B9	2,123
B10	1,187
B15	4,343
B20	7,982
B25	20,310
B30	16,651
B35	4,800
B40	6,991
B45	8,457
B50	5,761

## IP2 DEFUELED SAFETY ANALYSIS REPORT

**[Historical] TABLE 2.4-5**  
**Population Estimates, 1990, for Sector C (Northeast)**

<u>Sector, Zone</u>	<u>Population</u>
C1	0
C2	4,879
C3	9,102
C4	4,159
C5	5,534
C6	3,895
C7	2,382
C8	1,594
C9	630
C10	1,034
C15	10,371
C20	9,685
C25	8,200
C30	12,479
C35	13,687
C40	13,067
C45	7,901
C50	6,621



## IP2 DEFUELED SAFETY ANALYSIS REPORT

**[Historical] TABLE 2.4-6**  
**Population Estimates, 1990, for Sector D (East-Northeast)**

<u>Sector, Zone</u>	<u>Population</u>
D1	49
D2	2,379
D3	2,691
D4	1,899
D5	2,324
D6	2,272
D7	4,667
D8	4,713
D9	5,982
D10	3,900
D15	32,854
D20	14,721
D25	8,961
D30	82,240
D35	21,876
D40	18,762
D45	12,991
D50	60,032

## IP2 DEFUELED SAFETY ANALYSIS REPORT

**[Historical] TABLE 2.4-7**  
**Population Estimates, 1990, for Sector E (East)**

<u>Sector, Zone</u>	<u>Population</u>
E1	59
E2	560
E3	0
E4	289
E5	279
E6	345
E7	1,769
E8	1,138
E9	3,287
E10	3,762
E15	17,702
E20	5,099
E25	22,465
E30	20,987
E35	15,730
E40	159,720
E45	162,993
E50	101,121

## IP2 DEFUELED SAFETY ANALYSIS REPORT

[Historical] TABLE 2.4-8  
Population Estimates, 1990, for Sector F (East-Southeast)

<u>Sector, Zone</u>	<u>Population</u>
F1	147
F2	305
F3	336
F4	689
F5	260
F6	987
F7	475
F8	860
F9	758
F10	1,999
F15	19,121
F20	11,728
F25	49,821
F30	120,701
F35	58,734
F40	33,691
F45	0
F50	29,199

## IP2 DEFUELED SAFETY ANALYSIS REPORT

**[Historical] TABLE 2.4-9**  
**Population Estimates, 1990, for Sector G (Southeast)**

<u>Sector, Zone</u>	<u>Population</u>
G1	575
G2	2,298
G3	1,295
G4	769
G5	420
G6	3,702
G7	3,892
G8	2,672
G9	2,159
G10	6,890
G15	27,939
G20	23,849
G25	86,999
G30	44,001
G35	17,093
G40	79,903
G45	240,102
G50	328,012

## IP2 DEFUELED SAFETY ANALYSIS REPORT

**[Historical] TABLE 2.4-10**  
**Population Estimates, 1990, for Sector H (South-Southeast)**

<u>Sector, Zone</u>	<u>Population</u>
H1	109
H2	1,782
H3	1,363
H4	741
H5	93
H6	0
H7	0
H8	78
H9	5,039
H10	5,752
H15	22,162
H20	103,969
H25	226,002
H30	252,482
H35	209,921
H40	535,969
H45	723,004
H50	469,960

## IP2 DEFUELED SAFETY ANALYSIS REPORT

**[Historical] TABLE 2.4-11**  
**Population Estimates, 1990, for Sector J (South)**

<u>Sector, Zone</u>	<u>Population</u>
J1	531
J2	650
J3	20
J4	129
J5	1,351
J6	4,012
J7	3,133
J8	4,308
J9	5,189
J10	4,321
J15	40,993
J20	55,102
J25	220,032
J30	954,691
J35	1,472,384
J40	1,907,927
J45	1,601,010
J50	702,739

## IP2 DEFUELED SAFETY ANALYSIS REPORT

**[Historical] TABLE 2.4-12**  
**Population Estimates, 1990, for Sector K (South-Southwest)**

<u>Sector, Zone</u>	<u>Population</u>
K1	174
K2	1,245
K3	1,282
K4	2,049
K5	8,093
K6	4,124
K7	2,526
K8	2,531
K9	6,291
K10	9,371
K15	86,297
K20	72,902
K25	146,895
K30	427,391
K35	321,209
K40	534,296
K45	444,572
K50	353,770

## IP2 DEFUELED SAFETY ANALYSIS REPORT

**[Historical] TABLE 2.4-13**  
**Population Estimates, 1990, for Sector L (Southwest)**

<u>Sector, Zone</u>	<u>Population</u>
L1	0
L2	63
L3	1,621
L4	2,694
L5	2,184
L6	4,059
L7	2,876
L8	902
L9	2,087
L10	4,021
L15	26,019
L20	28,753
L25	41,514
L30	94,167
L35	31,725
L40	89,824
L45	124,188
L50	54,722



## IP2 DEFUELED SAFETY ANALYSIS REPORT

**[Historical] TABLE 2.4-14**  
**Population Estimates, 1990, for Sector M (West-Southwest)**

<u>Sector, Zone</u>	<u>Population</u>
M1	0
M2	359
M3	188
M4	399
M5	169
M6	274
M7	170
M8	15
M9	96
M10	271
M15	5,139
M20	4,976
M25	14,343
M30	8,817
M35	21,625
M40	18,889
M45	47,849
M50	30,319

## IP2 DEFUELED SAFETY ANALYSIS REPORT

**[Historical] TABLE 2.4-15**  
**Population Estimates, 1990, for Sector N (West)**

<u>Sector, Zone</u>	<u>Population</u>
N1	0
N2	292
N3	214
N4	0
N5	0
N6	0
N7	0
N8	23
N9	438
N10	63
N15	3,321
N20	8,827
N25	10,234
N30	7,794
N35	14,233
N40	9,028
N45	9,007
N50	2,109

## IP2 DEFUELED SAFETY ANALYSIS REPORT

**[Historical] TABLE 2.4-16**  
**Population Estimates, 1990, for Sector P (West-Northwest)**

<u>Sector, Zone</u>	<u>Population</u>
P1	0
P2	85
P3	52
P4	0
P5	32
P6	58
P7	9
P8	626
P9	357
P10	2,004
P15	17,997
P20	9,983
P25	12,394
P30	47,277
P35	5,927
P40	9,121
P45	3,960
P50	3,917

## IP2 DEFUELED SAFETY ANALYSIS REPORT

**[Historical] TABLE 2.4-17**  
**Population Estimates, 1990, for Sector Q (Northwest)**

<u>Sector, Zone</u>	<u>Population</u>
Q1	0
Q2	0
Q3	125
Q4	189
Q5	55
Q6	0
Q7	29
Q8	321
Q9	0
Q10	1,039
Q15	7,023
Q20	9,872
Q25	10,745
Q30	12,244
Q35	10,160
Q40	7,942
Q45	5,653
Q50	6,962

## IP2 DEFUELED SAFETY ANALYSIS REPORT

**[Historical] TABLE 2.4-18**  
**Population Estimates, 1990, for Sector R (North-Northwest)**

<u>Sector, Zone</u>	<u>Population</u>
R1	0
R2	109
R3	0
R4	76
R5	1,593
R6	251
R7	0
R8	42
R9	20
R10	0
R15	17,439
R20	44,219
R25	10,935
R30	12,144
R35	17,070
R40	5,893
R45	6,764
R50	3,504

# IP2 DEFUELED SAFETY ANALYSIS REPORT

**[Historical] TABLE 2.4-19**  
**Estimated Land Use in 1960 and Projected Land Use in 1980<sub>1</sub>**  
**Within a 55-Mile Radius**

	Intensive 1960 and 1980			Nonintensive 1960			Nonintensive 1980				11	12
	1	2	3	4	5	6	7	8	9	10		
	Residential	Industrial/ Commercial	Total	Institutional and Park	Public Rights- of-Way	Total	Community Facilities Institutions	Parks Recreation	Public Rights- of-Way	Total	Open	Grand Totals
1960												
Square miles	1032	216	1248	696	418	1114					4062	6424
Percentage of total developed land	43	9	52	29	19	48						
High	58	12	45		22							
Low	32	2	15		15							
1980												
Square miles	2040	368	2408				876	784	682	2342	1674	6424
Percentage of total developed land	43	8	51				19	16	14	49		
1960-1980												
Square miles of land to be developed	1400	220	1620							1228		
Percentage of total land to be developed			58							42		

## Notes:

- The averages were derived from the data in "Table 3. The Use of Developed Land in Selected Areas of the Regions." RPA Bulletin Number 100, Page 21, September 1962. The data for square miles excludes Monmouth County from the original Regional Plan Association (RPA) totals.

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**[Historical] TABLE 2.4-20 (Sheet 1 of 2)**  
**Land Use Projection by County for 1980 in Square Miles**  
**Within a 55-Mile Radius**

Counties in Con Ed Study Area			Intensive			Low Intensive		
<u>State</u>	<u>In RPA Region</u>	<u>Outside RPA Region</u>	<u>Residential</u>	<u>Industrial/Commercial</u>	<u>Community Facilities/Community Institutions</u>	<u>Parks Recreation</u>	<u>Public Rights-Of Way</u>	<u>Open</u>
Conn.	Fairfield	Litchfield New Haven	183 [30] <sub>1</sub> [88]	33 [6] [19]	92 [3] [73]	83 [3] [65]	71 [2] [55]	171 [5] [134]
N.J.	Bergen Essex Hudson Middlesex Morris Passaic Somerset  Union		118 83 26 126 (58) <sub>2</sub> 130 75 71 (24) [34] 63 [3]	22 16 5 22 (10) 23 14 13 (4) [8] 12 [1]	20 6 3 18 69 23 16 [107] 6 [9]	19 6 3 16 63 21 15 [97] 6 [9]	16 5 2 14 54 18 12 [83] 5 [7]	38 12 6 34 129 43 30 [199] 11 [18]
N.Y.	Dutchess Nassau Orange Putnam Rockland		106 230 110 37 56	19 41 20 6 10	152 5 154 42 25	138 4 140 38 23	117 4 119 32 19	283 9 286 79 46

# IP2 DEFUELED SAFETY ANALYSIS REPORT

**[Historical] TABLE 2.4-20 (Sheet 2 of 2)  
Land Use Projection by County for 1980 in Square Miles  
Within a 55-Mile Radius**

Counties in Con Ed Study Area			Intensive			Low Intensive			
State	In RPA Region	Outside RPA Region	Residential	Community Industrial/ Commercial	Facilities Community	Parks Recreation	Public Rights-Of Way	Open	
N.Y.	Suffolk	Sullivan Ulster	279 (199) <sub>2</sub>	50 (35) [4]	92 [117]	84 [106]	72 [90]	172 [217]	
			[8] <sub>1</sub>	[12]	[207]	[188]	[160]	[386]	
	Westchester		162	31	53	48	42	99	
	Bronx		25	4	3	3	2	5	
	Kings		42	7	4	4	4	8	
	New York		14	2	1	1	1	3	
	Queens		65	11	7	7	6	13	
P.A.	Richmond	39	7	3	2	2	5		
	Pike	[7]	[1]	[76]	[69]	[59]	142		
	Total RPA Region <sub>3</sub>	2040	368	794 <sub>3</sub> *	724 <sub>3</sub> *	617 <sub>3</sub> *	1482 <sub>3</sub> *		
Total Consolidated Edison Area		2078	383	1385	1261	1073	2583		

## Notes:

- Figures in brackets are for those counties outside RPA's Region. They are added to the total for Con Ed's area.
- Figures in parentheses are those portions of the RPA Region contained in the Con Ed area.
- Total RPA Region figures followed by \* indicate that only the portion of the counties in Con Ed's area are included.



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Figure 2.4-4	Ten Mile Sector/Zone Diagram [Historical]
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Figure 2.4-6	Map and Description Showing Land Usage [Historical]
Figure 2.4-7	Map and Description of the Area Showing Public Utilities [Historical]
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### 2.5 Hydrology

The hydrologic features of the Indian Point site are relevant to the analysis of radioactive liquid discharges from the facility. These features are the Hudson River, ground water and wells, and surface-water reservoirs. During the conduct of normal activities, liquid wastes are discharged to the Hudson River through the circulating water discharge canal. Ground-water contamination from accidental ground seepage or leakage from the facility will flow to the river because of the higher elevation of the facility relative to the river.

Between 2005 and 2007, GZA GeoEnvironmental (GZA), performed a comprehensive hydrogeologic investigation of the site. This investigation was initiated to understand groundwater flow and contaminant transport. During this investigation numerous borings were advanced to study the site geology, hydrology and aquifer properties. Details of the geology, hydrology and aquifer properties can be found in the GZA report<sup>5</sup>.

The hydrology in the environs of the Indian Point site has been extensively studied for Con Edison by numerous consultants, augmenting the data base established through the investigations of various governmental agencies. The initial Con Edison study was conducted in 1955 by Kennison,<sup>1</sup> who analyzed the flow characteristics of the river at the site. Metcalf and Eddy<sup>2</sup> further examined the river flow, and also investigated local groundwater hydrology and surface-water reservoirs. The salient aspects of these and other studies<sup>3,4</sup> are reported below.

The Hudson River below the dam at Troy (immediately below the confluence of the Hudson and Mohawk Rivers) is a tide-influenced, estuarine waterway. (see Figure 2.5-1.) Fresh water from the combined Hudson and Mohawk Rivers, as well as from numerous tributaries, discharges directly into the tidal portion of the river. Seawater enters the extreme lower reaches of the river through the Narrows and the Harlem/East River. The distribution of saltwater is influenced by fresh water flow, tides, physical characteristics of the river channel, and weather.

Flow in the Hudson River is controlled more by the tides than by the runoff from the tributary watershed. River width opposite the facility ranges from 4500 to 5000-ft. Water depths within 1000-ft of the shore near the site are variable with an average depth of 65-ft; at some points the depth exceeds 85-ft. River cross-sectional areas in the vicinity of the site range from 165,000 to 170,000-ft<sup>2</sup>. Tidal flow past the facility is about 80 million gpm about 80-percent of the time, and it has been estimated that this frequency flow is at least 9 million gpm in a section 500-600-ft wide immediately in front of the facility. Mean tidal flow in the vicinity of the site is over 70 million gpm.

The average downstream flow (for a 17-year period preceding 1930) is in excess of:<sup>2</sup>

- 11,700,000 gpm 20-percent of the time.
- 6,800,000 gpm 40-percent of the time.
- 4,710,000 gpm 60-percent of the time.
- 3,100,000 gpm 80-percent of the time.
- 1,800,000 gpm 98-percent of the time.

The facility is designed to use the dilution characteristics of the large tidal flow and releases will be managed such that discharges into the river would not contravene regulatory limitations.

Historical flow patterns were further examined by Quirk, Lawler, and Matusky<sup>3,4</sup> who reported both long-term (monthly) river discharges and potential drought flows. Quirk, Lawler, and Matusky also

## IP2 DEFUELED SAFETY ANALYSIS REPORT

analyzed and reported on the hydraulic conveyance properties of the estuary and the effects of tide and salinity on movement in the estuary.

Review of historical records indicates that flooding at the site is non-existent. Flood stages are primarily the effect of tidal influence, with the secondary influence of runoff. The highest recorded water elevation in the vicinity of the site was 7.4-ft above mean sea level (MSL), which occurred during an exceptionally severe hurricane in November 1950. Subsequent to that occurrence, the highest water elevation recorded at the site was 9-ft 8-in. above MSL, which occurred during the extra-tropical Superstorm Sandy in November 2012. Since the river water elevation would have to reach 15-ft 3-in. above MSL before it would seep into any of the Indian Point buildings, the potential for any flooding damage at the site appears to be extremely remote.

Seven different flooding conditions governing the maximum water elevation at the site were investigated, including the following:

1. Flooding resulting from runoff generated by a probable maximum precipitation over the entire Hudson River drainage basin upstream of the site.
2. Flooding caused by the occurrence of any upstream dam failure concurrent with heavy runoff generated by a standard project flood.
3. Flooding due to the occurrence of a probable maximum hurricane concurrent with a spring high tide in the Hudson River.

The severest flooding condition revealed by the study results from the simultaneous occurrence of a standard project flood, a failure of the Ashokan Dam and a storm surge in New York Harbor at the mouth of the Hudson River resulting from a standard project hurricane. The water level under these conditions would reach 14-ft above MSL. Local wave action due to wind effects has been determined to add 1-ft to the river elevation producing a maximum water elevation of 15-ft above MSL at the Indian Point Site. Since this maximum water elevation is 3-in. lower than the critical elevation of 15-ft 3-in. noted earlier, it is reasonable to conclude that flooding in the Hudson River will not present a hazard to the safe operation of Indian Point.

The three most severe hurricanes to hit New York Harbor (September 21, 1821; November 25, 1950 (mentioned previously); and September 12, 1960) produced tidal surges at the Battery of 11-ft, 8.2-ft and 6.3-ft, respectively. Accordingly, these surges would appear as 7.5-ft, 5.5-ft, and 4.3-ft surges at Indian Point. The 5.5-ft surge predicted for the November 25, 1950, hurricane agrees well with the actual surge that produced the 7.4-ft-high watermark recorded for Indian Point on that date.

The Quirk, Lawler and Matusky report indicated that the combination of the maximum probable runoff, upstream dam failures and maximum ebb tide in the Hudson River is a less severe condition than the one postulated above. This latter scenario would cause the water level at Indian Point to be 11.7-ft above MSL, also below the critical control elevation.

The report also indicates that the combination of probable maximum hurricane, spring high tide, and wave run-up will cause the water level at Indian Point to reach 14.5-ft above MSL. This is also below the critical control elevation of 15-ft-3-in. Table 2.5-1 summarizes the Indian Point water surface elevations resulting from the various combinations.

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In view of the recorded hydrologic history of the Hudson River and New York Harbor and the predicated maximum hurricane surge at Indian Point, flooding at the site is a highly unlikely possibility.

Within a 5-mile radius of the facility only one municipal water supply uses ground water. Other wells in this area are used for industrial and commercial purposes. The rock formations in the area and elevations of wells relative to the facility are such that accidental ground leakage or seepage percolating into the ground at Indian Point will not reach these sources of ground water, but will flow to the river.

Only two reservoirs within a 5-mile radius are used for municipal water supplies. The first, Camp Field Reservoir, is the raw-water receiving basin for the system, which serves the city of Peekskill. This system uses the Catskill Aqueduct and Montrose Water District as alternative sources of water supply. The second reservoir, the impounding reservoir for the Stony Point water system, serves the towns of Stony Point and Haverstraw, and the villages of Haverstraw and West Haverstraw. The Stony Point system is connected to the Spring Valley Water Company to provide an alternative source of supply. A third reservoir within 5-miles of the facility, Queensboro Lake, supplies water to a state park area only. The location of these reservoirs, and others within a 15-mile radius of the site, are shown on Figure 2.5-2. The city of New York's Chelsea Pumping Station is located about 1-mile north of Chelsea, New York, on the east bank of the Hudson River, about 22-miles upstream of the site. Water will be pumped from intakes in the river at the rate of up to 100 million gal per day into the city reservoir system as required to supplement the primary supply from watersheds during severe drought conditions. This source, however, was not used during the recent 1981 drought.

The discharge of any contaminant into a tidal estuary will result in its distribution throughout the estuary. Factors affecting this distribution include tide amplitude and current, river geometry, salinity distribution, and freshwater discharge. Quirk, Lawler, and Matusky investigated for Con Edison the influence of these factors and determined the effect of radioactive discharges on overall river concentrations, and specifically conditions at Chelsea Pumping station, as discussed in Section 4.1. During the conduct of normal activities, the facility discharge will not exceed its maximum permissible concentration. Compliance with regulatory release limits is further discussed in Section 4.1.

### REFERENCES FOR SECTION 2.5

1. Letter report of Karl L. Kennison, Civil and Hydraulic Engineer, to G. R. Milne, Con Edison, November 28, 1955.
2. Metcalf and Eddy Engineers, Hydrology of Indian Point Site and Surrounding Area, October 1965.
3. Quirk, Lawler, and Matusky Engineers, Transport of Contaminants in the Hudson River above Indian Point Station, May 1966.
4. Quirk, Lawler, and Matusky Engineers, Evaluation of Flooding Conditions at Indian Point Nuclear Generating Unit No. 3, April 1970.
5. GZA, Hydrogeologic Site Investigation Report for the Indian Point Energy Center, January 7, 2008.

## IP2 DEFUELED SAFETY ANALYSIS REPORT

**TABLE 2.5-1**  
**Water Surface Elevation at Indian Point**  
**Resulting from Stated Flow and Elevation Conditions**

<u>Component Flow at Indian Point</u>	<u>Elevation at the Battery - Datum MSL (ft)</u>	<u>Flow at Indian Point (million cfs)</u>	<u>Elevation at Indian Point - Datum-MSL (ft)</u>	<u>Elevation at Indian Point Including Local Oscillatory Wave Height Datum MSL (ft)</u>
1. Probable maximum flood	MSL 0.00	1.100	12.7	13.7
2. Probable Maximum flood and tidal flow	High water ±2.20	1.014	12.4	13.4
3. Probable Maximum flood and tidal flow	Low water -2.20	1.165	13.0	14.0
4. Standard project flood and Ashokan Dam failure	MSL 0.00	0.705	7.2	8.2
5. Standard project flood	Standard project hurricane +11.00	0.550	13.0	14.0
6. Standard project flood and Ashokan Dam failure	Standard project hurricane (+11.00)	0.705	14.0	15.0
7. Probable maximum hurricane and spring high tide	Probable maximum hurricane +17.5		13.5	14.5

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### 2.5 FIGURES

Figure No.	Title
Figure 2.5-1	Map & Description Showing Location of Sources of Potable & Industrial Water Supplies & Watershed Areas
Figure 2.5-2	Hudson River Drainage Basin

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### 2.6 Meteorology

#### 2.6.1 General

[Historical Information] Meteorological parameters related to atmospheric transport and diffusion have been extensively investigated in the Indian Point area since 1955. Studies of the wind flow characteristics, induced by the topography surrounding the site, illustrated the unique valley wind system and the channeling of low-level winds.

Meteorological studies<sup>1-3</sup> were conducted from 1955 to 1957 by the Research Division of New York University, under the direction of Prof. Ben Davidson, in support of Unit 1 licensing activities. Data from these studies illustrated the channeling of the air flow by the terrain into downvalley (north-northeast) and upvalley (south-southwest) regimes. Historical data collected by the U.S. Weather Bureau in 1932 also illustrated the valley wind system.

Subsequent meteorological investigations were conducted from 1968 through 1972 by New York University School of Engineering and Science, Department of Meteorology and Oceanography, under the direction of Dr. James Halitsky and Mr. Edward J. Kaplin. These studies supported the earlier findings of the valley wind system by Prof. Davidson and are documented in Appendix 2A of this DSAR and in the FSAR for Indian Point Unit 3.

The most recent meteorological programs and data analyses conducted in the Indian Point environs since 1972 were documented in a York Services Corporation report (Meteorological Update, September 1981). This report is included in Appendix 2A. The 10-m elevation on the 100-ft meteorological tower used for the Unit 2 siting studies is the backup tower to the 400-ft (122-m) primary tower. The 10-m tower installed at the Buchanan Service Center is also available as an additional contingency.

The York Services Report summarizes the meteorological activities conducted for Indian Point from 1955 to 1981. Included are topographic effects, wind correlations, data collection, diurnal wind distribution, trajectory analyses, atmospheric stability, and wind distributions. The report substantiates previous studies conducted on the existence of the valley wind system in the environs of Indian Point.

#### 2.6.2 Historical Information - Application of Site Meteorology to Safety Analysis of Loss-Of-Coolant Accident

Section 6.2.1.4 describes the application of meteorology data to the analysis of the Fuel Handling Accident. The following information is retained as historical information.

The atmospheric dispersion factors required for the original safety analysis were computed for the worst possible meteorological conditions that could prevail at the Indian Point site.

A search of the records indicated that the most protracted consecutive period during which the wind direction was substantially from the same direction was 5 days. The winds in this case were from the northwest and speeds ranged from 15 to 30 mph.

The most frequent wind flow at low heights under inversion conditions was down the axis of the valley. This direction, roughly 10- to 30-degrees, was also the direction of maximum wind frequency. Because of the relatively high frequency of inversion conditions associated with this wind direction, the original safety analysis assumed that the distribution of wind speed and thermal

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stability during the hypothetical accident was exactly that measured at the 100-ft tower level for the 5- to 20-degree wind direction.

The valley wind is diurnal in nature, that is, up-valley during unstable hours and down-valley during stable hours. In general, these local winds are most frequent under clear sky and relatively light prevailing wind conditions. The diurnal variation of the vector mean wind as measured 70-ft above river during September-October 1955 is shown in Figure 2.6-1 for conditions in which the large-scale flow was virtually zero (12 days) and in Figure 2.6-2 for conditions in which the large-scale flow (geostrophic wind) was less than 16 mph (35 days). It may be seen that for these virtually stagnant prevailing wind conditions, there is a regular diurnal shift in wind direction and that the mean vector wind associated with the down-valley flow is approximately 6 mph.

A measure of the magnitude of the diurnal shift in wind direction is shown in Figure 2.6-3, where the steadiness of the wind (vector) mean speed over the mean scalar speed is shown as a function of time and the strength of the prevailing flow. Where the steadiness is close to one, the persistence of a given wind direction is very high. These data indicate that a consecutive 24-hr down-valley flow with light wind speeds and inversion conditions is extremely improbable due to the diurnal variation of the steadiness.

The original safety analysis of the loss-of-coolant accident assumed that the accident occurred during down-valley inversion flow conditions and that this condition persisted for 24 hours with average wind speeds slightly less than 2 m/sec. Figures 2.6-1 and 2.6-2 indicate that the duration of the down-valley flow is about 12-hr rather than 24-hr and that the vector mean wind speeds are approximately 2.5 m/sec.

In view of the discussion above, it was concluded that the original safety analysis for the first 24-hr is conservative to within a factor of about 2.

The remainder of the original safety analysis assumed that for the next 30 days, 35-percent of the winds were in the 20-degree sector corresponding to the nocturnal down-valley flow and that wind speed and thermal stability were as observed over the period of 1 year as measured at the 100-ft tower location. If the observations were distributed uniformly throughout the year, slightly over 100 hr per month of 5- to 20-degree winds could be expected to occur. The original analysis assumed that 276-hr of 5- to 20-degree winds occurred in the first 31 days after the accident, and that about 130 of these hours were characterized by inversion conditions. Approximately 35 weak-pressure gradient days were observed in September-October 1955 or about 430 hr per month. From Figure 2.6-3, the hour during which the down-valley flow is quite persistent under weak-pressure gradient conditions are from 0 to 8 hr. Assuming that the steadiness is 1.0 during these hours (it is in fact about 0.9 or less), the number of down-valley inversion winds per month during September and October is on the order of 140 hr per month. This indicates that the meteorology assumed in the original safety analysis beyond the first 24-hr was reasonable for the worst months (September and October) and was undoubtedly conservative with varying degrees of conservatism for the remainder of the year.

The inversion frequency assumed for the original 30-day accident case was conservative because the evaluation was made from concurrent assumptions concerning the postulated meteorological conditions, namely:

1. Inversion conditions prevail for 42.4-percent of the time.
2. The wind direction is within a narrow 20-degree sector for 35-percent of the time.



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This was equivalent to assuming that in the model 20-degree sector, the inversion frequency is 14.8-percent for the 30-day period. The observed annual maximum inversion frequency for a 20-degree sector is 6.2-percent (p. 29, Table 3-3, Section 1.6 of Reference 3). If we assumed that the inversion frequency is spread uniformly throughout the year, almost 3 months worth of inversions in the model 20-degree sector were considered to occur in the first 31-day month after the original analyzed accident. The assumption of uniform spread of inversion frequency over the year was examined above where an attempt was made to isolate those local meteorological conditions at Indian Point, which might yield the highest 30-day dose. It was concluded that the "worst" meteorological conditions were associated with the nocturnal down-valley flow, which was most frequent during September and October.

### REFERENCES FOR SECTION 2.6 [HISTORICAL INFORMATION]

1. New York University, Research Division, A Micrometeorological Survey of the Buchanan, N.Y., Area, NYU Technical Report 372.1, November 1955, which was Exhibit L-1, Docket 50-3, given in its entirety. The topography of the area surrounding the site is described and the effects of the topography on meteorological conditions are discussed. The types of data collected, the methods and frequencies of collection, the description and location of the equipment, and the general scope of the meteorological program are indicated in this report. Seasonal wind characteristics, including speeds, directions, and frequencies are tabulated.
2. New York University, Research Division, Evaluation of Potential Radiation Hazard Resulting From Assumed Release of Radioactive Wastes to the Atmosphere From Proposed Buchanan Nuclear Power Plant, Sections 1, 2, and 3 of NYU Technical Report 372.3, April 1957. This report was submitted to the NRC in its entirety as Exhibit L-5, Docket 50-3. These sections discuss the diffusive conditions and the climatological data of the site. The basis for evaluating the diffusion parameters selected for the safety analysis is given on pages 19 to 21. Section 3 contains tables of frequency distribution of diffusion classes and wind directions, and also wind roses.
3. New York University, Research Division, Summary of Climatological Data at Buchanan, N.Y., 1956-1957, NYU Technical Report 372.4, March 1958, was Exhibit L-6, Docket 50-3. This report summarizes the final meteorological testing at Indian Point.

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### 2.6 FIGURES

Figure No.	Title
Figure 2.6-1 [Historical Information]	Diurnal Variation of Mean Vector Wind for Virtually Zero Pressure Gradient Conditions
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Figure 2.6-3 [Historical Information]	Steadiness of Wind as a Function of Time of Day for Indicated Pressure Gradient Conditions

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### 2.7 Geology and Seismology

[Historical Information] The Indian Point site and surrounding area were studied in 1955 by Sidney Paige, consulting geologist, before the construction of Unit 1. In 1965, Thomas W. Fluhr, P.E., an engineering geologist, reviewed the geology of the site and made additional borings at the location of Unit 2.

In 1982, a report by Woodward-Clyde Consultants was done to update this Section. The previous studies are listed in the reference list of the report. The report is included in Appendix 2B.

Between 2005 and 2007, GZA GeoEnvironmental (GZA), performed a comprehensive hydrogeologic investigation of the site. This investigation was initiated to understand groundwater flow and contaminant transport. During this investigation numerous borings were advanced to study the site geology, hydrology and aquifer properties. Details of the geology, hydrology and aquifer properties can be found in the GZA report<sup>1</sup>.

A seismic monitoring network exists in the vicinity of the site and data from this network is periodically evaluated.

#### REFERENCES FOR SECTION 2.7

1. GZA, Hydrogeologic Site Investigation Report for the Indian Point Energy Center, January 7, 2008.

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### 2.8 Environmental Radioactivity

Monitoring for environmental radioactivity in the vicinity of the Indian Point Station began in 1958, approximately 4 years before the operation of Unit 1. Measurements since that time have indicated that the present operation of Units 2 and 3 and the past operation of Unit 1 have had no significant effect on the environment. The 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. Measurements of radioactivity in the environment are summarized in the Annual Radiological Environmental Operating Report, which is submitted annually as required by the facility's Technical Specifications.

Determinations of radioactivity in the environment are made regularly. Sample locations are defined in the Offsite Dose Calculation Manual (ODCM).

The overall objectives of the environmental monitoring program are as follows:

1. To establish a sampling schedule for Indian Point Units 1 and 2 that will recognize changes in radioactivity in the environs of the facility.
2. To ensure that the effluent releases are kept as low as is reasonably achievable (ALARA) and within allowable limits in accordance with 10 CFR 20.
3. To verify projected and anticipated radioactivity concentrations in the environment and related exposure from releases of radioactive material from the Indian Point site.

Results of environmental surveys conducted by Con Edison have been verified by the Bureau of Radiological Health Service of the New York State Health Department in previous years and presently, by the New York State Bureau of Environmental Radiation.<sup>1, 2</sup>

Environmental surveys have also been confirmed by Dr. Merrill Eisenbud, Director of Environmental Radiation Laboratory, Institute of Environmental Medicine, New York University Medical Center, who has found that the levels of environmental radioactivity are associated with natural background and fallout of nuclear weapons testing.<sup>3</sup>

In a study of the radioactivity in the Hudson River, Mr. Sherwood Davis, Director, Bureau of Radiological Health Service, New York State Department of Health, et al., have concluded that the discharges from Indian Point Unit 1 "are a minute fraction of the federal limits."<sup>4</sup>

The above results were obtained in preoperational and operational periods of Units 1 and 2 in the late 1950s and in the 1960s. In the more recent years of the late 1970s, radiological impact evaluations have shown similar results. These evaluations of actual historical Unit 2 releases have been performed for inclusion in the effluent release reports and have shown that operation of the Unit 2 facility has had an insignificant impact on the environs.

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### REFERENCES FOR SECTION 2.8

1. New York State Department of Health, Division of Environmental Health Services, Consolidated Edison Indian Point Reactor, Post Operational Survey, August 1965.
2. New York State Department of Health, Division of Environmental Health Services, Consolidated Edison Indian Point Reactor, Post Operational Survey, July 1966.
3. New York University Medical Center Institute of Environmental Medicine, Ecological Survey of the Hudson River: Progress Report No. 2, submitted to Division of Radiological Health, USPHS, Contract PHS 86-95, Neg. 141, December 1966.
4. F. Cosolito, et al., Radioactivity in the Hudson River, Symposium on Hudson River Ecology, Hudson River Valley Commission of New York, October 4-5, 1966.

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**NOTE: This information is classified as Historical Information**

## APPENDIX 2A

### FACILITY SAFETY ANALYSIS REPORT (FSAR)

CONSOLIDATED EDISON COMPANY  
OF NEW YORK, INCORPORATED

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

METEOROLOGICAL UPDATE

SEPTEMBER, 1981

YSC PROJECT + 01-4122

prepared by:

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ATMOSPHERIC SERVICES DEPARTMENT

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## IP2 DEFUELED SAFETY ANALYSIS REPORT

### FACILITY SAFETY ANALYSIS REPORT (FSAR) CONSOLIDATED EDISON COMPANY OF NEW YORK, INC. INDIAN POINT NUCLEAR GENERATING UNIT 2 METEOROLOGICAL UPDATE

#### 1.0 GENERAL

#### 1.1 HISTORICAL BACKGROUND

##### 1.1.1 Introduction

Meteorological data were initially collected and evaluated with respect to Indian Point, Buchanan, New York and its environment during the period from 1955 through 1957. This work was accomplished under the direction of Professor Benjamin Davidson of New York University under contract with the Consolidated Edison Company of New York, Inc. (Con Edison). The data and studies during this period were the bases for the Environmental Reports relevant to Indian Point Nuclear Generating Units 1, 2 and 3.

With respect to the Facility Safety Analysis Report (FSAR) for Unit 2, the Environmental Report Supplement, Appendices Volume 1 as Appendices C, D and E contains:

- NYU Technical Report 372.1 (November, 1955), B. Davidson
- NYU Technical Report 372.3, Section 2 and 3, (April, 1957), B. Davidson and J. Halitsky
- NYU Technical Report 372.4 (March, 1958), B. Davidson

In 1968 under the direction and supervision of Dr. James Halitsky of New York University, Con Edison contracted to establish experimental meteorological monitoring, data collection and evaluation activities at the Indian Point site and at specified sites in its environment (Halitsky, Laznow and Leahy, February 1970). The original purpose of the above investigation, as noted in the reference, was modified after the studies had begun in order to provide the AEC Construction Hearings for Indian Point Unit 3 with clarification of aspects of the 1956-1957 meteorological data base for the Units 1, 2 and 3 diffusion models. This phase of data collection began in December, 1969.

A report dealing with the results of this latter phase (Halitsky, Kaplin and Laznow, NYU GSL Technical Report No. TR 7103, May 1971) appears as Appendix G in the FSAR Unit 2 Environmental Report Supplement Appendices Volume 1 and as Supplement 1 in the FSAR for Unit 3. The focus of the above report was to validate present site meteorology as representing no significant change in relation to site meteorology from the 1955-1957 period.

Data collection and evaluations continued under this program and a report was submitted by Kaplin and Laznow (1972) representing the data collection period from 1 January, 1970 through 31 December 1971. A copy of this report appears in FSAR for Unit 3 as Supplement 10 (January, 1973).

With respect to Indian Point on-site meteorological measurements, there were for the purpose of the reports that have been cited, three different meteorological towers at three different locations. These are specifically delineated in Figure 1 of Halitsky, Kaplin and Laznow (1971). The meteorological data collected during the 1955-1957 were from the 300 Foot Tower designated as IP1. Meteorological data collected and reported by Halitsky, Kaplin and Laznow (1971) and

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Kaplin and Laznow (1972) were from the 100 Foot Tower designated as IP3. The base of IP3 Tower was about 200 feet from the base location of the original IP1 Tower.

Using input data meteorological data from Indian Point Tower (IP3), Bowline Point and the Cape Charles along with sequences of upper air pilot balloon observations Kaplin, Laznow and Wurmbrand (1972) provided Con Edison with input information for the location of the 90-percent probability air monitoring sites, overlay patterns for the prediction of the distribution of gaseous releases and evaluation of the requirements of the AEC Safety Guide 23 (1972).

All data on the IP3 Tower were collected in accordance with U.S. AEC Safety Guide No. 23, On-Site Meteorological Programs as delineated 2/17/71. The IP3 Tower was maintained from March 1972 through December 1973 by York Research Corporation, Stamford, Connecticut under contract with Con Edison. The last formal report for this tower was prepared and submitted to Con Edison by Kaplin and Kitson (1974) for the 1973 data collection period.

In April of 1973, York Research Corporation under contract with Con Edison began work on the erection of an on-site 400 Foot Meteorological Tower approximately 1725 feet-S and 1750 feet-W of the IP3 Tower. The function of this tower was to develop micro-climatological data suitable for the design of cooling towers and the evaluation of their potential environmental impact on the Indian Point site and its environs. Concurrent studies were conducted to develop three dimensional aspects of the local valley flow using pibal balloons, constant level tetroons and balloon-sondes. In addition, a concurrent study was conducted to develop background levels of ambient air salt concentrations. The results of these studies were submitted in two reports: Kaplin, Kozenko and Kirshner (1974) and Kaplin, Kitson and Kozenko (1974). This latter report compared meteorological data from the IP3 Tower. At the conclusion of 1973, the primary source of reduced on-site meteorological data were from the IP4, 400 Foot Meteorological Tower. IP3 Tower systems maintenance was continued in accordance with Safety Guide 23 through October 1, 1976 and meteorological data were recorded on analog charts. Data collection was transferred to the IP4 Tower on July 1, 1976. The 400 Foot Tower (IP4) servicing, maintenance and data collection and data selective processing is on-going. Its systems have been updated to meet present requirements of NUREG-0654 Appendix 2 (1980) and proposed Revision 1 to NRC Safety Guide 1.23 (1980).

The continued maintenance services, etc., of the 400 Foot Meteorological Tower by York Research Corporation/York Services Corporation was continuous under contract with Con Edison until September 30, 1978, and from October 1, 1978 through the present time under contract with the Power Authority of the State of New York (PASNY).

In September, 1979, York Services Corporation under contract with Con Edison began a study of north to south surface air trajectories analyses and evaluations based on "real-time" wind data available from the Indian Point vicinity. This study incorporated local wind velocity data from the 400 Foot Meteorological Tower at Indian Point, the Orange and Rockland Utilities, 350 Foot Meteorological Tower in Haverstraw, New York, as well as from selected U.S. Department of Commerce, NOAA, Weather Stations. For this study, meteorological data were analyzed and evaluated for the period from August 1, 1978 through July 31, 1979. The final results of this study were presented in reports: Kaplin, E.J. and B. Wuebber, (1979) and Kaplin, E.J. (1979). As an outgrowth of these studies an expended network of surface (10M) wind velocity monitoring stations were sited at key locations along the Hudson River Valley north and south of Indian Point and inland to the east. This network consisted of new anemometer stations in addition to the Indian Point 400 Foot Meteorological Tower, the Bowline 350 Foot Meteorological Tower and the U.S. Department of Commerce, NWS Station at Westchester County. These wind data were

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digitalized and evaluated by New York Services Corporation under contract to Con Edison for the purposes of defining surface air flow patterns within a 10-15-mile range of Indian Point with emphasis on generating refined estimated of southward movements. As completed, ten consecutive months (March 1, 1980 - December 31, 1980) were evaluated and a total of 7,264 eight-hour parcel trajectories were created objectively using appropriate local one-hour wind velocity averages on a real time basis. The results of this study were submitted to Con Edison: Kaplin, Edward J. and B. Wuebber (1981).

For the purpose of this FSAR, second meteorological update, meteorological data have been analyzed and evaluated from the 400 foot (122 Meter) Indian Point Meteorological Tower for the two year period from January 1, 1979 through December 31, 1980.

### 1.1.2 Tower Siting and Instrumentation

#### 1.1.2.1 Hudson River

There are a number of pertinent facts about the Hudson River itself that are relevant to its ability to induce and/or influence mesoscale flow phenomena that are dominant factors in the Indian Point environs. The most important factor is that it is not a river but, rather a tidal estuary. From New York City, 154-miles north to Troy, there is no drop in the surface elevation of the river. Except for spring runoff from the Andirondacks, which can smother the tide down to Albany, there is almost imperceptible downstream current.

Since there is no slope to the river surface, it will not support its own gravity flow. Any air movement within its canyons during minimal atmospheric pressure gradient periods can be strictly local cells, which may actually block continuous horizontal air movement over the water surface.

Thermally induced air movement of the Atlantic sea breeze follows the natural path of the river. It has been noted, however, that Iona Island 45-miles north of the tip of Manhattan is considered the point of maximum inland intrusion. The northward movement of sea breeze does not proceed up the Hudson River Valley and Hackensack River Valley at the same speed. The inland movement along the Hackensack Valley lags the Hudson Valley movement. The Hackensack River is on the west side of the Hudson and is specifically delineated because its headwaters are just south of the South Mountains and isolated from the Hudson River by the Hook Mountains and the Palisades. The South Mountains are the east-west extension of the Hook Mountains. The South Mountains about the Ramapo Range and form a sheer wall from the Southern boundary of the west bank community of Haverstraw.

#### 1.1.2.2 General Topography

Each of the reports cited in Section 1.1.1 describe in some detail general topography in the Indian Point environs. The most recent was provided by Kaplin and Wuebber, 1981. Indian Point is located in the lower Hudson River Valley 27-miles due north of northern boundary of New York City (Manhattan Island).

The Indian Point area has been described by Halitsky, et. al., 1970, as being located roughly on the axis of a north-south valley enclosed by the Dunderberg and Buckberg Mountains to the west and Blue Mountain and Prickly Pear Hill on the east. The shape of the valley at the 200 foot and 400 foot elevation levels are given in Figures 1 and 2. At the 200 foot contour level the valley width is two miles at Dunderberg Mountain and opens southward to a width of five miles at Prickly Pear Hill.

## IP2 DEFUELED SAFETY ANALYSIS REPORT

The Hudson River, flowing southward through the valley, resembles a gourd with its curved  $\frac{3}{4}$ -mile thick northern neck nestling at the base of Dunderberg Mountain while the bulbous three mile thick body fills the southern part of the valley between South Mountain and Prickly Pear Hill. The Indian Point peninsula lies in the hollow of the curved neck.

Beyond its northern end, the valley is split into two branches by Manitou Mountain. The Hudson River passes through the steep, narrow northwest branch between Manitou and Dunderberg Mountains. The northeast branch, between Manitou and Blue Mountains, is about 1.5-miles wide at Manitou Mountain but degenerates with distance into three tributary valleys containing Annsville Creek, Sprout Brook and Peekskill Hollow Brook with sources in the mountainous region north of Peekskill.

South of Haverstraw Bay, the valley opens up rapidly to the southeast while the west bank of the Hudson River follows the blocking of the east-west orientated South Mountains to assume a southward course along the Hook Mountains to the Palisades Mountains.

At elevations higher than 200 feet the solidity of the eastern wall of the valley breaks, first between Blue Mountain and Prickly Pear Hill to form at 300 feet the irregular drainage system, which supplies Furnace Brook, and then at 400 feet into an irregular array of mountain tops. The western wall is till fairly solid at 300 feet but breaks at 400 feet into two well-defined valleys containing Cedar Pond and Minisceongo Creeks. However, just to the west are Ramapo Mountains whose elevations exceed 1000 feet. Figures 3 and 4 (Kaplin and Wuebber, 1981) show the elevations of significant mountain peaks and water courses in the region of Indian Point.

### 1.1.2.3 Site Configuration

The location of the specific meteorological towers associated with earlier studies in the Indian Point environs are available in Figure 1 of Kaplin and Laznow (1972), FSAR 3, Supplement 10, 1973 and in Figures 1a and 1d of Kaplin, Kitson and Kozenko (1974).

Table 1 lists the operational periods and the instrumentation associated with each tower. Not included in this listing are those wind monitoring sites that were used for the most recent study (Kaplin and Wuebber 1981, Sec. 2.4 & 2.5). In this study, an 11 site monitoring network was established equipped with Climatronics Mark III, wind speed and direction systems. This network was operable for all or part of the period from March, 1980 through December, 1980.



## IP2 DEFUELED SAFETY ANALYSIS REPORT

TABLE 1  
TOWER AND INSTRUMENTATION RECORD  
(INCLUDES PARAMETERS NOT REQUIRED BY PROPOSED SAFETY GUIDE 1.23)

Meteorological Station	Base Elevation Ft. MSL	<u>Operational Period</u>		Parameter	Instrument	Exposure M. Above Grade
		From	To			
Indian Pt (IP1)	130	1956	1957	Wind Temp. Diff.	Aerovane Honeywell	91 & 30 91-2 & 30-2
USS Jones (J)	0	1956	1957	Wind	Aerovane	21
Indian Pt (IP2)	60	1968	1969	Wind Temp. Diff.	Climet Bristol	30 29 - 1.5
Montrose (MP)	60	1968	6/71	Wind	Climet	30
Bowline Pt (BP)	5	9/68	11/69	Wind	Climet	30
		11/69	8/72	Wind	Aerovane	30
		9/68	11/69	Temp. Diff.	Honeywell	30 - 3
		11/69	2/72	Temp. Diff.	Bristol	30 - 3
		2/72	8/72	Temp. Diff.	Climet- (Rosemont)	30 - 3
Bowline Tower	10	Note 2 Note 2	Present Present	Wind Temp. Diff.	Climatronics Climatronics	100, 50 & 10 100-10 & 50-10
Trap Rock (TR)	90	1969	7/72	Wind	Climet	30
USS Cape Charles (CC)	0	3/70	9/70	Wind	Aerovane	30
Indian Pt (IP3)	120	11/69	9/76	Wind	Aerovane	30
		11/69	9/76	Wind	Climet	30
Backup Met System (IP3)		6/73	9/76	Wind	Climet	10
		12/81	Present	Wind	Climatronics	10
		11/69	10/71	Temp. Diff.	Honeywell	29 - 2
		8/72	9/76	Temp. Diff.	Climet- (Rosemont)	30-3 & 9-3
		8/72	9/76	Amb. Temp.	Climet- (Rosemont)	9
		8/72	9/76	Dew Point	Climet- (Foxboro)	30,9 & 3
		8/72	9/74	Net Radiation	Teledyne Geotech	9
		5/70	12/70	Turbulence	Bivane*	30

Note 2 - Bowline Tower is located at approximately Latitude 41° 13'N and Longitude 73° 58' W. This location is about 3000 feet NW of the earlier Bowline Point Tower. It began operation in the 1972/73 time period.

\* Intermittent usage.

## IP2 DEFUELED SAFETY ANALYSIS REPORT

TABLE 1 (Cont'd)

Meteorological Station	Base Elevation Ft. MSL	<u>Operational Period</u>		Parameter	Instrument	Exposure M. Above Grade
		From	To			
Indian Pt (IP4) 122 M. Tower	117	9/73	Present	Wind	Climatronics	122, 38 & 10
		9/73	6/80	Wind	Climatronics	85 - Note 1*
		6/80	Present	Wind	Climatronics	60
		9/73	9/79	Amb. Temp.	EG & G	10
		9/79	Present	Amb. Temp.	Climatronics	10
		9/73	9/79	Dew Pt.	EG & G	122, 61 & 10
		9/79	Present	Dew Pt.	Climatronics	10
		9/73	9/79	Temp. Diff.	EG & G	122-10 & 61-10
		9/79	Present	Temp. Diff.	Climatronics	122-10 & 60-10
		1/74	Present	Net Rad.	Teledyne Geotech	10
		7/80	Present	Precipitation	Climatronics	1
Indian Pt (IP4) ** 10M Tower	117	9/73	7/77	Visual Range	EG & G FSM	10
Emergency Control Center #	135	7/24/80	11/81	Wind	Climatronics	11.8

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Note 1\* - 85 meter wind speed and wind direction moved from the 85 meter level to the 60 meter level as required by proposed Revision 1 to NRC Safety Guide 1.23.

\*\* Tower and System Removed 07/22/80.

# Tower and System Removed 07/22/80.



## IP2 DEFUELED SAFETY ANALYSIS REPORT

### 2.0 122M METEOROLOGICAL TOWER

#### 2.1.1 Siting

The relative locations of the existing meteorological towers in historic perspective are shown in Figure 5. Specific details of site location are shown in Figure 6.

#### 2.1.2 Instrumentation

##### 2.1.2.1 Sensor Configuration

The sensor configuration and exposure on the existing operational 122M Meteorological Tower are shown in Figures 7 and 8.

##### 2.1.2.2 Instrumentation Specifications

The following specifications apply to specific operational sensors that are a part of the total meteorological support systems at Indian Point.

###### 2.1.2.2.1 Climatronics F460 Wind Speed Transmitter

Accuracy:	0.07 M/S or 1%
Range:	0-56 M/S
Threshold:	0.22 M/S
Distance Constant:	1.5 M

###### 2.1.2.2.2 Climatronics F460 Wind Direction Transmitter

Accuracy:	$\pm 2^\circ$
Range:	0-540°
Threshold:	0.22 M/S
Distance Constant:	1.5 M
Damping Ratio:	0.4 at 10° initial angle of attack

###### 2.1.2.2.3 Climatronics TS-10 and TS-10WA Motor Aspirated Shields

Shield Effectiveness:	Under radiation intensities of 110 W/m <sup>3</sup> (1.6 cal/cm <sup>2</sup> /min) radiation error not exceeding 0.1°C
Aspiration Rate:	3 M/S at sensor location

###### 2.1.2.2.4 Climatronics 100087, 100087-3 - Temperature-Delta Temperature (matched thermistor)

Temperature:	
Range:	-34 to +50°C
Accuracy:	$\pm 0.2^\circ\text{C}$
Time Constant:	10 sec. To 63% (in TS-10 Shield)
Linearity:	$\pm 0.2\%$

Delta Temperature:

## IP2 DEFUELED SAFETY ANALYSIS REPORT

Range:	$\pm 10^{\circ}\text{F}$
Sensitivity:	$0.02^{\circ}\text{F}$
Accuracy:	$0.1^{\circ}\text{F}$ or $\pm 5\%$ of delta-T not to exceed $0.3^{\circ}\text{F}$
Response Time:	10 sec. To 63% in TS-10 Shield

### 2.1.2.2.5 Climatronics DP-10 Dew Probe (YSI Lithium Chloride)

Range:	$-40^{\circ}$ to $42^{\circ}\text{C}$
Accuracy:	$\pm 0.5^{\circ}\text{C}$
Response Time:	$1^{\circ}\text{C}/\text{min.}$

### 2.1.2.2.6 Climatronics 1000971 - Heated Rain - Snow Gauge (Tipping Bucket)

Accuracy:	$\pm 1\%$ up to $3''/\text{hr}$
Resolution:	$0.01''$
Size:	$8''$ diameter x $24''$ height
Conversion Accuracy:	$\pm 0.2\%$

### 2.1.2.2.7 Data Collection Systems

#### Analog:

Wind Systems:	Esterline-Angus Model E1102R - Rectigraph Recorders Temperature, Dew Point, Delta Temperature: Tracer Westronics Model M11E, Multipoint
Precipitation:	Esterline-Angus Model MS 401C

#### Digital:

Climatronics Data Processor:	1MP/801
Tape Collection Interface:	Tandeberg TD1 10-50

### 2.1.3 Meteorological Support System

The meteorological systems at Indian Point are equipped, maintained and operated in compliance with the specification of NUREG-0654, Appendix 2 (1980); Proposed Revision 1 to NRC Regulatory Guide 1.23 (1980); and applicable regulatory requirements. The total system as presently operated is outlined in Figure 9.

## 2.2 DATA LOG

### 2.2.1 Indian Point Tower IP3

Meteorological data from the IP3 Tower were reduced and evaluated (Kaplin and Kitson, 1974) through December 1973. From the period 1974 through September 1976 the tower system was maintained, as previously noted. Analog records were provided to Con Edison for storage. Tower removed from service in September, 1976. (Reactivated as site for Backup Wind System: 12/01/81.)

### 2.2.2 122 Meter Meteorological Tower (IP4)

## IP2 DEFUELED SAFETY ANALYSIS REPORT

The data log for the 122 Meter Meteorological Tower for the period from October 1973 through August 1974 can be found in Kaplin, et. al., 1974. Subsequent to the completion of the above report and the data contained therein, the meteorological analog charts and the data collection magnetic tape were documented and transmitted to Con Edison for storage.

Commencing in August 1977, wind velocity data at the 10 and 122 meter levels and the delta temperatures: 60-10M and 122-10M were reduced to hourly averages and transmitted to Con Edison in addition to the analog charts. The summary of valid data for these parameters for the period from August 1977 - July 1981 is shown in Table 2 on concurrent and total hours basis. The concurrent basis assumes that if any parameter is missing. The total basis relates an individual missing data hour to the total number of possible data hours in a month.

On the concurrent basis, the average valid data collection was  $92.4 \pm 10.8$ -percent. On a total hour basis, the average valid data collection was  $98.2 \pm 2.5$ .

# IP2 DEFUELED SAFETY ANALYSIS REPORT

TABLE 2  
VALID DATA LOG\*

Month	1977		1978		1979		1980		1981	
	Concurrent	Total	Concurrent	Total	Concurrent	Total	Concurrent	Total	Concurrent	Total
January	N/A	N/A	89.8	98.2	94.8	99.0	98.4	99.7	99.9	99.9
February	N/A	N/A	92.1	97.4	98.1	99.7	90.5	98.4	67.7	89.2
March	N/A	N/A	95.1	98.9	97.4	99.6	97.3	99.1	98.0	99.3
April	N/A	N/A	98.1	99.6	96.7	99.4	100.0	100.0	100.0	100.0
May	N/A	N/A	95.3	98.6	90.3	98.0	88.4	98.0	100.0	100.0
June	N/A	N/A	86.8	95.6	95.0	99.1	96.7	99.0	100.0	100.0
July	N/A	N/A	94.1	96.8	92.6	98.8	71.1	94.1	99.4	99.8
August	94.2	98.7	95.3	99.2	52.3	92.0	77.7	92.7		
September	84.7	94.4	99.7	99.9	57.5	92.3	98.3	99.7		
October	95.0	98.9	98.1	99.4	94.0	98.2	96.1	99.4		
November	98.9	99.5	98.8	99.6	98.3	99.7	99.6	99.9		
December	75.3	95.5	96.4	99.4	100.0	100.0	98.9	99.8		

Concurrent Average: 92.4 ± 10.8%  
Total Parameter Hours: 98.2 ± 2.5%

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\* Based on six (6) parameters: wind data at 10 and 122M and delta-temperatures: 60-10M and 122-10M  
N/A - Values not available.

## IP2 DEFUELED SAFETY ANALYSIS REPORT

### 3.0 ANALYSES DATA

#### 3.1 INDIAN POINT TOWER IP3

A FSAR documented study with respect to the Indian Point IP3 Meteorological Tower was prepared by Kaplin and Laznow (1972) [Referenced FSAR 3 Supplement 10, 1973]. This report covered the data collection period through 1971. Additional data were subsequently provided through 1972 to provide composite joint wind velocity frequency distribution for Pasquill Stability Categories (FSAR 3 Supplement 13 and 16, 1973). Kaplin and Kitson (1974) provided an analyses of IP3 for the period March 1973 through December, 1973.

This report confirmed the earlier study that wind data in the Indian Point environs based on monthly diurnal wind distributions, wind frequency distributions and joint wind stability categories are comprised of two "seasons" with little apparent transition. The "winter season" reflects little or no average diurnal variation in the hourly resultant winds, dominant winds from the west to north. The "summer season" is characterized by dominant north-northeast winds during the evening and early morning hours with a sharp transition to south to southwest winds during the day and another transition in late afternoon to the evening pattern.

The wind frequency distributions and joint frequencies as a function of Pasquill Stability Categories were comparable in 1973 with data collected in 1970 and 1971.

It is noted that the temperature gradients on the IP3 Tower were derived from delta-temperatures: 99-7 feet and wind measured at 105 feet above a grade elevation of 120 feet MSL.

Kaplin, et. al., (1974) compared three months (October - December, 1973) of IP3 wind data as measured at 105 feet above grade with concurrent three months of data from the 125 foot level on the 122 Meter Meteorological Tower (IP4) (grade elevation: approximately 117 feet MSL) using a two station wind correlation program (Appendix B, Kaplin, et. al, 1974).

Figures 10A and 10B show the relationships obtained for October and December, 1973. The November results were similar with the directional relationships falling between that obtained for October and December. The maximum variations between the two sites occurred with winds from E-ESE and SW to WSW. These corresponded to sectors of minimum average wind speeds. Deviations between the two sites can be attributed to local factors including terrain elevation, land use and ground cover.

The wind direction displacement effects found in the two station correlations were confirmed in the monthly diurnal analyses.

#### 3.2 122M METEOROLOGICAL TOWER (IP4)

##### 3.2.1 October 1, 1973 to August 31, 1974

The purpose of the 122M Meteorological Tower at Indian Point (IP4) was to develop a three dimensional micro-climatological data file to be used to assess the impact of proposed cooling towers and to provide the basis for design criteria as required. The results of one year of operation of this tower were presented in a final report (with Appendices) by Kaplin, et. al., 1974.

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Except as noted in the previous section, the meteorological data collected and evaluated were not compared at the time of this study with historical meteorological data associated with FSAR 2.

This study determined that the two distinctive seasonal patterns existed at each of the four levels of wind velocity measurement: 10M, 38M, 85M and 122M. Wind directions tended to back with elevation assuming an orientation parallel to general terrain contours.

In times of weak synoptic pressure gradient patterns, there were abrupt transitions in the diurnal flow patterns consistent with valley flow winds particularly during the summer season. These transitions began at the surface and progressed up to the 122M level.

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TABLE 3  
COMPARISON OF ANNUAL PERCENT  
OCCURRENCE OF STABILITY CATEGORIES

<u>Year</u>	<u>Tower</u>	<u>Gradient (M)</u>	<u>Stability Category</u>						
			<u>A</u>	<u>B</u>	<u>C</u>	<u>D</u>	<u>E</u>	<u>F</u>	<u>G</u>
1970	IP3	29 - 3	21.68	2.20	3.39	33.35	24.75	9.01	5.62
1971	IP3	29 - 3	19.17	2.75	2.97	22.74	30.87	11.69	9.75
1970-72	IP3	29 - 3	16.25	1.82	2.95	29.71	26.61	13.27	9.45
1970-72*	IP3	29 - 3	6.76	2.67	2.13	32.65	40.57	11.78	3.31
1973	IP3	29 - 3	23.14	3.16	3.70	20.87	25.02	13.89	10.23
1973-74	122M-IP4	60 - 10	10.35	3.21	2.94	25.38	44.86	11.35	1.91
1979	122M-IP4	60 - 10	12.27	3.25	3.86	29.30	40.39	8.83	1.31
1980	122M-IP4	60 - 10	13.32	4.06	4.60	29.81	33.97	11.34	2.07
1979-80	122M-IP4	60 - 10	12.80	3.66	4.23	29.56	37.17	10.08	1.69

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\* Temperature difference corrected by a factor of 0.605; (FSAR 3, Supplement 16, April 1973)

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The morning transition during the summer season was sharply defined. At 0800 EST all levels had approximately the same resultant wind direction. The evening transition began just after 1800 EST and was sharply defined at the 10M and 38M levels. The 10M level reached its nocturnal northeast drainage wind by 2100 EST along with the somewhat more erratic 38M level. The 85M and 122M levels rotated systematically and did not reach their nocturnal directions (NNE-N) until 0200 EST. The systematic rotations are referenced to the average diurnal distributions on a real time basis, the upper wind levels could be "disconnected" from the lower wind levels with an intermediate shear zone generated by winds up to 180° out of phase.

The resultant winds of the 122M Tower (IP4) associated with the diurnal variation curves for the summer season, veered after the morning transition until about noontime, then steadied out at SW-WSW before backing into the nocturnal pattern after the evening transition. The resultant summer season winds for the IP3 Tower (Kaplin and Laznow, 1972) in 1970 and 1971 veered throughout the entire day. In Kaplin and Kitson (1974), the summer IP3 diurnal resultant winds exhibited the veering - backing trait. Kaplin and Laznow (1972) indicated that the question of backing or veering was related, on any given day, not only to strength of the valley drainage flow wind but also to relative strength of local land-sea circulations.

On an annual basis there were no significant differences between the percent frequency distribution of occurrences of stability categories (Pasquill) between the adjusted composite year 1970-1972 for IP3 (FSAR 3, Supplement 16, April 1973) and lower temperature gradient level on the 122M Tower (IP4). These comparisons are shown in Table 3. It is presumed that if all individual years of IP3 data were similarly adjusted prior to classification, they would also be reasonably comparable to results based on the temperature gradient 60-10M on the 122M Tower.

### 3.2.2 August 1, 1978 - July 31, 1979

Wind velocity data (10M level) from the 122M Meteorological Tower at Indian Point and the Orange and Rockland Utilities, Inc. 100M Meteorological Tower were used to evaluate the path of air parcels in the Indian Point environs without considering stability (Kaplin and Wuebber, 1979 and Kaplin, 1979).

Each hour a parcel movement was initialized from Indian Point. Each parcel was projected forward for eight consecutive hours in hourly increments. The average wind velocity at 10M level of the 122M Tower was used to determine the speed and direction of the parcel for its initial hour increment. Subsequent movement of each parcel was determined by the location of the parcel after the initial hour on a zone of influence file that assigned a wind vector to that location: Indian Point or Bowline.

Prior to usage the wind velocity for selected 1978-1979 data were assessed by comparison with historical data files (1973-1974) at Indian Point. There were no variations that could not be accounted for by climatological variations of at least synoptic scale when assessed with reference to U.S. Department of Commerce, NOAA, EDS, LOCAL CLIMATOLOGICAL DATA for LA GUARDIA AIRPORT, NEW YORK and SIKORSKY AIRPORT, Bridgeport, CONNECTICUT.

In considering persistent southward movement of an air parcel from Indian Point assuming that Bowline would be representative of air movement south of Grassy Point, an examination of resultant winds for August 1978 and January, 1979 (typical "summer" and



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"winter" seasons) indicated that such movements did not occur. While concurrent hourly average north winds were found 14 times in August and 17 times in January, these occurrences represent only 13.3-percent and 16.5-percent of all north winds relative to all north winds at Bowline. These results were anticipated, particularly during periods of light winds and weak synoptic pressure fields, from the opposing patterns of the diurnal variation curves for the two monitoring sites.

### 3.2.2.1 Surface Air Trajectories Analyses - Summary

Trajectory end points were derived on an objective basis using surface wind data from monitoring stations. The use of observed wind data appropriate to the moving air parcel's location at a given time is important since these data inherently account for local wind pattern aberrations that may be of topographic and/or unique micro-meteorological origin. No individual atmospheric stability category was explicitly considered.

The ability of the derived trajectories to generate realistic movement patterns is contingent on having sufficient wind monitoring sites to define the actual wind flow field in and around the area of interest on a concurrent real time basis.

The area of interest was limited to ten miles south of Indian Point. For practical purposes the study area was 21 x 21 square miles subdivided in a one mile grid as shown in Figure 11. Indian Point was located near the top center of the area at grid point 10, 16. This allowed for 15-miles of due south movement. The South, Hi Tor and Hook Mountains are emphasized because of the barrier that they form for air movement due south of Indian Point.

Trajectories were generated for each of the 12 months in the data file based on Indian Point and Bowline wind data. These were the only available data applicable to the study area. Trajectories were created for up to eight consecutive hours of movement.

For the first hour of the trajectory, Indian Point was used as the origin of an air parcel, which would travel a distance and in a direction determined by the hourly average wind velocity at the 33' (10M) level of the Meteorological Tower. Subsequent movement depended on the location of the trajectory end point after the first hour's movement. Zones for which the Bowline and Indian Point wind velocity measurements were considered as representative had been previously assigned. Trajectories for each hour of the month were computed. The end points were accumulated as summations of occurrences in their appropriate grid squares. In the process of generation, all end points were moved and accumulated whether or not they were in the 21 x 21-mile square. Only those end points within the study area boundary appear on tabular printouts. In any given period, an end point could pass out of the grid and move back in at a subsequent time interval.

For August, 1978, and January, 1979, two different patterns of weather station representative areas were used as shown in Figure 12. Pattern 1, which was used for all 12 months of data, had Bowline winds dominating after passage of a line three miles south of Indian Point (through Grassy Point). Pattern 2, used for August and January only, moved this line one mile further north (through Stony Point). The influence patterns are the same for the first hour's movement.

A summary of the August, 1978 and January, 1979, results in terms of percent of total possible trajectory end points remaining in the 21 x 21-mile area for selected trajectory time periods is shown in Table 4.

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TABLE 4  
SUMMARY OF TRAJECTORY END-POINTS

Pattern I	August, 1978		January, 1979	
	No. of Occur.	% Total	No. of Occur.	% Total
Hour 1	722 (729)	99.0	707 (721)	98.1
Hour 2	617 (728)	84.8	486 (720)	67.5
Hour 4	420 (726)	57.9	225 (718)	31.3
Hour 6	312 (724)	43.1	157 (716)	21.9
Hour 8	206 (722)	28.5	113 (714)	15.8
Pattern II				
Hour 1	722 (729)	99.0	707 (721)	98.1
Hour 2	595 (728)	81.7	486 (720)	67.5
Hour 4	414 (726)	57.0	226 (718)	31.5
Hour 6	308 (724)	42.5	157 (716)	21.9
Hour 8	207 (722)	28.7	115 (714)	16.1

---

( ) =Total number of trajectories generated.

The actual number of points within the grid network does not differentiate between those points that have never left the network and those that have recirculated. This feature takes on added importance if total distance of travel is a consideration.

Summaries of occurrences within designated grid sectors are shown in Table 5 for August, 1978 and Table 6 for January, 1979. In terms of totals in the grid area, there is no significant effect of influence pattern assignment. This effect does show up in Tables 5 and 6 when the occurrences south of Indian Point are totaled. For this purpose, a SW sector is defined encompassing the area below Indian Point from the grid edge to ordinate Line 9. The S sector extends one mile south of Indian Point along ordinate

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TABLE 5  
SUMMATION OF TRAJECTORY END POINTS  
AUGUST, 1978  
SECTOR KEY

17-21	NW	N	NE
16	W	I.P.	E
1-15	SW	S	SE
	1-9	10	11-21

								<u>% TOTAL</u>	
HR 1				35	164		133		46.0
				26	8		15		6.8
				<u>232</u>	<u>84</u>		<u>25</u>		<u>47.2</u>
% TOTAL				40.6	35.4		24.0		99.0
<u>PATTERN 1</u>						<u>PATTERN 2</u>			
				<u>% TOTAL</u>				<u>% TOTAL</u>	
HR 2	52	93	109	41.2		52	93	109	42.7
	20	10	7	6.0		18	10	7	5.9
	<u>192</u>	<u>51</u>	<u>83</u>	<u>52.8</u>		<u>148</u>	<u>50</u>	<u>108</u>	<u>51.4</u>
% TOTAL:	42.8	25.0	32.2	84.8	% TOTAL	36.6	25.7	37.6	81.7
				<u>% TOTAL</u>				<u>% TOTAL</u>	
HR 4	43	38	40	28.8		44	38	36	28.5
	7	2	2	2.6		7	1	2	24.4
	<u>128</u>	<u>27</u>	<u>133</u>	<u>68.6</u>		<u>111</u>	<u>29</u>	<u>146</u>	<u>69.1</u>
% TOTAL:	42.3	16.0	41.7	57.9	% TOTAL	39.1	16.4	44.4	57.0
				<u>% TOTAL</u>				<u>% TOTAL</u>	
HR 6	33	19	18	22.4		30	19	19	22.1
	7	0	1	2.6		6	0	1	2.3
	<u>100</u>	<u>10</u>	<u>124</u>	<u>75.0</u>		<u>86</u>	<u>17</u>	<u>130</u>	<u>75.6</u>
% TOTAL:	44.9	9.3	45.8	43.1	% TOTAL	39.6	11.7	48.7	42.5
				<u>% TOTAL</u>				<u>% TOTAL</u>	
HR 8	28	6	4	18.4		28	6	4	18.4
	3	0	1	1.9		3	1	1	2.4
	<u>67</u>	<u>7</u>	<u>90</u>	<u>79.6</u>		<u>53</u>	<u>15</u>	<u>96</u>	<u>79.2</u>
% TOTAL:	47.6	6.3	46.1	28.5	% TOTAL	40.6	10.6	48.8	28.7

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TABLE 6  
SUMMATION OF TRAJECTORY END POINTS  
JANUARY, 1979  
SECTOR KEY

17-21	NW	N	NE
16	W	I.P.	E
1-15	SW	S	SE
	1-9	10	11-21

				<u>% TOTAL</u>	
HR 1			25	52	86
			10	10	27
			<u>154</u>	<u>72</u>	<u>271</u>
% TOTAL			26.7	19.0	54.3
				98.1	
<u>PATTERN 1</u>				<u>PATTERN 2</u>	
<u>% TOTAL</u>				<u>% TOTAL</u>	
HR 2	22	21	58	24	21
	14	5	16	12	5
	<u>133</u>	<u>41</u>	<u>176</u>	<u>100</u>	<u>45</u>
% TOTAL:			67.5	% TOTAL	
34.8		13.8	51.4	28.0	
<u>% TOTAL</u>				<u>% TOTAL</u>	
HR 4	17	4	19	14	4
	7	0	3	7	0
	<u>67</u>	<u>16</u>	<u>92</u>	<u>57</u>	<u>14</u>
% TOTAL:			31.3	% TOTAL	
40.4		8.9	50.7	34.5	
<u>% TOTAL</u>				<u>% TOTAL</u>	
HR 6	16	2	7	15	2
	2	1	0	2	1
	<u>46</u>	<u>8</u>	<u>73</u>	<u>39</u>	<u>8</u>
% TOTAL:			21.9	% TOTAL	
40.8		7.0	51.0	35.7	
<u>% TOTAL</u>				<u>% TOTAL</u>	
HR 8	9	1	4	6	0
	3	1	0	3	1
	<u>34</u>	<u>10</u>	<u>51</u>	<u>29</u>	<u>9</u>
% TOTAL:			15.8	% TOTAL	
40.7		10.6	48.7	33.0	

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Line 10 to the grid base. The SE sector comprises the remaining area to the east of the S line and below Indian Point. These results are summarized below in terms of number of occurrences and percentage of total possible observations:

TABLE 7  
SUMMATION TRAJECTORY OCCURRENCES  
SOUTH OF INDIAN POINT

August, 1978

	Pattern 1						Pattern 2					
	Southwest		South		Southeast		Southwest		South		Southeast	
	Occur	%	Occur	%	Occur	%	Occur	%	Occur	%	Occur	%
Hour 1	232	31.8	84	11.5	25	3.4	232	31.8	84	11.5	25	3.4
Hour 2	192	26.4	51	7.0	83	11.4	148	20.3	50	6.9	108	14.8
Hour 4	128	17.6	27	3.7	133	18.3	111	15.3	29	4.0	146	20.1
Hour 6	100	13.8	10	1.4	124	17.1	86	11.9	17	2.3	130	18.0
Hour 8	67	9.3	7	1.0	90	12.5	53	7.3	15	2.1	96	13.3

JANUARY, 1979

Hour 1	154	21.4	72	10.0	271	37.6	154	21.4	72	10.0	271	37.6
Hour 2	133	18.5	41	5.7	176	24.4	100	13.9	45	6.3	207	28.8
Hour 4	67	9.3	16	2.2	92	12.8	57	7.9	14	1.9	106	14.8
Hour 6	46	6.4	8	1.1	73	10.2	39	5.4	8	1.1	82	11.5
Hour 8	34	4.8	10	1.4	51	7.1	29	4.1	9	1.3	63	8.8

The effect of the pattern change is not so much as to alter the total; rather, it is to shift the number of occurrences from the SW sector to the S and SE sectors. There are anomalies found that may be associated with recirculation.

After five miles of southward movement from Indian Point, the results seem to indicate the anomaly of surface wind impaction against the South Mountain and High Tor Ridges. This anomaly occurred since there were no local wind measurements available to induce deflections.

Historical studies have shown such deflections do exist. The present results cannot account for terrain unless the trajectory paths are deflected by observed surface winds. This requires a larger monitoring network, strategically placed, than was available. This need for further definition of local wind field is confirmed by the differences that appear in the results generated by Patterns 1 and 2.

At the present time, based on historical studies, Pattern 2 is probably the better representation of local trajectories for the available data.

Assuming a continuous 1 M/S wind speed (2.2 MPH), the number of occurrences in the south sector represent those parcels that have traveled with the effective speed (neglecting recirculation). Of the totals given, only four have traveled greater than ten miles for August (1); five for August (2); six for January (2); six of January (1); and three for January (2). These points would have to had passed through or over the South, High Tor and Hook Mountain Ridge lines.

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### 3.2.3 March 1980 - December 1980

#### 3.2.3.1 General

The results of the Trajectory II Study conducted by York Services Corporation for Con Edison have been recently submitted (Kaplin and Wuebber, 1981).

It was concluded from the initial trajectory study (Kaplin and Wuebber, 1980; Kaplin, 1980) that a lack of directional persistence of low speed surface winds (10M) at Indian Point and Bowline make recirculation of local air probable. There were indications of both convergence and divergence of local air streams. Objectivity created surface air parcel trajectories generated anomalies by passing over or through abrupt terrain features. The two local monitoring sites available were unable to resolve these anomalies.

In the Trajectory II Study, a supplemental network of ten surface wind monitoring stations were established for the express purpose of objectively assessing the southward movement of air parcels from Indian Point (see Figures 3 and 4). Sites were selected, specifically, in an attempt to resolve anomalous flow patterns with respect to terrain and tributary river drainage basins. A listing of sites used is shown in Table 8. A listing of valid data collected for the period is shown in Table 9.

#### 3.2.3.2 Wind Frequency Distributions

An historical evaluation of the representativeness of the data collected in 1980 was made for Indian Point and Bowline. Variations in wind frequency distributions were found to be associated with climatological variations on the synoptic-cyclonic scale.

These variations can be naturally expected between any given year or set of years. For example: Over a 20 year period (1960-1979) prior to 1980 Bridgeport, WBAS, for the month of July had an average wind directional frequency for a north wind of  $6.5 \pm 2.7$ -percent with an absolute maximum of 12.9-percent in 1974 and an absolute minimum of 3.0-percent in 1979. In 1980, the frequency set a new low of 2.8-percent. A further extreme example was found at La Guardia WBAS. Based on an eight year average (1972-1979) the northwest wind has a frequency of  $12.2 \pm 4.2$ -percent with minimum of 6.1-percent (1973) and a maximum of 16.5 (1974). In 1980, a new maximum of 18.2-percent was observed while in 1979, the frequency was 7.9-percent, which was the second lowest value in the period.

It was noted that the climatic variations of wind frequencies at Indian Point were generally minimal and less pronounced. This was attributed to topographic confinement. The wind frequency data for Indian Point and Bowline for the data collection period were adjudged to be representative (Figures 6.1-6.4, Kaplin and Wuebber, 1981). It was assumed that all concurrently collected wind velocities were representative of respective monitoring sites and relationships between sites could be evaluated.

It was found that wind frequency distribution patterns in themselves were deceptive representations of the continuity of air movement in the lower Hudson River valley unless there was an understanding of the patterns of wind velocity variations on a temporal basis.

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TABLE 8  
LOCATIONS OF STATIONS RELATIVE TO INDIAN POINT

<u>Station</u>	<u>Distance (miles) from Indian Point</u>	<u>Direction (degrees)</u>
Iona Island	2.50	334
Annsville	2.20	020
Watch Hill Road	3.15	132
Jurka	6.65	122
Croton Point	6.40	155
Ossining	8.80	151
Grassy Point	3.20	191
Bowline Point	4.15	190
South Nyack	13.60	174
Piermont	15.95	173
Kingsland	13.00	163
Eastview	11.90	145
Westchester County Airport	20.00	135

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TABLE 9  
YORK SERVICES CORPORATION  
ONE RESEARCH DRIVE, STAMFORD, CT  
CLIENT: CONSOLIDATED EDISON OF NEW YORK  
VALID DATA FOR TRAJECTORY WIND SITES  
PERIOD OF RECORD: 1980

SITE		PERCENT VALID DATA										
		MARCH	APRIL	MAY	JUNE	JULY	AUGUST	SEPT	OCT	NOV	DEC	
01-Piermont		79.23	78.75	63.71	66.67	100.00	100.00	71.25	37.10	0.00	0.00	
02-Ossining		98.59	100.00	100.00	100.00	100.00	99.80	99.93	96.77	100.00	100.00	
03-Iona Island		96.77	53.75	85.28	69.72	53.76	95.63	100.00	99.46	100.00	82.80	
04-Jurka/Grassy		78.76	55.97	0.00	0.00	0.00	0.00	0.00	81.85	100.00	84.95	
05-Kingsland		87.57	94.65	17.47	59.79	98.66	100.00	100.00	100.00	74.03	93.55	
06-Watch Hill		74.46	0.00	0.00	0.00	48.79	70.56	100.00	85.48	100.00	100.00	
07-South Nyack		79.17	100.00	83.87	100.00	100.00	100.00	100.00	100.00	100.00	100.00	
08-Annsville		100.00	100.00	90.86	100.00	77.28	100.00	71.81	98.92	100.00	100.00	
09-Eastview		100.00	100.00	100.00	100.00	100.00	100.00	100.00	100.00	100.00	100.00	
10-Croton Point		99.73	100.00	89.85	100.00	98.12	96.77	81.39	47.04	100.00	36.96	
11-West Cty Apt		99.87	100.00	100.00	99.86	100.00	100.00	100.00	100.00	100.00	100.00	
12-Indian Point		99.66	100.00	94.82	99.24	95.41	94.69	99.17	99.93	100.00	99.46	
13-Bowline Point		72.45	99.72	98.91	85.97	92.41	98.52	96.53	96.98	96.46	86.16	



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### 3.2.3.3 Diurnal Wind Distributions

The diurnal variation curves for Indian Point and Bowline for the data collection period (March through December 1980) were found to be historically representative. For selected winter and summer months, they demonstrated all the attributes of the two "season" characteristics (Figure 6.21-6.27, Kaplin and Wuebber, 1981).

With some variation at selected monitoring sites, it was found that the diurnal wind distributions were not only seasonally characteristic but characteristic of the monitoring site locations. They could, almost without exception, be uniquely categorized as Hudson River "west bank"; Hudson River "east bank"; or "inland" (Figures 6.28-6.33, Kaplin and Wuebber, 1981).

The characteristics of this uniqueness were examined by combining all appropriate sites to generate "average" east and west bank diurnal wind distributions (Figures 6-38-6.42, Kaplin and Wuebber, 1981). The average diurnal curves for March, June and December, 1980 are shown in Figures 13, 14, and 15. A computer check revealed that individual days at any given site could be found that had observed 24 hour diurnal wind variation patterns that matched their own monthly average distributions and/or the appropriate east or west bank average diurnal distribution based on the criteria 16 or more hours of fit  $\pm 45^\circ$  (not necessarily consecutive).

While there are unique common characteristics to the diurnal wind distribution patterns in the Indian Point environs, variations in local meso-scale factors dictate that the ultimate path of an air parcel whose movement is determined by surface (10M level) wind velocities is governed by time of departure as well as point of departure. Between wind velocity monitoring sites in the region, persistent wind direction and wind speeds are not supported. This is most obvious during the "summer" season or at any time that the area is under the influence of a weak synoptic-cyclonic pressure gradient pattern. Between individual monitoring sites there is apparent divergence and convergence of surface air.

### 3.2.3.4 Resultant and Concurrent Hourly Winds

A first approach at the evaluation of southward movement of air for prolonged periods of time was made for the data collection period March 1, 1980 through December 31, 1980, by examining the frequency distribution of the 24 hour resultant winds (Kaplin and Wuebber, 1981). These results are shown in Table 10 as a function of persistence category (the ratio of the resultant to arithmetic average wind speeds). At persistence levels greater than 0.9, a north wind was found in only four out of 273 possible valid cases (1.4%). The average wind speed 2.75 M/S. The high wind speeds associated with all northerly winds implies strong synoptic-cyclonic scale pressure gradient systems are the generating mechanism.

Simple liner relationships between high persistent, 24-hour resultant winds between Indian Point and Piermont (bearing  $173^\circ$  about 16-miles from Indian Point) showed that an average 24-hour resultant wind direction of  $012 \pm 22^\circ$  at Indian Point was related to an average 24-hour resultant wind at Piermont of  $359 \pm 24^\circ$ . At the same time, for corresponding cases, the average resultant wind speed at Indian Point was about  $2.5 \pm 0.9$  M/S and the concurrent average resultant wind speed at Piermont was  $5.6 \pm 1.2$  M/S. The angular offset implies terrain tracking and high average resultant wind speeds indicate the necessity for a strong pressure gradient field.

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Such replacements were also implied when concurrent hourly average wind data from the selected monitoring sites were correlated to Piermont. These results are shown in Table 11 for the available concurrent data collected during the period from March 1, 1980 through October 31, 1980. Out of 4,394 valid data hours, there were only 56 (1.3%) in which Indian Point and Piermont had concurrent winds from the north (350-011°). Almost half of these cases (26:0.59%) occurred in May, 1980. There was only one such hour out of 742 valid data hours in July, 1980. For July, in fact, for 3,355 concurrent data hours from five southern sites there was only one additional hour in which a site, South Nyack, had a north wind direction concurrent with Piermont.

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TABLE 10  
FREQUENCY DISTRIBUTION OF 24 HOUR RESULTANT WIND DIRECTIONS  
INDIAN POINT (10 METER LEVEL)

Resultant Wind Direction	No. Obs.	<u>Persistence</u> > 0.9 Aver. SPDS (MPH)		No. Obs.	<u>Persistence</u> > 0.8 < 0.9 Aver. SPDS (MPH)		No. Obs.
		<u>Result.</u>	<u>Mean</u>		<u>Result.</u>	<u>Mean</u>	
350-011	4	5.75	6.15	6	4.87	5.62	2
012-034	20	4.97	5.22	6	4.05	4.67	4
035-056	13	4.81	4.98	8	3.56	4.04	1
057-079	5	4.06	4.24	2	2.70	3.15	2
080-101	0			0			1
102-124	0			0			0
125-146	0			0			0
147-169	0			0			1
170-191	2	2.90	3.10	1	3.20	3.70	0
192-214	10	3.38	3.53	5	2.38	2.76	5
215-238	3	4.07	4.23	7	2.69	3.11	1
237-259	0			3	4.40	5.27	2
260-281	0			1	2.40	3.00	2
282-304	0			3	3.03	3.70	1
305-326	6	4.17	4.48	4	5.15	6.00	3
327-349	<u>13</u>	7.82	8.22	<u>8</u>	4.64	5.40	<u>7</u>
	76			54			31
Percent:	25.94			18.43			10.58

Number of Valid 24-hour Resultants ( >17 hours): 293  
Percent Valid Data: 97.39

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TABLE 11  
SUMMARY OF TWO-STATION WIND CORRELATIONS PIERMONT (SITE 1) REFERENCED TO SELECTED MONITORING LOCATIONS  
(SITE 2)

Month	Station (Site 2)	Total	Number of Observations			Concurrent Winds			North Wind at Piermont Resultant Wind @ Site 2			North Wind at Site 2 Resultant Wind @ Piermont		
			Piermont	Site 2	All Directions	North	Direct.	Speed (mph)	Persist.	Direct.	Speed (mph)	Direct.	Speed (mph)	Persist.
March	South Nyack	588	83	33	70	10	331	3.3	0.90	025	8.5	025	8.5	0.
	Kingsland	481	71	74	133	39	004	6.7	0.96	001	9.2	001	9.2	0.
	Ossining	586	83	25	31	4	327	3.8	0.92	023	7.5	023	7.5	0.
	Iona Island	565	76	16	65	3	327	5.0	0.73	042	9.4	042	9.4	0.
	Indian Point	589	83	67	137	7	035	4.2	0.89	334	9.3	334	9.3	0.
April	South Nyack	567	50	25	63	6	346	1.9	0.65	025	6.6	025	6.6	0.
	Kingsland	490	39	68	145	13	008	3.2	0.79	018	8.1	018	8.1	0.
	Ossining	567	50	28	45	2	316	2.6	0.85	030	8.8	030	8.8	0.
	Iona Island	354	30	21	26	2	308	4.5	0.73	043	11.0	043	11.0	0.
	Indian Point	567	50	58	129	6	020	2.4	0.64	340	8.5	340	8.5	0.
May	South Nyack	359	46	20	33	2	327	3.7	0.84	032	6.8	032	6.8	0.
	Kingsland	67	2	2	9	0	256*	2.1*	0.98*	031*	10.0*	031*	10.0*	1.
	Ossining	474	68	37	28	4	309	6.4	0.85	036	9.9	036	9.9	0.
	Iona Island	415	66	22	20	2	301	7.3	0.89	052	6.6	052	6.6	0.
	Indian Point	422	50	58	91	26	350	3.5	0.79	356	10.1	356	10.1	0.
June	South Nyack	480	17	19	57	2	332	4.0	0.92	036	8.2	036	8.2	0.
	Kingsland	284	5	2	30	1	301*	5.6*	0.95*	031*	5.4*	031*	5.4*	0.
	Ossining	480	17	16	41	0	318	7.3	0.90	046	6.7	046	6.7	0.
	Iona Island	368	12	8	16	0	301	8.5	0.88	062*	9.1*	062*	9.1*	0.
	Indian Point	475	17	28	80	4	326	3.1	0.73	006	5.6	006	5.6	0.
July	South Nyack	744	10	25	114	1	332	2.7	0.92	041	7.9	041	7.9	0.
	Kingsland	724	10	25	98	0	337	2.3	0.55	052	5.7	052	5.7	0.
	Ossining	744	10	37	66	0	326	4.5	0.79	047	6.8	047	6.8	0.
	Iona Island	401	5	9	45	0	302*	10.0*	0.95*	054*	5.7*	054*	5.7*	0.
	Indian Point	742	10	35	108	1	007	2.3	0.64	358	4.7	358	4.7	0.

\* Less than 10 valid data points

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TABLE 11 (Cont'd)

SUMMARY OF TWO-STATION WIND CORRELATIONS PIERMONT (SITE 1) REFERENCED TO SELECTED MONITORING LOCATIONS  
(SITE 2)

<u>Month</u>	<u>Station (Site 2)</u>	<u>Total</u>	<u>Number of Observations</u>			<u>Concurrent Winds</u>			<u>North Wind at Piermont Resultant Wind @ Site 2</u>			<u>North Wind at Site 2 Resultant Wind @ Piermont</u>		
			<u>Piermont</u>	<u>Site 2</u>	<u>All Directions</u>	<u>All Directions</u>	<u>North</u>	<u>North</u>	<u>Direct.</u>	<u>Speed (mph)</u>	<u>Persist.</u>	<u>Direct.</u>	<u>Speed (mph)</u>	<u>Persist.</u>
Aug.	South Nyack	744	24	44	95	95	1	334	3.0	0.75	0.75	058	11.9	0.95
	Kingsland	744	24	52	93	93	1	313	4.8	0.74	0.74	054	11.7	0.92
	Ossining	741	24	58	96	96	2	336	6.9	0.84	0.84	047	10.5	0.87
	Iona Island	679	24	15	80	80	0	308	4.9	0.66	0.66	036	9.8	0.80
	Indian Point	719	24	32	139	139	3	037	2.5	0.70	0.70	332	5.2	0.73
Sept.	South Nyack	513	37	36	63	63	10	348	2.5	0.88	0.88	018	12.6	0.91
	Kingsland	513	37	22	42	42	3	345	4.6	0.79	0.79	026	9.6	0.90
	Ossining	512	36	36	87	87	9	354	5.0	0.86	0.86	010	9.4	0.84
	Iona Island	513	37	10	34	34	3	322	4.8	0.76	0.76	024	11.7	0.91
	Indian Point	501	37	38	102	102	4	031	3.4	0.83	0.83	339	7.2	0.85
Oct.	South Nyack	379	24	28	83	83	5	335	2.8	0.91	0.91	019	10.7	0.92
	Kingsland	379	24	25	64	64	5	018	3.0	0.73	0.73	019	11.4	0.96
	Ossining	379	24	31	69	69	4	007	4.0	0.85	0.85	016	11.8	0.95
	Iona Island	379	24	12	38	38	3	291	2.8	0.77	0.77	016	12.9	0.99
	Indian Point	379	24	24	89	89	5	031	2.9	0.95	0.95	009	7.5	0.86

Table 11 can be summarized in terms of the valid wind direction data from the five designated sites concurrent with wind directions at Piermont (Note: The data from these sites should not be presumed concurrent with each other simultaneously). This summary is shown in Table 12.

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TABLE 12  
CONCURRENCE OF TWO-STATION WIND DIRECTIONS  
(Relative to Piermont)

	<u>5 Site Total Concurrent Data</u>	<u>% Valid 6 Site Basis</u>	<u>Percent (Concurrent Data Basis)</u>	
			<u>All Directions</u>	<u>North Wind</u>
March	2809	62.93	15.52	2.24
April	2545	58.91	16.03	1.14
May	1737	38.91	10.42	1.96
June	2087	48.31	10.73	0.34
July	3355	75.16	12.85	0.06
August	3627	81.25	13.87	0.19
September	2552	59.07	12.85	1.14
October	1895	42.45	18.10	1.16

Since these data are derived from hourly average wind directions, it is again shown that there is little likelihood of sustaining south bound movement of air from Indian Point beyond 15-miles.

### 3.2.3.5 Summary-Trajectory II Study

A modified version of the generic model TRAJECTORY (Kaplin and Wuebber, 1980) was used as a basis of a study, which involved the use of concurrent hourly average wind data from a network of 13-14 monitoring stations within 20-miles of Indian Point. All but four sites were located on the Hudson River shorelines. Only two sites were used to the north of Indian Point. As with the original study (Kaplin and Wuebber, 1980), emphasis was on the objective creation of the trajectories of air parcels originating hourly at Indian Point with a speed and direction equal to the average wind at the 10 meter level of Indian Point 122 Meter Meteorological Tower.

In the Trajectory II Study (Kaplin and Wuebber, 1981), each parcel of air was tracked for eight consecutive hours after its movement was initiated as dictated by two factors: The movement time interval and the wind velocity at the coordinate end point of the parcel at the end of movement time interval. As in the earlier study a 21 by 21-mile grid pattern of one mile squares was used to generate tabulations of the trajectory segment end-points. While the tabulations assumed each point to be located in the center of each square, the actual coordinates within the squares were used as a starting point for the next hourly trajectory segment. Subsequent movement of a parcel from a given set of coordinates was determined by the appropriate average hourly wind velocity assigned to grid square as determined by a "zone of influence" file. While trajectories were objectively created, the zone of influence file required the subjective assignment of each available wind velocity monitoring site to specific grid squares. The wind velocity for any given hour within these assigned grid squares would be the same as that of the specified control monitoring site.

Influence assignments were based on assessments of local wind patterns (historical and present) with some consideration for obvious meso-scale modification factors: topographic channeling, drainage flow patterns, thermally induced flow patterns, etc.

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For the purpose of the study, all movements of parcels past the grid boundaries were assumed to continue their movements under the influence of the site whose wind was being used at the time that a boundary was crossed.

When only a few wind monitoring sites are available to cover a large area, movement controlled solely by a single site's non-variant hourly average wind was not a critical factor. As the number of monitoring sites increased and zones of influence became smaller, discrete movement based on a single wind in a given hour increment would allow a parcel to move through a zone of influence without modification of its controlling wind, which could be substantially different in direction and speed than that associated with the by-passed zone. In the TRAJ3 model, as used in this study, parcel end point coordinates at the end of each hour, which had been previously determined by the non-variant wind at the parcel source at the beginning of each hour, were the resultant of 30 discrete movements (two minute intervals) within the hour interval. On this basis, a wind speed of 30 mph was required in order for a parcel to travel with a non-variant wind for more than one mile. A discrete wind velocity was reassigned to a parcel according to its coordinates (zone of influence) at the end of each two minute intervals. In effect, a parcel could, in extreme, alter its direction and speed 30 times in a given hour and not apparently move at all if it were trying to move from one zone of influence to another at a boundary line between zones and the wind in the two zones were in opposition. Such apparent anomalies were found as a matter of routine.

Figure 16 shows the grid system that was used in the Trajectory 2 Study, Indian Point was located at coordinates 7,17. In the development of the trajectories recirculation was allowed. This is a parcel could leave the grid boundaries and be brought back onto the grid at a later time if dictated by a change in the wind at the monitoring site controlling its movement.

In the first hour of movement for the ten months of data, only 390 parcels out of 7,344 (5.3) left the grid system and did not return. Of these 345 (4.7%) were crossings of the northern grid boundary, 40 (0.5%) were crossings of the western grid boundary and 5 (0.1%) were crossings of the southern grid boundary. There were no crossings of the eastern grid boundary. Of the five points crossing the southern boundary, all occurred in December, 1980. To cross the southern boundary a minimum speed of 7.6 M/S (17 mph) was required.

During the ten months (206 days) of record there were only 35 days (11.4%) in which there are no rotation and/or recirculation (flow reversal) characteristics in at least one of the 24 trajectories created daily. This does not take into account any trajectories that may have experienced reversals wholly outside of the grid area.

If flow reversals and rotational trajectories were due solely to synoptic scale meteorological patterns, then randomness would be expected in the starting time occurrence frequency as a diurnal function. The actual pattern observed for those eight hour trajectories, which contained at least one on grid or on-off-on grid flow reversal is shown in Table 13. These results include those reversals of synoptic origin and can be interpreted as consistent with the local average diurnal wind patterns induced by meso-scale phenomena when the following factors are taken into account:

- Nocturnal flow patterns generally north to south with minimal with speeds
- Afternoon flow patterns generally south to north with maximum wind speeds

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Summaries of the trajectory end-point counts and percentages are shown on a monthly basis in Tables 14 and 15 for selected time increments up to the complete eight hour trajectory. The idealized valley is shown in Figure 16. It is noted that the idealized valley contains 64 out of 441 (14.5%) possible grid box end-point coordinates: ten grid boxes are north of Indian Point; and 54 are on a line with Indian Point and south. With respect to total grid, 84 boxes are to the north and 357 are to the south.

The effect of dominant meso-scale factors are readily discernible in the results. Differences in north and south boundary crossings of trajectory points can be related to the normal seasonal distribution of local wind velocities as well as their diurnal distribution patterns. During the summer season it would take up to six hours of persistent north sector winds from Indian Point to Piermont to generate a south boundary crossing. Such a persistent diurnal time span is improbable. After eight hours, only 6.7 percent of all possible trajectory end-points for all time intervals in July 1980 (5,924 possible) were found to cross the southern boundary (a distance of 17-miles from Indian Point). The highest crossing percentage, 30.7 percent of 5,924 possible, occurred in December, 1980.



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TABLE 13  
Diurnal Distribution of Occurrences of  
Eight-Hour Trajectories with On Grid Reversals  
Number of Trajectory with Flow Reversals

Starting Hour	End Hour	Mar	Apr	May	June	July	Aug	Sept	Oct	Nov	Dec	Total
0100	0900	7	9	13	13	14	9	13	7	7	10	102
0200	1000	11	9	15	9	11	10	15	10	9	6	105
0300	1100	9	11	12	15	14	14	12	12	8	8	115
0400	1200	12	11	18	14	17	13	11	12	6	10	124
0500	1300	10	11	16	17	21	13	14	11	4	9	126
0600	1400	9	11	10	13	19	13	14	11	7	11	118
0700	1500	10	8	9	10	10	12	12	7	9	9	96
0800	1600	6	7	5	10	9	10	14	7	10	7	85
0900	1700	6	10	6	8	11	6	10	4	7	8	76
1000	1800	5	7	10	11	7	8	9	4	9	7	77
1100	1900	6	8	4	4	9	7	9	6	4	5	62
1200	2000	2	8	2	4	10	8	10	5	6	4	59
1300	2100	2	8	4	5	6	5	8	4	5	6	53
1400	2200	3	4	5	5	11	5	7	6	5	6	57
1500	2300	2	7	6	4	12	11	8	6	8	6	70
1600	2400	5	5	7	6	12	11	8	6	10	7	77
1700	0100	7	6	10	9	17	12	9	6	8	8	92
1800	0200	10	7	12	8	16	10	9	9	7	9	97
1900	0300	10	3	14	11	14	8	7	6	7	10	90
2000	0400	12	9	14	7	15	8	6	6	6	8	91
2100	0500	12	8	13	5	8	8	7	7	9	8	85
2200	0600	15	9	12	10	8	6	7	8	6	8	89
2300	0700	10	12	13	8	10	9	8	9	5	9	93
2400	0800	7	5	12	10	13	11	9	8	5	11	91
TOTAL		188	193	242	216	294	227	236	177	167	190	2130
TOTAL POSSIBLE		736	712	736	712	736	736	712	736	712	736	7264
OCCURRENCE (%)		25.5	27.1	32.9	30.3	39.9	30.8	33.1	24.0	23.5	25.8	29.3



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TABLE 14  
SUMMARY OF TRAJECTORY END-POINT COUNTS

		<u>March</u>	<u>April</u>	<u>May</u>	<u>June</u>	<u>July</u>	<u>August</u>	<u>Sept</u>	<u>Oct</u>	<u>Nov</u>	<u>Dec</u>
Number of Hours/Month											
Total Number of End-Points		744 5924	720 5732	744 5924	720 5732	744 5924	744 5924	720 5732	744 5924	720 5732	744 5924
<u>Elapsed Time</u>		<u>Number of End-Points</u>									
2 Hours -	Number of Trajectories	742	718	742	718	742	742	718	742	718	742
	Within Grid	490	440	577	559	593	599	498	595	538	513
	In Valley (North of Indian Point)	41	29	49	65	51	57	36	39	30	44
	In Valley (Indian Point and South)	95	94	150	148	305	185	178	204	139	123
Percent Grid Points in Valley		28.3	28.0	34.5	38.1	60.0	40.4	43.0	40.8	31.4	32.6
4 Hours -	Number of Trajectories	740	716	740	716	740	740	716	740	716	740
	Within Grid	275	244	356	355	433	361	328	371	285	297
	In Valley (North of Indian Point)	20	8	17	34	28	26	15	22	13	17
	In Valley (Indian Point and South)	43	59	118	106	139	146	109	156	97	78
Percent Grid Points in Valley		22.9	27.5	37.9	39.4	38.6	47.6	37.8	48.0	38.6	32.0
6 Hours -	Number of Trajectories	738	714	738	714	738	738	714	738	714	738
	Within Grid	173	183	257	257	342	277	250	255	175	213
	In Valley (North of Indian Point)	8	5	9	13	20	9	6	16	7	11
	In Valley (Indian Point and South)	18	42	93	91	105	119	80	106	63	46
Percent Grid Points in Valley		15.0	25.7	39.7	40.5	36.5	46.2	34.4	47.8	40.0	26.8
8 Hours -	Number of Trajectories	736	712	736	712	736	736	712	736	712	736
	Within Grid	121	135	190	194	276	226	199	188	130	160
	In Valley (North of Indian Point)	9	5	6	9	9	3	5	4	6	11
	In Valley (Indian Pint and South)	11	25	73	61	91	87	51	70	47	38
Percent Grid Points in Valley		16.5	22.2	41.6	36.1	36.2	39.8	28.1	39.4	40.8	30.6
Total Past South Boundary*		1643	958	1048	636	399	708	657	926	1706	1821
Percent Past South Boundary		27.7	16.7	17.7	11.1	6.7	12.0	15.0	15.6	29.8	30.7
Total Past North Boundary*		1013				1676					937
Percent Past North Boundary		17.1				28.3					15.8

\* This count includes all end-points for all time intervals.

# IP2 DEFUELED SAFETY ANALYSIS REPORT

TABLE 15  
SUMMARY OF TRAJECTORY END-POINTS (Percent)

<u>Month</u>	2 Hours			4 Hours			6 Hours			8 Hours		
	<u>Total</u> <u>Irai.</u>	<u>% On</u> <u>Grid</u>	<u>% In</u> <u>Valley</u>	<u>Total</u> <u>Irai.</u>	<u>% On</u> <u>Grid</u>	<u>% In</u> <u>Valley</u>	<u>Total</u> <u>Irai.</u>	<u>% On</u> <u>Grid</u>	<u>% In</u> <u>Valley</u>	<u>Total</u> <u>Irai.</u>	<u>% On</u> <u>Grid</u>	<u>% In</u> <u>Valley</u>
March	742	66.0	18.3	740	37.2	8.5	738	23.4	3.5	736	16.4	2.7
April	718	61.3	17.1	716	34.1	9.4	714	25.6	6.6	712	19.0	4.2
May	742	77.8	26.8	740	48.1	18.2	738	34.8	13.8	736	25.8	10.7
June	718	77.7	29.7	716	49.6	19.5	714	36.0	14.6	712	27.2	9.8
July	742	79.9	48.0	740	58.5	22.5	738	46.3	16.9	736	37.5	13.6
August	742	80.7	32.6	740	48.8	23.2	738	37.5	17.3	736	30.7	12.2
September	718	69.4	29.8	716	45.8	17.3	714	35.0	12.0	712	27.9	7.9
October	742	80.2	32.7	740	50.1	24.1	738	34.6	16.5	736	25.5	10.1
November	718	74.9	23.5	716	39.8	15.4	714	24.5	9.8	712	18.3	7.4
December	742	69.1	22.5	740	40.1	12.8	738	28.9	7.7	736	21.7	6.7
Average		73.7	28.1		45.2	17.1		32.7	11.9		25.0	8.5
Standard Deviation		6.5	8.4		7.0	5.3		6.7	4.6		8.1	3.3

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In any given month of the ten that were investigated and out of the 7,242 complete eight hour trajectories that were generated, there were a number of basic pattern types. There were those whose sequence of temporal end-points exhibited basic straight line tendencies. These were generally associated with high wind speeds. There were those whose sequence of end-points that rotated in a more or less smooth pattern but they were not usually associated with a meso-scale diurnal rotation or terrain induced deflections. In addition, there were those that exhibited characteristics of recirculation and those in which there were sharp reversals. These were induced by wind velocity changes of synoptic scale origin and/or most frequently they appeared in those trajectories, which included the morning or evening transitional periods, meso-scale induced. In the latter cases, of rotation, recirculation and reversals, the sequences may occur wholly on the grid or on-off-on on the grid.

Examination of details of specific trajectories and concurrent opposing winds suggests that three block regions can be projected in the lower Hudson Valley within the grid system. These regions are:

- Peekskill Bay
- Haverstraw Bay
- Tappan Zee

The reality of the zones of divergence and convergence on a concurrent wind basis were premised on the continuity of air flow movements locally and for air streams that were projected to cross the Hudson River. This latter feature was not uniquely demonstrated with respect to surface (10 meter) level wind velocities. If some local wind patterns were induced by thermal differentials between land and water during periods of weak geostrophic pressure gradients in accordance with sea breeze concepts: during the day air will move from cool water to warm land with return flow aloft; at night, a reverse flow pattern may develop. If this occurred on opposite shorelines of a wide river, concurrently, then there should be vertical motions induced by convergence and divergence in the mid river area or a region of air flow directionally independent of the shoreline circulations. The light wind speeds that were normally found at Croton Point during periods of weak geostrophic flow may result from its proximity to a mid river transition zone. (During periods of strong northwesterly gradients Croton Point had a high frequency of west to west northwest winds implying a cross river flow parallel to South Mountains.)

In addition to the land-water effects in so far as they generate local on-shore and off-shore winds, the effect of nocturnal drainage winds should be considered. The Kingsland Park site was one example. A zone of convergence frequently develops at night between that site and Ossining. This was also apparent from the average diurnal monthly diurnal wind distributions. If only drainage winds are considered, from Figure 4, Ossining would reflect drainage from the Croton River and/or a secondary local river both of which would generate air movements from the north-northeast. Kingsland Park is, however, at the outfall to Gory Brook, while this drains from the north-northeast, it hooks in its final section and outfalls into the Hudson from the southeast. There were no measurements available upon which to alter these directions after the air streams flow into the Hudson River itself. For the objective creation of eight hour trajectories, these winds were presumed to extend into the Hudson River and generate local blocks on a concurrent hourly basis.

The dominance of mesoscale flow factors on surface winds in this study have been demonstrated over and over since local meteorological data have been collected. The surface wind data sets used for this study are from the most extensive network of concurrent monitoring stations that have ever been deliberately located in the region.

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These data have been evaluated by many of the routine methodologies common to earlier local studies. All of the data sets were found to exhibit characteristics of complex meso-scale flow fields distortions. The east shore stations were found to share some common characteristics on a daily basis and the same was true for the west shore stations. These characteristics were frequently in opposition to each other. At the same time inland stations had characteristics that were entirely different from either the east or the west shore stations.

The creation and interpretation of eight hour trajectories from these data sets could not be truly separated from the concurrent flow fields on an hourly basis. The eight hour trajectories were a result of the constantly changing concurrent flow field. They were a distinct function of movement interval when based on hourly average wind velocities; and therefore, it may be presumed that in a dynamic flow field they would be equally sensitive to the wind averaging interval itself. As noted earlier, this study did not account for vertical air movement, the trajectories were therefore extremely sensitive to one crucial factor - the assumption of continuity of air movement across the Hudson River without midstream directional distortions.

The results of this study indicated that continued southward movement of air parcels in the Hudson River Valley could not generally be sustained past Piermont, if, in accordance with the data evaluated, Piermont's winds are assumed to be representative of the full width of Hudson River.

### 3.2.4 January 1, 1979 Through December 31, 1980

#### 3.2.4.1 Data Analyses

In the previous sections, with reference to the Trajectory I and Trajectory II studies, portions of 1979 and 1980 data from the Indian Point Meteorological Tower were analyzed and evaluated with reference to the studies that were in progress. Some of these analyses included references to historical data. In both of these studies it was concluded that the meteorological data being obtained at Indian Point were representative of that site and that any observed variations in wind frequency distributions and diurnal variations were assignable to transient climatological deviations from the norm on, at least, the synoptic-cyclonic scale of meteorological events. There was no indication that any changes could be attributed to local physical or dynamic modification, and/or in monitoring equipment and analyses techniques, which could introduce permanent data bias.

To maximize the recent data analyses, all of 1979 and 1980 have been evaluated and compared to historical data as available. The amount of valid data for these years has been previously tabulated in Table 2.

#### 3.2.4.2 Wind Frequency Distributions

For Climatological perspective, historical comparisons have been made for selected months: March, July and December, between the wind direction, wind frequency distributions at the 10 meter level of the 122 Meter Indian Point Tower and climatological data available from Bridgeport, Connecticut and La Guardia Airport in New York City. These latter sites are within the synoptic-cyclonic scale range of Indian Point. The distributions are shown in Table 16. With respect to March and July, climatological frequency extremes are noted on a nine year basis for La Guardia and a 21 year basis for Bridgeport.

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The fluctuations of the frequency distributions and the extremes can be associated with the frequency distribution fluctuations at Indian Point when consideration is given to the fact that winds at Indian Point are channeled by the west bank terrain.

Tables 17 and 18 give the percent frequency distribution of wind direction at Indian Point at the on "seasonal" basis for the 10 meter and 122 meter sensor levels. These data are compared to comparable results for 1973-1974 (Kaplin, et. al.1974, Appendix D).

In the summer season (Table 17), there is a frequency shift at the 10 meter level from SW and SSW in 1974 to SSW and S in the 1979-1980 period. This shift (with a directional bias) can be related to a similar shifting pattern at Bridgeport and La Guardia as found for July in Table 15. This shift tendency is also found implied at the 122 meter level. There is no reason to expect that these pattern changes are permanent.

For the winter season (Table 18) there is a recent bias of wind frequencies to the NNW and N sectors at both the 10 meter and 122 meter levels at Indian Point. These shifts have their counterpart in the March and December distributions of Table 16.

It is concluded that the wind velocity data that has been collected in recent years is consistent with the data base for FSAR 2 at all measurement levels when normal climatological variations are considered.

# IP2 DEFUELED SAFETY ANALYSIS REPORT

TABLE 16A  
HISTORICAL COMPARISONS OF  
WIND FREQUENCY DISTRIBUTIONS  
MARCH

	Indian Point			LaGuardia			Bridgeport		
	1974	1979	1980	1974	1979	1980	1974	1979	1980
N	.120	.078	.125	.044	.137*	.085	.044	.083	.077
NNE	.108	.093	.104	.032	.038	.016**	.012	.020	.040
NE	.115	.174	.173	.109	.105	.085	.097	.031	.052
ENE	.019	.039	.057	.085	.032	.061	.069	.043	.097
E	.008	.011	.030	.000**	.013	.024	.020##	.095	.061
ESE	.004	.009	.007	.008**	.013	.008**	.008	.018	.031
SE	.011	.004	.011	.008**	.023	.032	.016	.035#	.004##
SSE	.027	.014	.016	.040	.044*	.032	.012	.031#	.016
S	.057	.094	.085	.085	.199*	.145	.036	.082#	.069
SSW	.051	.133	.091	.036	.035	.057	.040	.051	.040
SW	.045	.065	.046	.052	.018	.024	.085	.057	.069
WSW	.020	.042	.024	.028	.024	.028	.057	.035	.032
W	.036	.049	.039	.089	.069	.048	.113	.065	.113
WNW	.076	.047	.024	.137	.097	.077	.145	.077	.101
NW	.124	.065	.034	.165	.079	.182*	.133	.097	.137
NNW	.129	.085	.134	.081	.066	.081	.093	.134	.048
CALM	.000	.000	.000	.000	.008	.016	.020	.047#	.012

\* 9 Year High (1972-1980)

\*\* 9 Year Low (1972-1980)

## 21 Year Low (1960-1980)

# 21 Year High (1960-1980)



# IP2 DEFUELED SAFETY ANALYSIS REPORT

TABLE 16B  
HISTORICAL COMPARISONS OF  
WIND FREQUENCY DISTRIBUTIONS  
JULY

	Indian Point			LaGuardia			Bridgeport		
	<u>1974</u>	<u>1979</u>	<u>1980</u>	<u>1974</u>	<u>1979</u>	<u>1980</u>	<u>1974</u>	<u>1979</u>	<u>1980</u>
N	.034	.066	.047	.093*	.036	.044	.129#	.030	.028##
NNE	.141	.112	.111	.032	.035	.040	.057	.012##	.016
NE	.148	.104	.158	.057	.059	.052	.028	.015	.012##
ENE	.054	.085	.073	.048	.055	.040##	.020	.028	.032
E	.026	.044	.045	.020	.013	.016	.069	.082	.057
ESE	.011	.019	.022	.012	.013	.008	.016	.054	.024
SE	.020	.013	.022	.044*	.031	.032	.048	.020	.016
SSE	.024	.028	.026	.016**	.032	.040	.020	.022	.020
S	.053	.079	.085	.081**	.184	.161	.073	.102	.129
SSW	.110	.212	.123	.069	.085	.073	.052##	.079	.113
SW	.129	.097	.094	.061**	.095	.133*	.085	.157	.125
WSW	.044	.028	.050	.081	.063	.040**	.154	.114	.109
W	.047	.038	.061	.149*	.073	.073	.145#	.105	.117
WNW	.043	.024	.031	.089	.063	.081	.048	.057	.052
NW	.034	.026	.024	.073	.067	.093	.004##	.069	.069
NNW	.015	.015	.031	.057	.043	.052	.048	.026	.069
CALM	.000	.000	.000	.020	.051	.020	.004	.030	.012

\* 9 Year High (1972-1980)

\*\* 9 Year Low (1972-1980)

# 21 Year High (1960-1980)

## 21 Year Low (1960-1980)

# IP2 DEFUELED SAFETY ANALYSIS REPORT

TABLE 16C  
HISTORICAL COMPARISONS OF  
WIND FREQUENCY DISTRIBUTIONS  
DECEMBER

	Indian Point			LaGuardia			Bridgeport		
	<u>1973</u>	<u>1979</u>	<u>1980</u>	<u>1973</u>	<u>1979</u>	<u>1980</u>	<u>1973</u>	<u>1979</u>	<u>1980</u>
N	.068	.101	.082	.081	.044	.093	.101	.058	.048
NNE	.172	.071	.148	.077	.016	.044	.048	.019	.057
NE	.162	.067	.124	.081	.046	.065	.097	.030	.040
ENE	.041	.023	.061	.040	.046	.044	.044	.031	.020
E	.015	.018	.023	.020	.013	.008	.024	.035	.012
ESE	.004	.004	.011	.016	.009	.004	.016	.013	.000
SE	.004	.008	.016	.024	.011	.000	.024	.015	.004
SSE	.015	.030	.023	.040	.008	.004	.024	.012	.024
S	.033	.100	.070	.093	.047	.056	.044	.023	.016
SSW	.048	.091	.081	.040	.058	.056	.048	.020	.024
SW	.049	.042	.057	.061	.122	.081	.048	.040	.069
WSW	.019	.042	.024	.044	.071	.044	.028	.093	.085
W	.026	.108	.031	.089	.206	.081	.141	.233	.157
WNW	.044	.094	.057	.149	.144	.121	.141	.238	.157
NW	.126	.120	.070	.069	.078	.157	.101	.066	.153
NNW	.074	.062	.114	.069	.059	.133	.044	.036	.097
CALM	.016	.000	.000	.008	.022	.008	.024	.038	.036

# IP2 DEFUELED SAFETY ANALYSIS REPORT

TABLE 17  
COMPARISON OF PERCENT WIND FREQUENCY DISTRIBUTIONS – SUMMER

Wind Direction	10 Meter Level				122 Meter Level			
	<u>1974</u>	<u>1979</u>	<u>1980</u>	<u>1979-80</u>	<u>1974</u>	<u>1979</u>	<u>1980</u>	<u>1979-80</u>
N	3.66	5.53	7.15	6.34	6.54	7.25	8.19	7.72
NNE	10.71	10.27	10.07	10.17	10.67	11.22	10.27	10.74
NE	15.89	12.58	13.68	13.13	6.40	4.85	5.31	5.08
ENE	6.20	7.46	7.78	7.62	2.41	1.88	2.26	2.07
E	2.85	2.68	3.27	2.97	1.83	1.88	1.60	1.74
ESE	1.59	1.59	1.97	1.78	1.66	1.00	1.28	1.14
SE	2.20	1.31	1.77	1.54	2.91	1.61	1.83	1.72
SSE	2.85	2.52	2.09	2.30	2.88	2.40	2.86	2.63
S	8.47	13.28	9.05	11.17	13.04	18.49	14.20	16.36
SSW	11.86	17.39	12.75	15.07	9.82	12.74	9.95	11.35
SW	12.23	9.14	8.17	8.65	10.64	10.22	8.32	9.28
WSW	3.63	3.26	4.08	3.67	5.56	5.14	5.21	5.18
W	3.08	4.28	4.95	4.62	4.44	4.12	5.24	4.68
WNW	2.61	2.74	3.88	3.31	4.17	3.85	4.37	4.11
NW	3.19	2.43	3.65	3.04	5.05	6.84	7.71	7.27
NNW	2.41	2.77	3.59	3.18	3.79	5.71	8.28	6.99
VAR.	6.51	0.00	0.00	0.00	1.56	0.00	0.00	0.00
CALM	0.07	0.00	0.00	0.00	0.14	0.00	0.00	0.00
MISS.	0.03	0.77	2.09	1.43	6.50	0.79	3.13	1.96
NO VALID HOURS	2951	4411	4407	8818	2952	4413	4373	8786
% HRS IN DISTR	100.	99.9	99.8	99.8	100.	99.9	99.0	99.5

# IP2 DEFUELED SAFETY ANALYSIS REPORT

TABLE 18  
COMPARISON OF PERCENT WIND FREQUENCY DISTRIBUTIONS - WINTER

Wind Direction	10 Meter Level				122 Meter Level			
	<u>1974</u>	<u>1979</u>	<u>1980</u>	<u>1979-80</u>	<u>1974</u>	<u>1979</u>	<u>1980</u>	<u>1979-80</u>
N	7.19	10.73	11.90	11.32	7.23	11.15	10.97	11.06
NNE	13.67	10.34	10.05	10.20	13.67	13.22	14.33	13.78
NE	12.22	11.45	15.11	13.29	5.63	4.79	4.90	4.85
ENE	4.75	5.90	6.30	6.10	1.93	1.80	1.79	1.79
E	1.59	2.86	3.02	2.94	1.60	1.45	1.79	1.62
ESE	0.95	0.71	0.85	0.78	1.16	0.92	1.65	1.29
SE	0.99	0.67	1.26	0.96	1.36	1.75	1.44	1.60
SSE	1.81	1.98	1.49	1.73	1.79	1.82	1.76	1.79
S	6.44	6.50	5.63	6.06	8.67	10.57	8.26	9.41
SSW	7.39	8.11	7.65	7.88	8.25	8.64	8.04	8.34
SW	5.76	6.24	5.72	5.98	9.36	5.87	5.59	5.73
WSW	2.58	4.31	3.04	3.67	3.51	3.18	3.37	3.27
W	3.32	4.77	4.01	4.39	2.94	3.78	4.14	3.96
WNW	5.42	5.90	4.62	5.26	5.08	6.24	5.24	5.74
NW	11.88	9.63	7.10	8.36	14.01	14.56	13.46	14.01
NNW	8.80	8.80	12.07	10.44	10.42	10.27	13.28	11.78
VAR.	4.23	0.00	0.00	0.00	1.18	0.00	0.00	0.00
CALM	0.66	0.00	0.00	0.00	0.04	0.00	0.00	0.00
MISS.	2.80	1.11	0.18	0.64	5.50	0.00	0.00	0.00
NO VALID HOURS	4967	4341	4368	8709	4924	4342	4368	8710
% HRS IN DISTR	97.6	99.9	100.	100.	96.8	100.0	100.	100.

## IP2 DEFUELED SAFETY ANALYSIS REPORT

### 3.2.4.3 Diurnal Wind Direction Distributions

Seasonal diurnal distributions of the resultant wind directions for the combined 1979-1980 data period are shown in comparison to the 1973-1974 data collection period in Table 19 and in Figures 17 and 18.

The diurnal patterns with the exception of the summer season at the 122 meter level from the 2300 to 0900 are nearly identical for the 1979-80 data set and the historical 1973-74 data set. The deviation of the 122 meter level during the nocturnal hours is also consistent when considered with respect to the summer wind frequency shift at the 122 meter level to a sharply defined south wind maximum.

It is concluded on the basis of the diurnal wind distributions that the patterns at all levels are consistent with the data base for FSAR 2 at all measurement levels with consideration for normal climatological variations.

### 3.2.4.4 Wind Speed Distributions

All variable valid wind speeds at the 10 meter and 122 meter levels have been evaluated on a seasonal basis to determine their diurnal characteristics and the cumulative probability distributions. The results of these analyses are shown in Tables 20 and 21 for the summer season and Tables 22 and 23 for the winter season. For visual comparison, the diurnal variability is shown in Figure 19. The probability distributions are shown in Figure 20. In this latter Figure, the annual cumulative probabilities have been included. These curves were generated by combining the cumulative points in Tables 20-23.

The maximum average diurnal wind speeds from this data set can be compared with those of the 1973-1974 season as shown below in Table 24. There are no significant differences.

## IP2 DEFUELED SAFETY ANALYSIS REPORT

TABLE 19  
COMPARISON OF DIURNAL RESULTANT WIND DIRECTIONS

Time	10 Meter Level				122 Meter Level			
	Summer		Winter		Summer		Winter	
	1974	1979-80	1973-74	1979-80	1974	1979-80	1973-74	1979-80
0100	051	034	357	359	351	300	327	329
0200	054	036	358	002	008	313	328	333
0300	050	036	359	003	007	321	329	334
0400	042	043	360	004	355	325	334	335
0500	048	048	001	003	004	327	335	335
0600	050	041	000	007	007	331	336	337
0700	042	040	001	007	011	338	339	338
0800	017	023	357	004	012	341	340	338
0900	337	348	352	356	350	334	337	339
1000	293	302	343	351	317	312	336	336
1100	258	269	335	344	260	279	325	331
1200	246	259	329	336	241	272	320	325
1300	246	247	327	336	248	261	319	321
1400	246	249	323	333	246	260	316	316
1500	233	246	322	332	233	251	313	316
1600	228	244	316	336	221	245	306	315
1700	230	233	321	337	223	239	307	312
1800	226	237	327	340	222	239	303	313
1900	232	245	330	345	233	239	308	317
2000	290	346	340	348	248	250	313	320
2100	049	033	345	355	259	263	315	325
2200	049	018	347	358	277	276	319	329
2300	055	033	351	358	297	282	321	331
2400	045	037	354	360	317	287	321	330

# IP2 DEFUELED SAFETY ANALYSIS REPORT

Job Number: 01-4122-00

TABLE 20  
YORK RESEARCH CORPORATION  
ONE RESEARCH DRIVE, STAMFORD, CONNECTICUT 06906  
CLIENT: CON EDISON CO. OF NY  
SITE: INDIAN POINT (10M)  
PARAMETER: WIND SPEED (SUMMER SEASON)  
UNITS: MPH

DIURNAL ANALYSIS  
MAY 1, 79,80 - OCT. 31, 79,80

OBSERV #	PARAM AVG	PARAM STD DEV	MAX VALUE	RANGE	VALID PTS
1	2.775	1.643	10.500	10.000	362
2	2.748	1.634	9.000	8.500	362
3	2.780	1.643	10.000	9.500	362
4	2.805	1.713	10.000	9.400	362
5	2.748	1.615	9.500	8.900	359
6	2.813	1.790	10.000	9.500	358
7	2.936	1.867	13.000	12.500	358
8	3.103	1.806	12.000	11.400	363
9	3.323	1.899	10.500	9.900	363
10	3.481	1.663	10.000	9.400	361
11	3.809	1.712	10.500	9.900	362
12	3.969	1.618	10.000	9.400	363
13	4.133	1.801	14.000	13.400	363
14	4.104	1.825	12.000	11.000	362
15	4.019	1.833	12.500	11.500	361
16	3.825	1.804	13.000	12.400	359
17	3.562	1.759	12.000	11.400	360
18	3.076	1.641	9.000	8.500	361
19	2.669	1.609	12.000	11.500	362
20	2.593	1.634	10.000	9.500	361
21	2.706	1.763	12.000	11.500	361
22	2.702	1.752	10.000	9.500	361
23	2.710	1.744	10.000	9.500	361
24	2.694	1.684	10.000	9.500	362
TOTAL	3.171	1.802	14.000	13.500	8669

CUMULATIVE PROBABILITY DISTRIBUTION  
MAY 1, 79, 80 - OCT 31, 79, 80

CATEGORY OP. LIMIT	CATEGORY POINTS	CATEGORY PERCENT	CUMULATIVE POINTS < LIMIT	CUMULATIVE PERCENT < LIMIT
0.500	39	0.4	39	0.450
1.000	1177	13.6	1216	14.025
2.000	1996	23.0	3212	37.047
3.000	1977	22.8	5189	59.850
5.000	2474	28.5	7663	88.385
7.000	772	8.9	8435	97.290
9.000	181	2.1	8616	99.377
12.000	48	0.6	8664	99.931
16.000	5	0.1	8669	99.988
23.000	0	0.0	8669	99.988
30.000	0	0.0	8669	99.988
40.000	0	0.0	8669	99.988
50.000	0	0.0	8669	99.988
70.000	0	0.0	8669	99.988

NUMBER OF VALID DATA POINTS = 8669  
NUMBER OF MISSING DATA POINTS = 163  
REPRESENTING 98.2-PERCENT VALID DATA

# IP2 DEFUELED SAFETY ANALYSIS REPORT

Job Number: 01-4122-00

TABLE 21  
YORK RESEARCH CORPORATION  
ONE RESEARCH DRIVE, STAMFORD, CONNECTICUT 06906  
CLIENT: CON EDISON CO. OF NY  
SITE: INDIAN POINT (122M)  
PARAMETER: WIND SPEED (SUMMER SEASON)  
UNITS: MPH

## DIURNAL ANALYSIS MAY 1, 79,80 - OCT. 31, 79,80

OBSERV #	PARAM AVG	PARAM STD DEV	MAX VALUE	RANGE	VALID PTS
1	7.821	5.161	30.000	29.500	357
2	7.595	5.070	24.000	23.500	356
3	7.581	5.153	26.000	25.400	355
4	7.413	5.097	27.000	26.500	356
5	7.233	5.043	32.000	31.400	353
6	7.436	5.383	35.000	34.500	354
7	7.499	5.616	47.000	48.400	355
8	7.332	5.442	47.000	48.400	359
9	7.717	5.320	38.000	37.400	360
10	8.086	5.399	33.000	32.000	362
11	8.881	5.312	30.000	28.500	361
12	9.595	5.217	28.000	26.000	359
13	10.232	5.394	40.000	39.000	358
14	10.409	5.450	36.000	34.500	359
15	10.762	5.485	30.000	28.000	357
16	11.087	5.397	32.000	31.000	355
17	11.262	5.184	31.000	30.400	353
18	10.764	4.780	25.000	23.500	354
19	10.266	4.919	33.000	32.000	354
20	9.879	5.015	28.000	27.000	354
21	9.256	5.022	27.000	28.400	356
22	8.781	5.299	28.000	27.400	356
23	8.383	5.140	28.000	27.400	356
24	8.070	4.944	26.000	25.500	357
TOTAL	8.888	5.388	47.000	46.500	8556

## CUMULATIVE PROBABILITY DISTRIBUTION MAY 1, 79, 80 - OCT 31, 79, 80

CATEGORY OP. LIMIT	CATEGORY POINTS	CATEGORY PERCENT	CUMULATIVE POINTS < LIMIT	CUMULATIVE PERCENT < LIMIT
0.500	7	0.1	7	0.082
1.000	169	2.0	176	2.057
2.000	511	6.0	687	8.029
3.000	638	7.5	1325	15.484
5.000	1303	15.2	2628	30.712
7.000	1205	14.1	3833	44.794
9.000	1132	13.2	4965	58.023
12.000	1565	18.3	6530	76.312
16.000	1258	14.7	7788	91.013
23.000	652	7.6	8440	98.633
30.000	99	1.2	8539	99.790
40.000	15	0.2	8554	99.965
50.000	2	0.0	8556	99.988
70.000	0	0.0	8556	99.988

NUMBER OF VALID DATA POINTS = 8556  
NUMBER OF MISSING DATA POINTS = 276  
REPRESENTING 98.9-PERCENT VALID DATA



# IP2 DEFUELED SAFETY ANALYSIS REPORT

Job Number: 01-4122-00

TABLE 22  
YORK RESEARCH CORPORATION  
ONE RESEARCH DRIVE, STAMFORD, CONNECTICUT 06906  
CLIENT: CON EDISON CO. OF NY  
SITE: INDIAN POINT (10M)  
PARAMETER: WIND SPEED (WINTER SEASON)  
UNITS: MPH

DIURNAL ANALYSIS  
NOV 1, 79, 80 - APR 30, 79, 80

OBSERV #	PARAM AVG	PARAM STD DEV	MAX VALUE	RANGE	VALID PTS
1	4.508	2.958	21.000	20.500	362
2	4.462	2.940	14.000	13.500	362
3	4.524	3.163	18.000	17.500	362
4	4.453	3.043	16.000	15.500	361
5	4.298	2.886	15.000	14.400	361
6	4.292	2.961	15.000	14.500	362
7	4.461	3.102	18.000	17.500	363
8	4.556	3.141	15.000	14.400	363
9	4.906	3.212	17.000	16.400	363
10	5.153	3.182	16.000	15.400	363
11	5.310	3.050	16.000	15.500	361
12	5.556	2.995	16.000	15.400	362
13	5.731	3.163	18.000	17.400	362
14	5.702	3.240	16.500	15.900	362
15	5.589	3.109	17.000	16.400	361
16	5.336	3.243	20.000	19.400	360
17	5.030	3.084	20.000	19.500	360
18	4.671	2.858	15.000	14.400	360
19	4.611	2.906	16.000	15.500	361
20	4.502	2.883	15.000	14.400	361
21	4.500	2.972	16.000	15.400	362
22	4.457	2.970	16.000	15.400	361
23	4.381	2.855	16.000	15.400	361
24	4.398	2.874	18.000	17.500	362
TOTAL	4.808	3.088	21.000	20.500	8678

CUMULATIVE PROBABILITY DISTRIBUTION  
NOV 1, 79, 80 - APR 30, 79, 80

CATEGORY OP. LIMIT	CATEGORY POINTS	CATEGORY PERCENT	CUMULATIVE POINTS < LIMIT	CUMULATIVE PERCENT < LIMIT
0.500	18	0.2	18	0.207
1.000	788	9.1	806	9.287
2.000	1332	15.3	2138	24.634
3.000	1178	13.6	3316	38.207
5.000	2122	24.4	5438	62.657
7.000	1591	18.3	7029	80.989
9.000	841	9.7	7870	90.679
12.000	614	7.1	8484	97.753
16.000	181	2.1	8665	99.839
23.000	13	0.1	8678	99.988
30.000	0	0.0	8678	99.988
40.000	0	0.0	8678	99.988
50.000	0	0.0	8678	99.988
70.000	0	0.0	8678	99.988

NUMBER OF VALID DATA POINTS = 8678  
NUMBER OF MISSING DATA POINTS = 34  
REPRESENTING 99.6-PERCENT VALID DATA

# IP2 DEFUELED SAFETY ANALYSIS REPORT

Job Number: 01-4122-00

TABLE 23  
YORK RESEARCH CORPORATION  
ONE RESEARCH DRIVE, STAMFORD, CONNECTICUT 06906  
CLIENT: CON EDISON CO. OF NY  
SITE: INDIAN POINT (122M)  
PARAMETER: WIND SPEED (WINTER SEASON)  
UNITS: MPH

## DIURNAL ANALYSIS NOV 1, 79, 80 - APR 30, 79, 80

OBSERV #	PARAM AVG	PARAM STD DEV	MAX VALUE	RANGE	VALID PTS
1	10.919	6.869	49.000	48.500	359
2	10.669	5.562	38.000	37.400	359
3	10.636	6.799	40.000	39.400	358
4	10.297	6.804	37.500	36.900	358
5	10.075	6.354	34.000	33.400	358
6	9.832	6.485	32.000	31.500	358
7	9.902	6.811	44.000	43.500	356
8	10.283	7.067	38.000	37.400	356
9	10.525	6.978	36.000	35.400	358
10	10.596	7.032	37.000	36.400	360
11	11.086	6.961	36.000	35.000	362
12	11.828	7.011	34.000	33.500	362
13	12.309	7.218	48.000	47.000	362
14	12.530	7.250	45.000	44.400	362
15	12.727	7.184	43.000	42.000	362
16	12.635	6.992	40.000	38.500	362
17	12.302	6.778	42.000	41.000	361
18	11.921	6.035	32.000	31.400	361
19	11.645	6.061	32.000	31.400	360
20	11.398	6.093	37.000	36.400	360
21	11.106	6.216	40.000	39.400	360
22	10.658	6.206	39.000	38.400	358
23	10.488	6.182	36.000	35.000	358
24	10.551	6.178	41.000	40.500	359
TOTAL	11.124	6.726	49.000	48.500	4629

## CUMULATIVE PROBABILITY DISTRIBUTION NOV 1, 79, 80 - APR 30, 79, 80

CATEGORY OP. LIMIT	CATEGORY POINTS	CATEGORY PERCENT	CUMULATIVE POINTS < LIMIT	CUMULATIVE PERCENT < LIMIT
0.500	6	0.1	6	0.070
1.000	105	1.2	111	1.286
2.000	418	4.8	529	6.130
3.000	458	5.3	987	11.437
5.000	910	10.5	1897	21.981
7.000	895	10.4	2792	32.352
9.000	1009	11.7	3801	44.044
12.000	1622	18.8	5423	62.839
16.000	1518	17.6	6941	80.429
23.000	1230	14.3	8171	94.681
30.000	372	4.3	8513	98.992
40.000	78	0.9	8621	99.896
50.000	8	0.1	8629	99.988
70.000	0	0.0	8629	99.988

NUMBER OF VALID DATA POINTS = 8629  
NUMBER OF MISSING DATA POINTS = 83  
REPRESENTING 99.0-PERCENT VALID DATA

## IP2 DEFUELED SAFETY ANALYSIS REPORT

TABLE 24  
MAXIMUM DIURNAL WIND SPEEDS (MPH)

<u>Season</u>	<u>Level (M)</u>	<u>1973-1974</u>	<u>1979-1980</u>
Summer	10	4.0	4.1
Summer	120	11.0	11.3
Winter	10	5.0	5.7
Winter	122	13.0	12.7

The median wind speeds extracted from Figure 20 at the 50 percent probability level are 1.1 M/S, 1.7 M/S and 1.4 M/S at the 10 meter level for the summer season, winter season, and annual basis, respectively. At the 122 meter level, these values are 3.4 M/S, 4.4 M/S and 3.9 M/S on the summer, winter and annual basis, respectively. These values bracket those presented by Kaplin and Laznow (1972) for the IP3 Tower. Corrected for exposure elevation no significant change would be expected between the two sets of values.

The variation of winds during the 1979-1980 season are consistent with data obtained during the 1973-1974 operational period. There is no reason to expect any significant variations with respect to the meteorology as used in FSAR 2.

### 3.2.4.5 Wind Velocities and Atmospheric Stability

#### 3.2.4.5.1 Joint Frequency Distribution of Wind Direction and Stability

Stability categorizations as referenced in this study are in accordance with NRC Pasquill Tables as derived from local temperature change with elevation. Except as noted, actual temperature measured gradients have been converted to °C/100M directly from temperature difference values (°F) per difference between sensor height levels.

Tables 25, 26 and 27 show the summary frequency distributions for the 10 meter level of wind direction and stability categories for the 1979-1980 data collection period. The tables show the annual, summer season and winter season summaries respectively.

# IP2 DEFUELED SAFETY ANALYSIS REPORT

TABLE 25  
ANNUAL SUMMARY OF WIND DIRECTION  
PERCENT FREQUENCY DISTRIBUTION AS A FUNCTION  
OF STABILITY - 10M LEVEL  
(JANUARY 1, 1979 - DECEMBER 31, 1980)

Wind Direction	Stability Class						
	<u>A</u>	<u>B</u>	<u>C</u>	<u>D</u>	<u>E</u>	<u>F</u>	<u>G</u>
N	1.28	0.36	0.48	3.39	2.67	0.50	0.09
NNE	1.76	0.40	0.46	3.15	3.33	0.80	0.17
NE	0.63	0.35	0.58	4.22	4.66	2.12	0.40
ENE	0.06	0.07	0.17	1.59	2.61	1.84	0.43
E	0.01	0.03	0.03	0.64	1.49	0.59	0.11
ESE	0.01	0.01	0.01	0.27	0.73	0.21	0.04
SE	0.03	0.01	0.02	0.23	0.67	0.26	0.02
SSE	0.09	0.03	0.04	0.45	1.04	0.31	0.05
S	2.04	0.25	0.29	1.74	3.39	0.76	0.11
SSW	2.58	0.51	0.38	2.14	5.04	0.72	0.05
SW	1.16	0.33	0.35	1.89	3.03	0.51	0.03
WSW	0.49	0.17	0.16	0.96	1.44	0.39	0.02
W	0.56	0.22	0.17	1.40	1.64	0.43	0.06
WNW	0.47	0.15	0.26	1.64	1.49	0.21	0.03
NW	0.70	0.31	0.32	2.36	1.85	0.10	0.01
NNW	0.80	0.40	0.49	3.26	1.60	0.17	0.04
CALM	0.00	0.00	0.00	0.00	0.00	0.00	0.00
MISSING	0.12	0.05	0.03	0.21	0.51	0.15	0.02
TOTAL %	12.80	3.66	4.23	29.56	37.17	10.08	1.69
NO. OF HOURS	2244	641	742	5183	6519	1768	297

# IP2 DEFUELED SAFETY ANALYSIS REPORT

TABLE 26  
SUMMARY OF WIND DIRECTION PERCENT FREQUENCY  
DISTRIBUTION AS A FUNCTION OF STABILITY  
SUMMER SEASON - 10M LEVEL  
(MAY 1, 1979, 80 - OCTOBER 31, 1979, 80)

Wind Direction	Stability Class						
	<u>A</u>	<u>B</u>	<u>C</u>	<u>D</u>	<u>E</u>	<u>F</u>	<u>G</u>
N	1.68	0.26	0.37	1.25	2.06	0.57	0.07
NNE	2.65	0.42	0.43	2.90	2.41	1.01	0.18
NE	0.58	0.31	0.46	3.46	4.44	3.17	0.35
ENE	0.11	0.10	0.24	1.38	2.66	2.62	0.39
E	0.02	0.07	0.01	0.57	1.57	0.61	0.05
ESE	0.01	0.01	0.00	0.31	1.01	0.36	0.06
SE	0.05	0.02	0.01	0.17	0.84	0.40	0.02
SSE	0.15	0.06	0.05	0.50	1.07	0.40	0.08
S	3.32	0.36	0.43	2.47	3.58	0.85	0.05
SSW	4.10	0.75	0.59	2.93	5.70	0.85	0.01
SW	1.84	0.49	0.48	2.23	3.03	0.51	0.05
WSW	0.87	0.20	0.18	0.94	1.05	0.34	0.00
W	0.88	0.28	0.19	1.38	1.42	0.34	0.07
WNW	0.80	0.09	0.25	0.94	1.03	0.15	0.05
NW	1.05	0.19	0.17	0.84	0.63	0.10	0.02
NNW	0.78	0.19	0.24	0.97	0.74	0.20	0.02
CALM	0.00	0.00	0.00	0.00	0.00	0.00	0.00
MISSING	0.22	0.06	0.01	0.31	0.68	0.22	0.03
TOTAL %	19.11	3.86	4.11	23.54	33.92	12.69	1.48
NO. OF HOURS	1687	341	363	2078	2994	1120	131

## IP2 DEFUELED SAFETY ANALYSIS REPORT

TABLE 27  
SUMMARY OF WIND DIRECTION PERCENT FREQUENCY  
DISTRIBUTION AS A FUNCTION OF STABILITY  
WINTER SEASON - 10M LEVEL  
(NOVEMBER 1, 1979, 80 - APRIL 30, 1979, 80)

Wind Direction	Stability Class						
	<u>A</u>	<u>B</u>	<u>C</u>	<u>D</u>	<u>E</u>	<u>F</u>	<u>G</u>
N	0.87	0.46	0.60	5.56	3.28	0.44	0.11
NNE	0.85	0.38	0.49	3.41	4.26	0.59	0.16
NE	0.86	0.40	0.69	4.99	4.88	1.06	0.45
ENE	0.01	0.05	0.10	1.79	2.56	1.06	0.48
E	0.00	0.00	0.06	0.72	1.40	0.56	0.18
ESE	0.00	0.01	0.01	0.23	0.45	0.06	0.02
SE	0.02	0.00	0.02	0.29	0.49	0.11	0.01
SSE	0.02	0.01	0.03	0.40	1.01	0.23	0.02
S	0.75	0.13	0.14	1.01	3.19	0.68	0.17
SSW	1.04	0.28	0.17	1.34	4.37	0.59	0.08
SW	0.48	0.16	0.22	1.55	3.03	0.51	0.02
WSW	0.10	0.13	0.14	0.99	1.84	0.45	0.03
W	0.24	0.15	0.14	1.42	1.86	0.52	0.06
WNW	0.13	0.22	0.28	2.34	1.95	0.28	0.02
NW	0.34	0.42	0.47	3.90	3.09	0.10	0.00
NNW	0.83	0.62	0.75	5.58	2.47	0.14	0.06
CALM	0.00	0.00	0.00	0.00	0.00	0.00	0.00
MISSING	0.02	0.03	0.05	0.11	0.33	0.09	0.01
TOTAL %	6.39	3.44	4.35	35.65	40.47	7.44	1.91
NO. OF HOURS	557	300	379	3105	3525	648	161

## IP2 DEFUELED SAFETY ANALYSIS REPORT

There are distinctive seasonal biases that coincide with variations in wind direction occurrence frequencies. It will be seen that these biases are consistent with variations in stability and wind speed on diurnal basis.

### 3.2.4.5.2 Frequency of Occurrence of Stability Categories

Table 28 shows a summary of historical comparison of percent occurrence of stability categories between the various reporting periods for the IP3 Tower and of the 122M Tower (IP4) for 1973/74. (Based on concurrent wind speed and temperature gradients). The former gradients were based on temperature differences from the 30M and 2M levels while the latter were based on differential measurements between the 60M and 10M levels.

On an annual basis there is generally good agreement between the results for the 122M Tower and the IP3 Tower composite year with temperature correction (FSAR 3, Supplement 13, 16).

The variation in percentages at the stability extreme A and G are most probably related to the lower gradient base level of measurement on the IP3 Tower - 2 meters. One would expect higher or lower temperatures closer to the ground with less accuracy in the thermal adjustment factor in these extreme ranges.

Table 29 shows a similar comparison for the 122 Meter Meteorological Tower for the 1973-1974 data collection period and the 1979-80 data collection period. These results are based on current wind speed data for the tower levels as noted. The 122M is shown based on two gradient differences: 122-10M and 122-60M. The percent occurrences of stability categories are sharply defined functions of season and, for the upper level, the vertical defined functions of season and, for the upper level, the vertical temperature gradient. This is apparent in both data sets.

There are seasonal differences in the two data sets particularly in the A and G stability category extremes at the 122M level, however, is noted that the percent occurrences in the 1973-1974 data set are based on only one or less than ten observations where these percentage differentials are most extreme. Where more data points are available in each stability category as for the lower gradient level (61-10M), there is generally good agreement.

## IP2 DEFUELED SAFETY ANALYSIS REPORT

TABLE 28  
HISTORICAL COMPARISONS OF  
PERCENT OCCURRENCE OF STABILITY

Temperature Gradient (M)	Stability Class							No. of Observ.
	A	B	C	D	E	F	G	
<u>Summer (1974)</u>								
61-10	21.44	5.46	4.37	27.92	30.21	9.28	1.31	2747
1973 (IP3)*	25.52	2.62	3.64	17.38	26.81	13.39	10.63	2935
<u>Winter (1973/74)</u>								
61-10	4.00	1.92	2.13	23.93	53.24	12.53	2.25	4797
1973 (IP3)*	21.49	3.52	3.74	23.29	23.76	14.24	9.96	4229
<u>Annual (1973/74)</u>								
61-10	10.35	3.21	2.94	25.38	44.86	11.35	1.91	7544
1970/72 (IP3) <sup>1</sup>	6.42	2.55	2.23	31.19	38.75	11.25	3.16	8366
1970 (IP3)*	21.68	2.20	3.39	33.35	24.75	9.01	5.62	NA
1971 (IP3)*	19.17	2.75	2.97	22.79	30.87	11.69	9.75	NA
1970/72 (IP3) <sup>1*</sup>	15.52	1.74	2.82	28.38	25.42	12.68	9.03	8366
1973 (IP3)**	23.14	3.16	3.70	20.87	25.02	13.89	10.23	NA

---

\* Concurrent basis

<sup>1</sup> Composite year with temperature correction - concurrent basis (FSAR 3)

<sup>1\*</sup> Composite year concurrent basis (FSAR 3)

\*\* March-December only concurrent basis

NA Not Available

NOTE: Gradient for IP3 Tower: 30-2 M



## IP2 DEFUELED SAFETY ANALYSIS REPORT

TABLE 29  
COMPARISON OF PERCENT OCCURRENCE OF STABILITY  
ON 122 METER TOWER

Date	Wind Freq	Stability Class							No. of Observ	Temperature Gradient(M)
	LVL (M)	A	B	C	D	E	F	G		
Summer										
1974	10	21.44	5.47	4.37	27.92	30.21	9.28	1.31	2747	61 - 10
1979/80	10	19.34	3.93	4.22	23.96	34.35	12.73	1.48	8557	60 - 10
1974	122	0.51	0.66	2.15	47.40	41.54	7.72	0.04*	2747	122 - 10
1979/80	122	4.54	5.50	5.81	42.84	34.95	6.23	0.14**	7838	122 - 10
1974	122	0.04*	0.04*	0.11**	20.13	74.01	5.42	0.25**	2747	122 - 61
1979/80	122	0.17**	0.42	1.49	68.09	26.72	2.82	0.29	7838	122 - 60
Winter										
1973/74	10	4.00	1.92	2.13	23.93	53.24	12.53	2.25	4797	61 - 10
1979/80	10	6.44	3.47	4.37	35.79	40.55	7.46	1.92	8648	60 - 10
1973/74	122	3.17	1.70	1.63	41.46	48.86	3.09	0.08**	4797	122 - 10
1979/80	122	0.59	1.18	2.78	56.07	34.75	4.29	0.31	8594	122 - 10
1973/74	122	0.02*	0.31	1.19	41.94	49.59	6.82	0.13**	4797	122 - 61
1979/80	122	0.28	0.14	0.37	71.89	25.35	1.84	0.13	8594	122 - 60
Annual										
1973/74	10	10.35	3.21	2.94	25.38	44.86	11.35	1.91	7544	61 - 10
1979/80	10	12.86	3.70	4.30	29.90	37.47	10.08	1.70	17205	60 - 10
1973/74	122	2.63	1.33	1.82	43.62	46.20	4.77	0.07	7544	122 - 10
1979/80	122	2.48	3.24	4.23	49.76	34.84	5.22	0.23	16432	122 -10
1973/74	122	0.03	0.08	0.80	34.00	58.48	6.31	0.17	7544	122 - 61
1979/80	122	0.23	0.27	0.91	70.08	26.00	2.31	0.21	16432	122 - 60

It may be inferred that, except as stated for the reasons noted, the percent frequency of stability classes with the existing tower system is representative and consistent with data referenced in FSAR 2.

---

\* Single observations

\*\* Less than ten observations

## IP2 DEFUELED SAFETY ANALYSIS REPORT

### 3.2.4.5.3 Average Wind Speed and Diurnal Variation as a Function of Stability Categories

Tables 30, 31, and 32 show the average wind speed and number of observations as functions of time of day and stability category for the summer and winter seasons of the combined 1979 and 1980 data collection period.

The results are derived from valid wind speeds measured at 10M relative to the temperature difference 60 - 10M and 122M wind speeds based on the temperature difference between 122-10M and 122-60M. The latter gradient was generated by subtraction of gradient levels:

$$(122-60) = (122-10) - (60-10)$$

These tables indicate a distinctive diurnal pattern to the stability. During the summer season at the 10M level for all practical purposes G stability does not occur between 0700 to 1900 EST and F stability does not occur between 0900 and 1400. During the nocturnal hours between 1900 to 0600, for all practical purposes A, B, and C stability categories do not appear. Stability Category A appears from 0700 to 1800 EST and is the dominant gradient between 0900 and 1600 EST. Out of 2,855 observations during this time interval, the percent frequency of occurrences were 52.9 for A, 8.1 for B, 8.4 for C, 24.2 for D, 6.1 for E, 0.2 for F, and 0.04 for G.

In the winter season at 10M, the A, B, and C stability categories do not appear during the nocturnal hours from 1900 to 0600 (with random singular exceptions). D is the dominant day time stability category and E is the dominant nighttime category.

Somewhat similar patterns are found with the upper level gradients based on the 122-10M and 122-60M temperature gradients. Between these latter two gradients

**TABLE 30**  
**Diurnal Variation of Stability Class and Wind Speed**  
**(concurrent data)**

**TABLE 31**  
**Diurnal Variation of Stability Class and Wind Speed**  
**(concurrent data)**

**TABLE 32**  
**Diurnal Variation of Stability Class and Wind Speed**  
**(concurrent data)**

## IP2 DEFUELED SAFETY ANALYSIS REPORT

TABLE 33  
COMPARISONS OF AVERAGE WIND SPEEDS (MPH)  
AS A FUNCTION OF STABILITY

Year/ Season	Anemom LVL	Stability Class							Temperature Gradient (M)
		A	B	C	D	E	F	G	
<u>Summer</u>									
1974	10M	4.2	4.4	4.0	3.8	2.7	2.3	2.6	60 - 10
1979/80	10M	4.0	3.6	3.5	3.7	2.7	2.1	2.2	60 - 10
1974	122M	4.3**	14.3	12.3	9.5	7.7	4.5	4.5*	122 - 10
1979/80	122M	11.2	10.4	9.8	10.0	7.6	4.8	4.3**	122 - 10
1974	122M	3.5*	49.0*	3.8**	10.6	8.2	4.4	2.6**	122 - 60
1979/80	122M	9.3**	12.6	12.1	10.0	6.4	4.3	4.0	122 - 60
<u>Winter</u>									
1973/74	10M	4.8	5.2	4.8	6.1	5.0	2.4	2.3	60 - 10
1979/80	10M	5.9	6.4	6.0	5.9	4.0	2.2	2.8	60 - 10
1973/74	122M	15.8	11.5	10.1	13.5	9.0	5.7	3.6**	122 - 10
1979/80	122M	13.9	12.4	13.9	12.9	8.6	5.6	8.6	122 - 10
1973/74	122M	16.0*	15.0	13.9	13.3	9.7	6.0	4.7**	122 - 60
1979/80	122M	19.9	21.0	20.4	12.3	7.8	7.2	6.3**	122 - 60

for the same data base the number of occurrences of A, B, and C stability categories in both summer and winter seasons are substantially reduced and become almost random when based on the 122-60M gradient. The obvious implication is that temperature gradient extremes are controlled by the surface level. This is consistent with Kaplan, et. al., 1974.

At the upper levels, for the summer season, F stability has a distinct diurnal function. It rarely occurs during the daytime from 0800 to 2000 EST. The distribution of G stability during the summer season, while a diurnal function, is clearly biased to the early morning hours. It occurs more frequently with relation to the 122-60M gradient than the 122-10M gradient. These factors are consistent with a nocturnal cool air surface drainage flow. This routine drainage flow does not exist during the winter season, and the occurrence frequencies, while still diurnal, are clearly related to the local surface air temperatures.

It is noted that during the winter seasons, drainage flow patterns are not routine occurrences. They can occur, however, during periods of weak pressure gradients. They are dependent on the horizontal and vertical temperature gradients that develop between the land (snow covered or bare ground) as well as the Hudson River (free water or ice bound).

The average wind speeds as a function of stability category are shown in Table 33 for the 1973-1974 data collection period and the 1979-1980 data period. As noted, there are some apparent anomalies with respect to the 122 meter level wind speeds. These are probably induced by the few data points available in 1973-1974 rather than a reality.

---

\* Single observations

\*\* Less than 15 observations

## IP2 DEFUELED SAFETY ANALYSIS REPORT

Where the total number of occurrences were in a more reliable range, at the 10 meter and the 122 meter levels, there was good agreement between average wind speeds.

Based on these data, it is concluded that within the range of normal climatological variations, there are no significant changes in the local meteorological parameters on a seasonal and annual basis. The existing meteorological data are consistent with data referenced in FSAR 2.

### 4.0 SUMMARY

### 4.1 METEOROROLOGY

#### 4.1.1 General

The meteorology of the Indian Point site and its environs has been thoroughly studied over the span of years. For the past nine years the source of on-site meteorology has been the 122 Meter Meteorological Tower that became fully operative as of October 1, 1973. This tower is located at latitude: 41° 15' 55" N and longitude 73° 57' 08" W (N38 + 31.453 and E22 + 49.473 on the Indian Point Grid).

A review of data and literature has revealed that the original Coordinates above were based on pre-1983 topographical maps that used North American Datum 1927 (NAD27) for its basis. The geodetic gurus revised the standards in the 1980s, and in the United States the USGS adopted the North American Datum 1983 (NAD83) model of the Earth's curvature, and the international community adopted WGS84 (World Geodetic System 1984) the following year, which is essentially the same model. WGS84 is the default map datum built into all GPS receivers, and is also the basis for most electronic maps including Google, MapQuest, Microsoft, Bing, etc. Also, depending upon the GPS unit or mapping system used, the numbers vary. The latitude and longitude coordinates for the met tower were only used to provide location information to the FAA when tower lights were out and the FAA no longer requests latitude and longitude information. Based on this information, the above coordinates should not be used to locate the primary tower.

Meteorological data from the 122 Meter Tower have, in previous studies and in this report, have been compared, in so far as possible, to those meteorological data, which were the data base for the FSAR 2 Report.

#### 4.1.2 122 Meter Meteorological Tower System

The 122 Meter Tower and support systems as presently comprised, maintained, and operated are in compliance with the meteorological measurement programs included in Regulatory Guides 1.23 and 1.97 and the criteria set forth in NUREG-0654, -0696, and -0737.

## IP2 DEFUELED SAFETY ANALYSIS REPORT

The system is outlined in Figure 4. It consists of an instrumented 122 Meter Tower. The critical sensors are for winds at the 10, 60, and 122 meter levels; ambient temperature and dew point at the 10 meter level and temperature difference between the 60-10 meter and 122-10 meter levels. In addition, a precipitation gage is located within the tower complex. All sensor signals are carried to a trailer, which houses:

- Signal Conditioners
- Analog Recorders
- Data Acquisition System - Magnetic Tape
- Terminal Printer
- LED Satellite Displays for Control Rooms 2 and 3
- Telephone Modems
- Dedicated Telephone Lines
- Air Conditioning Systems

All systems are operated on primary AC voltage. A backup diesel generator within the complex provides for the automatic transfer of power if the primary source is cut off.

Appropriate meteorological data are transmitted from the Meteorological Trailer to:

- Reactor Control Rooms 2 (CON-EDISON) and 3 (PASNY)
- MIDAS and ARAC Computers

The Emergency Control Center in addition to the computer systems and interrogation systems receives wind data from backup wind sensors.

The status of the backup wind system and the MIDAS computer may be assessed by remote telephone interrogation.

### 4.1.3 Local Meteorological Characteristics

All earlier studies and the data evaluated and included in this report indicates that the most important characteristic of the Indian Point area is the prevalence of winds from the north and south sectors. These winds are induced by meso-scale factors: terrain channeling at all times and drainage flow and land-sea circulations during periods of weak synoptic-cyclonic scale pressure gradient field.

At all wind levels there are distinctive diurnal variation patterns to local winds as well as to the local winds in addition to the local atmospheric stability as determined by vertical temperature gradients related to Pasquill stability categories. Unstable A, B, and C categories are dominant daytime occurrences. Stable F and G categories are nocturnal occurrences.

Because of the dominance of meso-scale factors in the Indian Point and lower Hudson Avenue Valley environs, persistent straight line flow of air from Indian Point is impossible. Paths of movement of air parcels are best generated by the use of local data on a real time basis. Recirculation of air parcels within a time frame of eight consecutive hours is a likely event. Within ten miles of Indian Point there are three zones, which indicated the probability of convergence and divergence of local surface air streams:

- Peekskill Bay
- Haverstraw Bay
- Tappan Zee

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### 4.1.4 Conclusion

From the evaluation of previous studies and the recent data years January 1, 1979 - December 31, 1980, it has been concluded that the meteorological data being collected at the 122 Meter Meteorological Tower are representative of the Indian Point site and are consistent with the original and expanded meteorological data basis of FSAR for Unit No. 2. All deviations of data at any given time (not otherwise specifically assigned to measurement techniques, methodologies, and, evaluation procedural changes to comply with existing Regulatory Guides) can be assigned to normal regional climatological variations in any given year on, at least, the synoptic-cyclonic meteorological scale.

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**NOTE: This information is classified as Historical Information**

### APPENDIX 2B

#### INDIAN POINT FSAR UPDATE

#### REVISED

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## INDIAN POINT FSAR UPDATE

### INTRODUCTION

This report is intended as an update and synthesis of previous geologic reports on the area surrounding Con-Edison's Indian Point nuclear facilities. The report reflects current thinking on the geology, structure, tectonic history, neotectonics and recent seismicity in the region. Main sources of information include Ratcliffe's (1976) final report on the Ramapo fault system, the Dames and Moore Geotechnical Investigation of the Ramapo Fault (1977), and the recent literature on the subject of the geology of the Manhattan and Reading Prongs.

Con Edison's Indian Point power plants are located in Buchanan, New York, on the east bank of the Hudson River. The site is situated in the central portion of the Peekskill Quadrangle.

### Physiography

The rocks in the vicinity of the Indian Point generating stations belong to three geologic provinces, the Hudson Highlands, the Manhattan Prong and the Newark Basin (Figure 1). Rocks that outcrop within the provinces range in age from Precambrian through Triassic (Jurassic?).

The landscape consists of northeast trending ridges and rather broad swampy valleys. Ridges are supported by bedrock and tend to follow prominent generally northeast, structural trends. Valley walls tend to be steep, the result of modification by Pleistocene glaciation. Elevations in the area reach a maximum of 1000 feet, and range from 50 to 300 feet above sea level in low lying areas.

### General Geology and Tectonics

The eastern third of the North American continent has been the site of episodic tectonism since Precambrian time (see Table 1). Paleozoic aged tectonism has molded a broad geologically varying zone known as the Appalachian orogen. The core of the orogen is marked by intrusive rocks modified by ductile and brittle deformation and regional metamorphism. The Indian Point site lies within the Manhattan Prong of the Appalachian Mountains.

The earliest recognized event in the area occurred in Precambrian time, and is known as the Grenville Orogeny. The Grenville orogeny, dated at 1.1 b.y., produced brittle and ductile deformation accompanied by regional granulite facies metamorphism and intrusive activity. The deformation and metamorphism affected the rocks of the Reading Prong and the Precambrian rocks of the Manhattan Prong. The Grenville events are not tectonically or temporarily related to the development of the Appalachian Orogen, which began in the latest Precambrian.

The earliest tectonic activity in the Appalachian Orogen probably involved latest Precambrian continental rifting and associated intrusive activity. The opening of the Proto-Atlantic Ocean (Iapetus) set the stage for the development of the Appalachian geosyncline. The geosyncline received sediments from earliest Precambrian through the mid-Ordovician time. The Taconic orogeny occurred in Mid-Ordovician time, and resulted

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in extensive thrust faulting, folding, metamorphism and intrusion in the northern Appalachians. The Taconic orogeny, generally interpreted as a continent-island arc collision was very intense in the Manhattan Prong region, and produced most of the structure evident in the current map pattern.

The Acadian orogeny (Devonian), possibly a continent-continent collision, was the next pulse of orogenic activity. The Acadian orogeny caused considerable deformation, metamorphism and intrusion in New England, but was not as intense as the Taconic orogeny in the Manhattan Prong.

The rocks of the Hudson Highlands (see Figure 1), an extension of the Reading Prong in New York State, consist of Precambrian gneisses and granites of Grenville (1.1 b.y.) age. The Manhattan Prong is underlain by Precambrian basement. An unconformity separates Cambro-Ordovician aged metasedimentary rocks from Precambrian rocks. The Newark Basin is filled with Triassic (Jurassic?) arkosic sediments diabase intrusives and basaltic flows.

### Geology of the Hudson Highlands

The Hudson Highlands outcrop in a northeast (040°) trending belt, approximately 10-miles wide, north, northwest and west of the Indian Point site (Figure 1). Four major rock types are present in the vicinity of Dunderburg Mountain, across the Hudson River from Indian Point. They are quartzo-feldspathic  $\pm$  calc-silicate hornblende gneiss; migmatitic quartzo-feldspathic biotite  $\pm$  garnet gneiss; calc-silicate bearing quartzite; and gneissic hornblende granite. Granite probably intruded the gneisses during Precambrian time.

Heleneck and Mose (1978) mapping in Highlands rocks near Lake Carmel, New York, recognize a mappable sequence of five rock units consisting of, gray migmatitic quartzo-feldspathic gneiss; amphibolite hornblende gneiss; leuco-granite gneiss and amphibolite; layered quartzo-feldspathic gneiss; and interlayered feldspathic quartzite and amphibolite. Highland rocks represent a sequence of Precambrian aged mio- and eugeosynclinal deposits, that have undergone a complex sequence of metamorphism and deformation. The rocks typically yield Rb/Sr ages of 1.1 billion years, the time of Grenville regional metamorphism. Mineralogic and textural evidence indicates that the rocks were metamorphosed to granulite facies, and multiply deformed during the Grenville orogeny. Recrystallization from granulite to amphibolite facies in Highland rocks near Lake Carmel accompanied folding during the Taconic orogeny (mid-Ordovician). Evidence of Taconic recrystallization in other areas of the Highlands remains equivocal. The Highlands are separated from the rocks of the Manhattan Prong and the Newark Basin by a complex fault system known as the Ramapo Fault Zone.

### Geology of the Manhattan Prong

The Manhattan Prong is a sequence of highly deformed metamorphic rocks, trending north-northeast, from New York City through Westchester County and western Fairfield County, Connecticut. The prong is bounded on the east by Cameron's Line, a complicated structure possibly representing a suture between two crustal blocks. On the west, the prong is bounded by the Newark Basin border fault and the Hudson River.

The stratigraphy of the Manhattan Prong has long been the subject of controversy and frequent revision, however, most recent workers recognize five formations within the prong. In order of decreasing age they are the Fordham Gneiss, the Yonkers-Pound Ridge Granite, the Lowerre Quartzite, the Inwood Marble and the Manhattan Schist (see Table

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2). The Fordham and Yonkers formations are Precambrian in age, and are separated from the Cambro-Ordovician aged Lowerre, Inwood and Manhattan formations by an angular unconformity. The relative ages of the Fordham and Yonkers formations are not known with absolute certainty. The Fordham is generally considered at least Grenville in age, and the Yonkers latest Precambrian.

Hall's (1968) detailed subdivision of the rocks of the Manhattan Prong near White Plains, New York, served as a basis for more recent workers. Correlations of the White Plains stratigraphy to other parts of the prong are tenuous due to the complex structural and metamorphic history of the area. Difficulties in correlation are compounded by the possibility of changes in original sedimentary facies.

The Fordham was divided into five members by Hall. They are:

Fordham A - Brown weathering garnet biotite quartz-feldspathic gneiss.

Fordham B - Gray garnet biotite quartz feldspar gneiss interlayered with amphibolite.

Fordham C - Gray biotite hornblende quartz-feldspathic gneiss with some amphibolite.

Fordham D - Rusty weathering sillimanite-garnet biotite quartz-feldspathic gneiss.

Fordham E - Siliceous biotite-quartz plagioclase gneiss.

Most of the Fordham rocks probably represent metamorphosed eugeosynclinal deposits, interbedded with mafic volcanics. The Fordham formation was deformed and metamorphosed to granulite facies during the Grenville orogeny. Mineralogic and structural evidence indicates that the Fordham was recrystallized and deformed during Plaeozoic orogenesis.

The Yonkers Granite is thought to represent a metamorphosed rhyolite, emplaced during the opening of the Proto-Atlantic in late Precambrian time.

The assignment of formation status to the Lowerre, has been the subject of debate for nearly 100 years. It was first named and described by Merrill in 1896, but many workers in the 20<sup>th</sup> century preferred to consider the Lowerre as part of the Fordham. In addition, the unconformity between the Cambro-Ordovician rocks and the Fordham, has not been recognized by all workers in the prong. The Lowerre is a relatively thin (40-ft. thickness), discontinuous unit representing an arkosic sandstone. The discontinuity of the unit is probably the result of deposition on an irregular Precambrian aged erosional surface. The Lowerre consists of quartz, with considerable amounts of potassium feldspar and minor biotite. The Lowerre is always observed in the same stratigraphic position, at the base of the Cambro-Ordovician aged cover rocks, overlying and truncating various members of the Fordham.

The Inwood Marble, consisting of dolomite and calcite marble interlayered with calc-silicate schists, overlies the Lowerre. Hall has divided the Inwood into five members, that lens in and out. In map pattern, the Inwood does not appear to be continuous, the result of tectonic thinning of fairly ductile marble during deformation. The Inwood represents

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deposition on a carbonate bank, widespread in the Appalachian orogen during the late Cambrian and early Ordovician.

In the White Plains area, Hall recognizes three mappable members of the Manhattan, 'A', 'B', and 'C' (see Table 2). Manhattan 'A' (basal Manhattan) is a fissile sillimanite garnet biotite schist, interlayered with marble and calc-silicate schist. The Manhattan 'A' may be a transitional facies between carbonate and clastic sedimentation. It is recognized by Ratcliffe (1976) near the Cortlandt complex, and by Brock (1977) north of White Plains. Manhattan 'B' is a discontinuous amphibolite and Manhattan 'C' is a brown weathering, garnet muscovite biotite schistose gneiss. The Manhattan Formation was originally deposited in a miogeosyncline, and represents pelites, mafic volcanics and greywackes. The Manhattan Prong was metamorphosed, deformed and intruded during two major orogenic episodes, the Taconic (late Ordovician) and the Acadian (Devonian). A late Acadian metamorphic and deformational event can be recognized in some locations within the prong (Brock and Mose 1979). Mose and Hall (1979) infer a mid-Ordovician unconformity within the New York City group based on structural and isotopic evidence.

Brock (1977) has worked out a detailed sequence of events for the Manhattan Prong near Croton Falls, New York (see Table 3). The rocks of the prong were metamorphosed to K-feldspar sillimanite grade (upper amphibolite facies) at the peak of the Taconic orogeny. At this time the rocks underwent intense deformation, reflecting the effects of four distinct fold events during the orogeny. Taconic aged recrystallization affected the Precambrian rocks of the Manhattan Prong, destroying most Grenville aged metamorphic and structural features. Granulite facies mineralogy and textures survive as relicts within the Fordham, but are not present in the Cambro-Ordovician rocks, supporting the inferred unconformity between Fordham and younger rocks.

During Silurian time, deformation eased, but the prong was intruded by the Croton Falls and Cortlandt mafic complexes. The Acadian orogeny (Devonian) produced another set of folds ( $F_5$ ), metamorphism of kyanite-staurolite (mid-amphibolite facies) grade, and the intrusion of the Peekskill Granite. The "final" Paleozoic metamorphic and deformational event occurred late in the Acadian Orogeny or during the Mississippian, causing local retrograde metamorphism (muscovite grade) and folding ( $F_6$ ) (Brock, 1977). The "final" orogenic event is seen locally as tight isoclinal folds. Late metamorphism is evidenced by recrystallization on undeformed joint faces.

The ductile deformation occurred coevally with brittle deformation along the border fault and within the Manhattan Prong. The relationship of Precambrian, Paleozoic and post Paleozoic aged faulting will be discussed in a following section.

One of the significant problems involving rocks of the Manhattan Prong is correlation on a regional scale (see Table 2). The Fordham Formation is often correlated with the Precambrian Highlands gneisses. Some workers have tried tracing the Highlands across the boundary fault, comparing structure and metamorphic details with Fordham rocks. Correlation is tenuous since the Grenville age yielded by the Highlands and Fordham rocks, is a metamorphic age, not a time of deposition. Correlation of the rest of the New York City Group with rocks of the surrounding region is based on similarities in lithology, structural position, radiometric age determinations and fossil evidence. The Lowerre is considered the metamorphosed equivalent of the Poughquag Sandstone. The Inwood is correlated with the Wappinger Limestone Group, and the Manhattan with the Annsville Phyllite and Hudson River Shale. Metamorphism increases from chlorite grade near the Hudson River, to K-feldspar-sillimanite grade near the Connecticut border.

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### The Geology of the Newark Basin

The third geologic province in the area is the Newark-Gettysburg Basin. The basin extends 140-miles from York County, Pennsylvania, to Rockland County, New York (Figure 1). The basin, a down dropped crustal block, formed during Mesozoic time. Deposition was continuous from the late triassic through the upper Jurassic (Dames and Moore, 1977). Intrusion of the Palisades sill apparently occurred during deposition of sediments in latest Triassic-earliest Jurassic. The extrusion of the Watchung basalt flows followed later in the Jurassic. Rocks of the Newark series are in contact with the crystalline rocks of both the Manhattan and Reading Prongs, but the nature of the contact varies. At the northeastern edge of the basin, Triassic sediments unconformably lie over the Highland rocks, while the northeastern edge of the basin is in fault with the rocks of the Highlands.

The Newark Group is divided into four formations, the Hammer Creek Conglomerate, the Stockton Arkose, the Lockatong Argillite, and the Brunswick Shale and Sandstone. Deposits of conglomerate and sedimentary breccia lie at the edges of the basin, reflecting proximity to the uplifted Precambrian and Paleozoic rocks that are the sources of the Triassic aged sediments.

The boundary fault between the Newark series and older crystalline rocks is the Ramapo fault. Movement along the fault and subsidence of the basin, concurrent with sedimentation, produced the half-graben configuration of the basin. The rocks within the basin are not greatly deformed, displaying broad open folds of uncertain origin, gentle dips of strata, and minor faults with small offsets.

### History of Brittle Deformation

A series of north-northeast trending faults pass through the area surrounding the Indian Point sites. The faults, some of which have been episodically active since the Precambrian time are collectively known as the Ramapo fault system. The system is composed of a number of parallel to sub-parallel branches and draws its name from the Ramapo fault, the boundary between the Reading Prong and the Newark Basin.

Ratcliffe (1976) mapped the faults in the vicinity of the Indian Point site, and interpreted a chronologic sequence of fault movements. Ratcliffe classified faults utilizing radiometric ages, cross-cutting lithologic relationships, and textural evidence. Dames and Moore (1977) utilize Ratcliffe's conclusions, and present evidence for timing fault movements based on geothermometry of fluid inclusions in calcite. More recently, Nelson (1980) and others have examined the stratigraphy and pollen remains in swamps and sag ponds along the Ramapo, seeking evidence of Post-Pleistocene faulting in the area. In addition, Aggarwal and Sykes (1978), Yang and Aggarwal (1981) Dames and Moore (1977), and Woodward-Clyde Consultants (Quarterly Reports, Jan 1977 - Jan 1982) have studied the recent seismicity in the region, solving for magnitude and location of earthquake epicenters.

The earliest documented movement along the Ramapo fault system is Grenville age (Ratcliffe 1976). Textural evidence seen in the Canopus Pluton indicates that movement along the Canopus Hollow fault (north of Peekskill) was synchronous with emplacement of the pluton. Flow structures and mylonitization displayed within the pluton indicate crystallization during shearing. Drag folds and the overall shape of the pluton suggest right lateral strike-slip motion. The Canopus Pluton, a diorite-monzontie, has been radiometrically dated at 1150 m.y. (Rb/Sr), thus providing a minimum age for movement along the fault.



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The Lake Peekskill fault is defined by a shear zone in the Precambrian gneisses that has not affected the Annsville Phyllite (Ordovician). The fact that the Annsville has not been deformed, places a limit on the last motion along the fault. Ratcliffe (1976) states that both the Canopus Hollow and Lake Peekskill faults were not reactivated during Paleozoic time, however, in a more recent article (Ratcliffe 1980) he suggests that movement as recent as Triassic has taken place within the Canopus Fault Zone.

Dames and Moore (1977) disagree with Ratcliffe's 1976 opinion, citing as evidence a sheared inlier of Poughquag Quartzite near Canopus Lake. The shear zone is considered by Dames and Moore to reflect thrust faulting along a northeast trending fault, sub-parallel to the Canopus Hollow fault. This implies post-Precambrian reactivation along the Canopus Hollow Fault. An additional strike-slip shear zone was mapped by Dames and Moore in Highland rocks, near the southwest corner of Canopus Lake, along the strike of the Canopus Hollow fault. This shear is considered additional evidence for reactivation of the Canopus Hollow fault in Paleozoic time. It is important to note that local activity occurring along part of a fault does not require movement along the entire length of a fault. Furthermore, the shear zone that displaces the Poughquag is not necessarily an extension of the Canopus shear zone, and may in fact be related to Paleozoic aged folding. Thus, Paleozoic aged movement along the Canopus Hollow fault is not required in the vicinity of Indian Point.

A number of faults of Paleozoic age separate the Manhattan Prong from the Hudson Highlands. Most prominent are the Thiells fault, the Annsville fault, the Peekskill fault and the Croton Falls fault. The Peekskill and Croton Falls faults outcrop on the east bank of the Hudson River, and generally trend east-west. The Thiells fault, outcropping on the west bank of the Hudson, and the Annsville fault on the east bank trend northeast. The Ramapo fault extends northeast from Peapack, New Jersey separating the Newark Basin from the Reading Prong. At Ladentown, it splays into two branches, trending 020° and 060°, respectively. The 060° branch connects with the Thiells fault, and the more northerly trending branch extends into the Highlands, through Tomkins Cove, New York.

The Peekskill, Croton Falls, Thiells and Annsville faults are primarily Paleozoic in age. The faults are marked by mylonite and ultra-mylonite, displaying retrograde chlorite grade, green schist facies mineral assemblages. Movement along the faults is generally right-lateral-strike-slip. Mid-Ordovician minimum ages of movement are inferred from cross-cutting relationships with dikes related to the Rosetown Pluton (mid-Ordovician), and radiometric ages of undeformed biotites from within shear zones (Ratcliffe, 1976). A lower Devonian K-Ar age of 396 m.y. places the most recent probable movement along the Roa Hook branch of the fault squarely within the Acadian orogeny. Similar data is available for the Thiells fault.

Younger faults can be distinguished from Paleozoic aged and older faults (Ratcliffe, 1980) by their different mineralization, and cataclastic textures. Younger faults are characterized by open work breccias, clay gouge, platy fracture and deeply incised fault scarps. Older faults display healed breccia, semi-ductile mylonite shear zones, and higher temperature minerals that reflect the general pattern of Paleozoic aged regional metamorphism.

Reactivation and development of new faults occurred during Mesozoic time (Ratcliffe, 1980). Deep seated zones of weakness in crystalline basement were utilized in the development of the Triassic basin, particularly the Ramapo-Cheesecote and Mott Farm Road faults (Ratcliffe, 1976). Structural evidence indicates that normal faulting was dominant during Mesozoic time, with the latest activity along the Mott Farm Road branch

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of the Ramapo, dated at 163 m.y. by K-Ar methods (Ratcliffe, 1976). Late north-south vertical strike-slip faults with both left-lateral and right-lateral movements are present in the vicinity of Indian Point. The relationship of these faults to the Ramapo is uncertain, although the faults are probably Mesozoic in age (Ratcliffe, 1976; Dames & Moore, 1977).

Detailed work (Ratcliffe, 1976) shows that north-south faults at Tomkins Cove across the river from Indian Point are the youngest in the area. Mineralogy and textures found in the young, Tomkins Cove faults bear a strong resemblance to faults that have been radiometrically dated as Mesozoic. In addition, the time of last movement along the faults is constrained by the lack of fault related deformation in overlying Pleistocene sediments (Ratcliffe, 1976, 1980; Dames and Moore, 1977). A group of faults located at the Indian Point site was mapped in detail by Dames and Moore (1977). Displacement along the faults is not significant, no more than a few feet. The faults are filled with undeformed euhedral calcite crystals, many of which contain fluid inclusions. Temperature equilibrium studies on the fluid inclusions indicate average formation temperatures of 160°C. Dames and Moore (1978), infer a depth of formation, by applying the geologically conservative geothermal gradient of 50°C/km. This yields a temperature of 150°C at 3 km.

The amount of time necessary to expose rocks that form at a depth of 3 km is a function of denudation rates. A minimum of 45 m.y. is required to remove 3 km of material, if the rather rapid denudation rate of 15,000 yrs/meter is applied (Dames and Moore, 1978). This calculation sets another constraint on the possible minimum age of last movement on the faults. The growth of calcite in the fault zones has been attributed to circulating hydrothermal fluids related to Mesozoic igneous activity (Dames and Moore, 1978), suggesting a time of last movement in the Mesozoic.

Radiometric age determination of undeformed minerals that have grown within fault zones (Ratcliffe, 1976) and the lack of fault related deformation of Pleistocene deposits and surface features (Dames and Moore, 1977); Ratcliffe 1976, 1980), provide the best evidence that the faults in the Indian Point area have not moved in the last two million years. Data from recent drill cores that intercepted the Ramapo fault plane, show that the dip of the structure is highly variable. The cores, taken at four locations along the fault, indicate that the dip is consistently to the southeast, ranging from 45° through 70° (Ratcliffe, 1980). Textural evidence observed in the cores, indicate that the dominant latest motion in the fault has been right oblique normal faulting (Ratcliffe, 1980).

### Recent Seismicity

In the last twenty years, the catalogue of instrumentally recorded seismic events in the northeast has grown tremendously. Locally, this is the result of a dense network of seismic stations, situated in the area around Indian Point, that has been operated since 1975. Data collection by regional seismic stations has continued in the same area since 1970 with a reported detection threshold of about magnitude 2 mb (Lg). Seismic networks have provided a basis for accurately determining the location, magnitude, and in some case focal mechanism solution for many small magnitude seismic events in the area of the power plant and the Ramapo fault zone. This recent seismicity is not markedly dissimilar from the historical seismicity reported for the region. The composite data set does not define or suggest a structural association of earthquakes, however, a regional overview does suggest a higher level of activity in the northern New Jersey, southeastern New York area than that of the surrounding areas.

A number of hypotheses have been proposed to explain the observed pattern of seismicity. Current seismicity in the northeastern United States has been attributed to: a proposed



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stress system in which the maximum compressive stress trends to the northwest (Yang and Aggarwal, 1981); proximity to relatively young igneous bodies (McKeown, 1978); the response of the crust to glacial unloading (Stein and others, 1979); and, in the immediate region about the site, proposed reactivation of pre-existing fault zones based on a spatial correlation with surface traces of faults exposed within the region (Aggarwal and Sykes, 1978).

Earthquakes occurring near Indian Point have been characterized as shallow focus (<10 km) and low magnitude (1.0-3.0) (Aggarwal and Sykes, 1978). Focal mechanism solutions reported for earthquakes near the Ramapo fault or the margins of the Triassic basin indicate thrust movement on faults that parallel the dominant structural grain in the exposed bedrock (Figure 2) (Aggarwal and Sykes, 1978; Yang and Aggarwal, 1981). The stress field required for this interpretation, must be compressional and oriented to the northwest (Yang and Aggarwal, 1981). Whereas some of the seismic activity in southern New York may be related to northeast trending structures, a number of other trends, transversely oriented with respect to the dominant structures are present (Ratcliffe, 1976; Pomeroy et al, 1976; Blackford and Statton, 1978; Quarterly Report for the Indian Point Seismic Monitoring Network, November 1979 through January 1980) (see Figure 2). Thompson and Bebel (1979) describe northeast and northwest trends of epicenters in the coastal plain area of New Jersey, Delaware and Pennsylvania.

Many studies have attempted to quantify the stress regimes operating in the vicinity of the Ramapo fault. Dames and Moore (1977) determined the near surface stress by in-site measurements. While these results are variable, a fairly consistent northeast to eastwest trend for the horizontal component of compression was determined. This stress direction is transverse to that suggested by others (Aggarwal & Sykes, 1978) who base their interpretations on reported focal mechanism solutions. It is, however, in general agreement with regional stresses inferred from measurements and observations made throughout the northeastern United States (Sbar and Sykes, 1973).

An examination of the distribution of earthquakes in the vicinity of Indian Point indicates that not all earthquakes in the region can be attributed to northeast trending faults. A sequence of earthquakes occurred near Annsville, New York, from 17 January to 23 January 1980. A composite focal mechanism solution was constructed using data recorded by the Indian Point Seismic Monitoring Network. The solution indicates thrust faulting along one of two possible planes, oriented N2°W 29°E or N16°W 62°W (Quarterly Report for the Indian Point Seismic Monitoring Network, November 1979 - January 1980). This trend is obliquely oriented to the dominant structural fabric in the region, and requires a compressive stress field oriented east-west to northeast-southwest.

Low level microseismicity existing in the region is evidence that crustal adjustments are continuing in response to regional stresses. No evidence exists at the surface or in drill cores of the fault zones (in particular the Ramapo fault zone), that suggest any contemporary movement along faults exposed at the surface since the major period of activity during Mesozoic time. In fact geologic data obtained from cores of the Ramapo fault zone (Ratcliffe, 1980) show evidence of normal faulting as the last movement, which is consistent with the Mesozoic faulting regime and inconsistent with the thrust mechanism proposed by Aggarwal and Sykes (1978). To date, no satisfactory stress regime has been proposed that adequately accommodates the observed pattern of low level seismicity, regional stress measurement data and those stresses inferred from recently reported focal mechanism solutions.

## CONCLUSION

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Low level microseismicity in the region is evidence that crustal adjustments are continuing in response to regional stresses. Detailed field investigations (e.g., Ratcliffe, 1976, 1980; Dames and Moore, 1977) have been conducted in the immediate vicinity of Indian Point, and along the major faults in the region. To date, no evidence has been found in the rocks exposed at the surface or sediments overlying fault traces or in cores obtained in the vicinity of Indian Point, that might support a conclusion that displacement has occurred along major fault systems within the New York Highlands, the Ramapo or its associated branches during Quaternary time (the last 1.5 m.y.). In the vicinity of Indian Point, evidence that no displacement has occurred in the last 65 m.y. (since the Mesozoic) along specific major structures has been observed.

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### GLOSSARY

**amphibole** - A complex chain silicate mineral rich in iron and magnesium.

**amphibolite** - A metamorphic rock whose main components are amphibole and plagioclase feldspar

**amphibolite facies** - Rocks formed at moderate temperature and pressure conditions during regional metamorphism.

**angular unconformity** - An unconformity recognizable by the deposition of sediments over deformed rocks.

**argillite** - A compact mudstone, generally not laminated and not fissile.

**arkose** - A feldspar rich, generally coarse grained, sandstone derived from continental rocks.

**breccia** - A generally coarse grained rock composed of angular or broken fragments of rock, which may be formed tectonically in a fault zone, or by sedimentary processes.

**brittle deformation** - A term used to describe faulting and fracturing of rocks.

**calc-silicate** - A descriptive term applied to minerals or rocks consisting of calcium bearing silicates, such as diopside.

**chlorite** - A green iron-magnesium rich platy mineral.

**chlorite grade metamorphism** - Low-grade regional metamorphism indicated by the first appearance of chlorite in rocks of appropriate composition.

**dike** - An igneous intrusion that cuts across planar features of a rock.

**disconformity** - A break in the time-stratigraphic record separating two sequences of rock, both of which are bedded parallel to the unconformity.

**ductile deformation** - Occurs where rocks fold or flow when subjected to a stress field.

**eugeosyncline** - A geosyncline or basin in which vulcanism is associated with clastic sedimentation.

**euohedral** - A crystal bounded by well developed crystal faces.

**facies** - A set of conditions that specify the environment of formation of rocks (metamorphic or sedimentary).

**geosyncline** - A long linear basin, characterized by subsidence coincidental with sedimentation.

**geothermal gradient** - The relationship of temperature to depth (pressure) in the earth's crust.

**gneiss** - A metamorphic rock formed by regional metamorphism, generally high grade.

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**granulite facies** - Rocks formed at very high temperatures and pressures during regional metamorphism.

**intrusion** - The emplacement of an igneous body in a pre-existing rock.

**isoclinal folds** - A fold in which the limbs are parallel.

**mafic** - A term used to describe dark rocks or minerals containing large amounts of magnesium or iron.

**marble** - A metamorphic rock consisting primarily of calcite or dolomite.

**metamorphic grade** - Rocks of any composition that have been metamorphosed under a specific range of temperature and pressure conditions.

**metasediment** - A metamorphosed sedimentary rock.

**migmatite** - A mixed rock composed of metamorphic material containing segregation of igneous material formed by injection or in-situ partial melting.

**miogeosyncline** - A geosyncline lacking volcanic deposits, commonly located adjacent to continental margins.

**neotectonics** - Post-Miocene structural history of the earth's crust.

**orogen** - A region that has been subjected to orogeny.

**orogeny** - The development of structures, metamorphism and igneous activity relating to mountain building.

**pelite** - A sedimentary or metamorphic rock rich in aluminum.

**quartzite** - A metamorphic or sedimentary rock consisting mainly of quartz.

**retrograde metamorphism** - Recrystallization of metamorphic rocks at conditions that are lower grade than those at which the rock was originally metamorphosed.

**schist** - A well foliated metamorphic rock that easily separates into flakes or slabs due to an abundance of platy minerals.

**sedimentary facies** - An restricted area within a litho-stratigraphic unit representing a particular depositional environment.

**sill** - An igneous intrusion that parallels the layering of the country rock.

**tectonic** - Pertaining to the forces that cause crustal deformation.

**throw** - The vertical component of fault motion.

**unconformity** - A primary feature representing erosion or non-deposition, resulting in a break in the stratigraphic sequence.

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### TABLE 1 GEOLOGICAL TIME SCALE

### TABLE 2 PROPOSED CORRELATION OF THE STRATIGRAPHIC SUBDIVISIONS IN THE MANHATTAN PRONG WITH ROCKS IN ADJACENT AREAS

### TABLE 3 GEOLOGIC HISTORY IN THE CROTON FALLS AREA (after Brock & Mose, 1979)

### FIGURES

Figure No.	Title
Figure 1	Geological Map, Southeastern New York (after Brock & Mose, 1979)
Figure 2	Seismicity of Southeastern New York and New Jersey (after Dames & Moore, 1977)

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## CHAPTER 3 FACILITY DESIGN AND OPERATION

### 3.0 Introduction

Section 3.1 discusses the nuclear fuel.

Sections 3.2 through 3.11 discuss the auxiliary systems required to ensure the safe operation or servicing of the spent fuel pit.

These sections consider systems in which component malfunctions, inadvertent interruptions of system operation, or a partial system failure may lead to a hazardous or unsafe condition. The extent of information provided for each system is proportional to the relative contribution of, or reliance placed upon, each system with respect to the overall facility operational safety.

The following systems are considered under this category:

#### Chemical and Volume Control System

This system supports the transferring and storage of liquid radwaste.

#### Auxiliary Coolant System

This system provides for transferring heat from the stored spent fuel and other components to the service water system and consists of the following two loops:

1. The spent fuel pit loop removes decay heat from the spent fuel pit.
2. The component cooling loop cools the spent fuel pit water.

#### Sampling System

This system provides the equipment necessary to obtain liquid and gaseous samples from facility systems.

#### Facility Service Systems

These systems include fire protection, service water, and auxiliary building ventilation.

#### Fuel Handling System

This system provides for handling fuel assemblies.

#### Equipment and Decontamination Processes

These processes provide for the removal of radioactive deposits from system surfaces.

#### Primary Auxiliary Building Ventilation System

This system maintains ambient operation temperatures and provides purging of the auxiliary building to the plant vent.



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### Control Room Ventilation System

This system maintains the required environment in the control room.

### Circulating Water System

This system provides dilution flow for liquid waste discharges.

Section 3.12 discusses the leakage provisions regarding the cooling loops.

Section 3.13 discusses information displays and alarms.

Section 3.14 discusses the communications systems.

Section 3.15 discusses the Electrical Systems.

Section 3.16 discusses the containment systems.

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### 3.1 Nuclear Fuel

The fuel rods are cold worked Zircaloy-4 or ZIRLO™ tubes containing slightly enriched uranium dioxide fuel.

The fuel assembly is a can-less type with the basic assembly consisting of the rod cluster control guide thimbles attached to the grids and the top and bottom nozzles. The fuel rods are held by the spring clip grid in this assembly, which provides support for the fuel rods.

High parasitic (HIPAR) fuel was used for the initial fuel and reload fuel through Cycle 4. Low parasitic (LOPAR) fuel was loaded for Cycles 5 through 9, and optimized fuel assemblies (OFA) were loaded for Cycles 10, 11, and 12. For Cycles 13, 14 and 15, 15x15 VANTAGE+ fuel assemblies were loaded as the feed fuel. For Cycle 16, 15x15 VANTAGE+ fuel assemblies with Performance+ enhancements were loaded as feed fuel. For Cycles 17 through 24, 15x15 Upgraded fuel design assemblies were loaded as feed fuel.

Rod cluster control assemblies and wet annular burnable absorber rods are inserted into the guide thimbles of the fuel assemblies. The absorber sections of the control rods are fabricated of silver-indium-cadmium alloy sealed in stainless steel tubes.

#### 3.1.1 Fuel Assemblies

The fuel assemblies are designed to perform satisfactorily throughout their lifetime. The assemblies are structurally designed to maintain sufficient integrity to permit safe removal from the core, subsequent handling during cooldown, shipment, and fuel reprocessing.

The fuel rods are supported at nine locations along their length within the fuel assemblies by grid assemblies, which are designed to maintain control of the lateral spacing between the rods through the design life of the assemblies. The magnitude of the support loads provided by the grids is established to minimize possible fretting without overstressing the cladding at the points of contact between the grids and fuel rods and without imposing restraints of sufficient magnitude to result in buckling or distortion of the rods. In addition, there are three Intermediate Flow Mixing (IFM) grids spaced along the fuel assembly and a protective grid (P-grid) on the bottom of the assembly. These grids do not provide any support function.

The fuel rod cladding is designed to maintain encapsulation of the fuel throughout the design life.

#### 3.1.2 Rod Cluster Control Assemblies

The criteria used for the design of the cladding on the individual absorber rods in the rod cluster control assemblies are similar to those used for the fuel rod cladding. The cladding is designed to be free standing under all operating conditions and will maintain encapsulation of the absorber material throughout the absorber rod design life.

#### 3.1.3 Mechanical Design

The fuel is in the form of slightly enriched uranium dioxide ceramic pellets. The pellets are stacked to an active height of 144-in. (previously 142 in.) within ZIRLO™ (previously Zircaloy-4) tubular cladding, which is plugged and seal welded at the ends to encapsulate the fuel.

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The control rods, designated as rod cluster control assemblies, consist of groups of individual absorber rods, which are held together by a spider assembly at the top end and actuated as a group. In the inserted position, the absorber rods fit within hollow guide thimbles in the fuel assemblies. The guide thimbles are an integral part of the fuel assemblies and occupy locations within the regular fuel rod pattern where fuel rods have been deleted. Figures 3.1-1 and 3.1-2 show a typical rod cluster control assembly in a fuel assembly.

### 3.1.3.1 Fuel Assembly

The assemblies are square in cross section, nominally 8.426-in. on a side, and have an overall height of approximately 159.975 inches. The fuel rods in a fuel assembly are arranged in a square array with 15 rod locations per side and a nominal centerline-to-centerline pitch of 0.563-in. between rods. Of the total possible 225 rod locations per assembly, 20 are occupied by guide thimbles for the rod cluster control rods and one for incore instrumentation. The remaining 204 locations contain fuel rods. In addition to fuel rods, a fuel assembly is composed of a top nozzle, a bottom nozzle, grid assemblies, 20 absorber rod guide thimbles, and one instrumentation thimble. Additional information regarding the fuel rods and fuel assemblies is provided in Table 3.1-1.

The guide thimbles in conjunction with the grid assemblies and the top and bottom nozzles comprise the basic structural fuel assembly skeleton. The grid assemblies are bulge attached to the guide thimbles at each location along the height of the fuel assembly at which lateral support for the fuel rods is required. Within this skeletal framework the fuel rods are contained and supported and the rod-to-rod centerline spacing is maintained along the assembly.

The original fuel design for Indian Point 2 was the Westinghouse High Parasitic (HIPAR) fuel assembly (Figure 3.1-4). This consisted of Zircaloy-4 clad fuel rods, nine Inconel grids and stainless steel instrumentation and guide thimbles. Burnable absorbers used were Pyrex glass.

Starting with Cycle 5, the Westinghouse Low Parasitic (LOPAR) fuel assembly (Figure 3.1-5) was introduced. This design consisted of Zircaloy-4 clad fuel rods, nine Inconel grids and Zircaloy-4 instrumentation and guide thimbles.

For Cycle 8, Wet Annular Burnable Absorbers (WABA) were introduced.

For Cycle 10, the Westinghouse Optimized Fuel Assembly (OFA) (Figure 3.1-6) was introduced. This consisted of Zircaloy-4 clad fuel rods, two Inconel grids (top & bottom), seven Zircaloy-4 grids and Zircaloy-4 instrumentation and guide thimbles. In addition, thimble plugs were removed from the core this cycle based on analysis performed to support removal. The assembly top nozzle design was changed to a Reconstitutable Top Nozzle (RTN) design to facilitate reconstitution of failed fuel.

For Cycle 11, the OFA fuel assembly design incorporated Debris Filter Bottom Nozzles (DFBN) and Integral Fuel Burnable Absorbers (IFBA).

For Cycle 13, the Westinghouse VANTAGE+ fuel design was introduced (see Figures 3.1-3 and 3.1-7). This design included ZIRLO clad fuel rods, two Inconel grids, seven low pressure drop (LPD) Zircaloy-4 grids, three Zircaloy-4 Integral Flow Mixing grids (IFM), ZIRLO instrumentation and guide thimbles, annular axial blankets along with the DFBN, IFBA and RTN. Use of WABAs was continued.

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For Cycle 15, the VANTAGE+ fuel assembly design incorporated Performance+ features of ZIRLO grids and IFMs and a hardened coating of zirconium oxide on the bottom section of the fuel rod clad to increase debris resistance.

For Cycle 16, the VANTAGE+ fuel assembly design was further enhanced with Performance+ features that include debris mitigation features of an additional grid located at the bottom end plug of the fuel rod, a longer fuel rod end plug and a revised DFBN. Other Performance+ enhancements include longer fuel rods and longer annular axial blanket (see Figure 3.1-15).

In addition to the above fuel design changes, the maximum rod average burnup was increased from 60,000 MWD/MTU to 62,000 MWD/MTU starting in Cycle 16.

For Cycle 17, the 15x15 Upgraded fuel assembly design was used (Figure 3.1-16). This design has features to address grid-to-rod fretting fuel failures. These include I-spring mid-grids, enhanced IFMs and balanced mixing vanes. In addition, the tube-in-tube thimble design was incorporated with a single-dashpot, which improves straightness.

For Cycle 18, solid axial blanket pellets were introduced for the non-IFBA fuel rods.

For Cycle 19, the top nozzle spring design was changed from the VANTAGE+ design to the standard spring design.

Cycle 20 and 21 fuel was the same as Cycle 19, there were no fuel design changes.

Cycle 22 uses the 15x15 Upgraded design with changes to the bottom nozzle and the protective grid. Five flow holes on each side of the bottom nozzle were removed to eliminate possible debris intrusion into the fuel through the holes. It is now the modified Debris Filter Bottom Nozzle (mDFBN). The manufacturing of the protective grid was changed to prevent dimple cracking. It is now the Robust Protective Grid (RPG). In addition, secondary sources were removed from the core.

Cycle 23 introduced the Westinghouse Integral Nozzle (WIN) top nozzle design replacing the previous RTN design that is susceptible to stress-corrosion cracking of the hold-down screws.

### Bottom Nozzle

Two types of nozzle designs were used for the HIPAR fuel assemblies. One type, which is square in cross section, is fabricated from type 304 stainless steel consisting of four side plates, 12 cross bars and four pads or feet. The side plates are welded together at the corners to form a plenum for inlet coolant to the fuel assembly. The cross bars are welded at each end to the top edges of the side plate and function as the bottom end support for the fuel rods. The bottom support surface for the fuel assembly is formed by the four pads, which are welded to the side plates in the corners. This design was used in a majority of the first core fuel assemblies. The previously used LOPAR and OFA fuel incorporate an equivalent bottom nozzle design utilizing a square perforated plate rather than the cross bars and side plate. On both designs, their respective cross bars and perforated plate prevent the fuel rods from falling through the bottom nozzles of the assembly.

Coolant flow to the fuel assembly is directed from the plenum in the bottom nozzle upward to the interior of the fuel assembly and to the channel between assemblies.

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Axial loads imposed on the assembly, as well as the weight of the assembly are distributed through the guide thimbles and the bottom nozzle to the lower core support plate. Indexing and positioning of the fuel assembly in the core is controlled through two holes in diagonally opposite pads, which mate with locating pins in the lower core plate. Lateral loads imposed on the fuel assembly are also transferred to the core support structures through the locating pins.

The OFA and VANTAGE+ bottom nozzle used the reconstitutable feature found on the previously installed LOPAR fuel design, which uses a locking cup to lock the thimble screws on the guide thimble assembly, instead of the lockwire used in earlier LOPAR designs. The OFA nozzle assembly is shorter when compared to the previously installed LOPAR assembly to enhance fuel rod growth allowances.

The two bottom nozzle designs used in the OFAs are both square in cross section and fabricated from 304 stainless steel. The design used in earlier regions consists of a perforated plate, four angle legs, and four pads of feet. The angle legs are fastened to the plate forming a plenum space for the coolant inlet to the fuel assembly.

The remaining OFA regions and the VANTAGE+ and 15x15 Upgraded fuel regions (starting with Cycle 13, Region 15) incorporate an equivalent bottom nozzle design denoted as the Debris Filter Bottom Nozzle (DFBN). This nozzle adds side plates or "skirts" to the previous design increasing structural capability for abnormal loads and providing a more defined plenum space below the nozzle. Additionally, the relatively large adapter plate flow holes of the earlier design are replaced with a new pattern of smaller flow holes. The decrease in size of the holes provides a "screen" for larger debris particles, which would otherwise cause damage if allowed to pass into the assembly.

In both designs, the adaptor plates prevent accidental downward ejection of the fuel rods from the fuel assembly. The nozzles are fastened to the assembly guide tubes by stainless steel screws, which penetrate through the nozzle and mate with a threaded plug in each guide tube (Figure 3.1-8). The screw possesses a circular locking cup around the screw head, which is crimped into mating detents (lobes) on the bottom nozzle, preventing the screw from loosening.

The DFBN was modified starting with Cycle 22 to eliminate five holes on each side (mDFBN) to eliminate the potential for intrusion of debris.

### Top Nozzle

The Reconstitutable Top Nozzle (RTN) used in OFA, VANTAGE+ and 15x15 Upgraded fuel assemblies is a box-like structure, which functions as the fuel assembly upper structural element and forms a plenum space where the heated fuel assembly discharge coolant is mixed and directed toward the flow holes in the upper core plate. The nozzle is comprised of an adaptor plate enclosure, top plate, clamps, hold-down leaf springs and assorted hardware. Each nozzle has four sets of leaf springs. All parts, with the exception of the springs and their hold-down bolts/screws, are constructed of type 304 stainless steel. The springs are made from age hardenable Inconel 718 and the bolts/screws from Inconel 600 for Region 16 and earlier regions, and from shot peened Inconel 718 for Regions 17 through 24. The bolts/screws were eliminated with the WIN design.

The adaptor plate portion of the nozzle is square in cross section, and is perforated by machined slots to provide for coolant flow through the plate. At assembly, the top ends of the LOPAR thimble stainless sleeves are fitted through individual bored holes in the plate and welded to the plate

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around the circumference of each hole. In the OFA removable top nozzle design, a groove is provided in each thimble thru-hole in the nozzle plate into which a stainless steel nozzle insert is mechanically connected by means of a preformed circumferential bulge near the top of the insert. Thus, the adaptor plate acts as the fuel assembly top end plate, and provides a means of distributing evenly among the guide thimbles any axial loads imposed on the fuel assemblies.

The nozzle enclosure is actually a square tubular structure, which forms the plenum section of the top nozzle. The bottom end of the enclosure is pinned and welded to the periphery of the adaptor plate and the top end is welded to the periphery of the top plate.

The top plate is square in cross section with a square central hole. The hole allows clearance for the rod cluster control absorber rods to pass through the nozzle into the guide thimbles in the fuel assembly and for coolant exit from the fuel assembly to the upper internals area. Two pads containing axial through-holes, which are located on diametrically opposite corners of the top plate provide a means of positioning and aligning the top of the fuel assembly. As with the bottom nozzle, alignment pins in the upper core plate mate with the holes in the top nozzle plate. Hold-down forces of sufficient magnitude to oppose the hydraulic lifting forces on the fuel assembly are obtained by means of the leaf spring sets, which are mounted on the top plate. The springs are fastened in pairs to the top plate at the two corners where alignment holes are not used and radiate out from the corners parallel to the sides of the plate. Prior to the WIN design, fastening of each pair of springs is accomplished with a clamp, which fits over the ends of the springs and two bolts/screws (one per spring set), which pass through the clamp and spring, and thread into the top plate. At assembly, the spring mounting bolts/screws are torqued sufficiently to preload against the maximum spring load and then lockwelded to the clamp, which is counterbored to receive the bolt/screw head. The spring load is obtained through deflection of the spring pack by the upper core plate. The spring pack form is such that it projects above the fuel assembly and is depressed by the core plate when the internals are loaded into the reactor. The free end of the spring pack is bent downward and captured in a key slot in the top plate to guard against loose parts in the reactor in the event (however remote) of spring fracture. The capture of the loose end has been deleted in latter designs.

Starting with Cycle 14, Region 16, the fuel has a cast top nozzle. This is a two-piece design incorporating a machined stainless steel adapter plate welded to a low-cobalt investment casting. The cast top nozzle is functionally interchangeable with the previous design and meets design criteria for the top nozzle.

Westinghouse has developed a top nozzle design to eliminate the potential of generating a loose part from fracture of the fuel assembly holddown spring screws. This design is called the Westinghouse Integral Nozzle (WIN). The WIN uses the same basic two piece nozzle as the standard design except that the top nozzle casting has been modified to include an integral pad in place of the previously separate clamp. As the name implies, these pads are cast as integral parts of the top nozzle casting. The WIN springs includes manufacturing process modifications for added margin against primary water stress corrosion cracking. Unlike previous bolted designs, the WIN design provides a wedged rather than a clamped joint. The tails of the spring packs are a slip-fit interface in the respective clamp pad cavities. Once preloaded, each spring pack is effectively locked in place during normal operation by the reaction forces generated at its tail. The flow holes, thermal characteristics, and method of attachment are all unchanged from the previous top nozzle design. The design was first introduced with the Cycle 23 Region 25 15x15 Upgraded fuel assemblies.



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In addition to its plenum and structural functions, the nozzle provides a protective housing for components, which mate with the fuel assembly. In handling a fuel assembly with a control rod inserted, the control rod spider is contained within the nozzle. During operation in the reactor, the nozzle protects the absorber rods from coolant cross flows in the unsupported span between the fuel assembly adaptor plate and the end of the guide tube in the upper internals package. Plugging devices, which fill the ends of the fuel assembly thimble tubes at unrodded core locations, neutron source rods and burnable absorber rods are all contained within the fuel top nozzle.

For the RTN and WIN designs, a stainless steel nozzle insert is mechanically connected to the top nozzle adaptor plate (Figure 3.1-10) via the engagement of the preformed circumferential bulge near the top of the insert and the mating groove in the wall of the adapter plate thimble tube through-hole. The insert has four equally spaced axial slots, which allow the insert to deflect inwardly at the elevation of the bulge, thus permitting the installation and removal of the nozzle. The insert bulge is positively held in the adapter plate mating groove by placing a lock tube with a uniform OD identical to that of the thimble tube into the insert. The lock tube is secured in place by a top flare, which creates a tight fit and six non-yielding projections on the OD, which interface with the concave side of the insert to preclude escape during core component transfer. The adaptor plate acts as the fuel assembly top end plate and provides a means of evenly distributing any axial loads imposed on the fuel assemblies to the guide thimbles.

### Guide Thimbles

The control rod guide thimbles in the fuel assemblies provide guided channels for the absorber rods during insertion and withdrawal of the control rods. Up to and including Region 18 (VANTAGE+), they are fabricated from a single piece of tubing, which is drawn to two different diameters. The OFA thimbles are made of Zircaloy-4 and the VANTAGE+ thimbles are made of ZIRLO™. The larger inside diameter at the top provides a relatively large annular area for rapid insertion during a reactor trip and accommodates a small amount of upward cooling flow during normal operations. The bottom portion of the guide thimble has two sections of reduced diameter producing a "double dashpot" action when the absorber rods near the end of travel in the guide thimbles during a reactor trip. The transition zones at the dashpot sections are conical in shape so that there are no rapid changes in diameter in the tube.

Starting with Region 19 (15x15 Upgraded fuel design), the guide thimbles incorporate the tube-in-tube dashpot design. The tube-in-tube design utilizes a separate dashpot tube assembly that is inserted into the guide thimble assembly pulled to a press fit over the thimble end plug and bulged into place. To maintain the same diametrical clearance between the guide thimble ID and the dashpot OD, the 15x15 upgraded nominal dashpot thickness was reduced from 0.0165 to 0.0160 inches. As the dashpot tube in the design can provide additional lateral support in that bottom thimble span, it is expected that there will be additional resistance to lateral deformation and Incomplete Rod Insertions as a result of the design modification. The 15x15 Upgraded fuel thimbles are made of ZIRLO™.

Flow holes are provided just above the first dashpot transition to permit the entrance of cooling water during normal operation, and to accommodate the outflow of water from the dashpot during reactor trip.

The dashpot is open at the bottom by means of the drainage hole in the thimble screws that secure the bottom nozzle to the welded end plugs of the guide thimbles. This geometry is shown in Figure 3.1-8.

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The top ends of the thimble tubes are mechanically attached to the sleeve of the top grid. An insert is also bulge attached to the thimble and the insert upper end is in turn mechanically attached to the top nozzle as shown in Figure 3.1-10.

### VANTAGE+ Grids

Prior to Region 18, the VANTAGE+ assembly has twelve grids. Starting with Region 18, a thirteenth grid, the protective grid (P-grid), was added to the VANTAGE+ assembly. The top and bottom grids, as in the OFA assembly, and the P-grid are Inconel 718 non-mixing vane grids. Low Pressure Drop (LPD) Zircaloy-4 grids are used for the middle grids with Zircaloy-4 IFMs located in the three uppermost middle grid spans. The VANTAGE+ fuel assembly with Performance+ options has ZIRLO™ grids for the three IFMs and seven mid grids. The LPD grids have mixing vanes, diagonal springs and a reduced grid height, relative to the OFA grids. The LPD grid cells use the standard four dimples and two springs per cell for support locations. The IFMs provide mid-span flow mixing in the hottest fuel assembly spans. Each IFM cell contains four dimples, which are designed to prevent midspan channel closure and fuel rod contact with the mixing vanes. With the additional Performance+ enhancements added to the fuel starting with Region 18, a new Protective Bottom Grid (PBG) has been added. The PBG is a wider, extra grid at the very bottom of the fuel assembly that protects the fuel from debris. Its purpose is to filter out debris and hold it at an elevation below the bottom of the active core. The PBG is not a structural grid. The bottom of the PBG lies below the tops of the lower end plugs within the fuel rod. This means that any debris caught in the PBG cannot fret through the cladding and expose fuel pellets.

All VANTAGE+ outside grid straps contain mixing vanes, which also act as guides during fuel handling. The grids are also attached to the thimble tubes via the bulging mechanism as shown in Figure 3.1-14. Top grid nozzle attachment is shown in Figure 3.1-10. All grids employ the anti-snag outer strap design. A mixing vane grid is shown in Figure 3.1-11.

### 15x15 Upgraded Design Grids

The 15x15 Upgraded fuel design still contains twelve grids with the top and bottom grids unchanged from the VANTAGE+ design. The thirteenth grid, the protective grid (P-grid) also remains the same as the VANTAGE+ design. The middle grids have changed to an I-spring design. The changes were made to improve fuel rod fretting margin. In addition to the spring change the size of the dimples was increased. The strap thickness was decreased to help offset the pressure drop increase due to the I-spring and increased grid strap height. The strap height increased to create space to accommodate the increased dimples and the I-spring. The IFM grid design was enhanced to increase contact area also.

The 15x15 Upgraded design protective grid has been redesigned for Cycle 22 to reduce stresses that caused dimple cracking. The Robust Protective Grid (RPG) dimensions changed and vibration mitigation features were added.

### Fuel Rods

The fuel rods consist of uranium dioxide ceramic pellets in slightly cold worked ZIRLO™ tubing, which is plugged and seal welded at the ends to encapsulate the fuel. Sufficient void volume and clearances are provided within the rod to accommodate fission gases released from the fuel, differential thermal expansion between the cladding and the fuel, helium released from poison



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burnup (IFBA rods), and fuel swelling due to accumulated fission products without overstressing of the cladding or seal welds. Shifting of the fuel within the cladding is prevented during handling or shipping prior to core loading by a stainless steel helical compression spring, which bears on the top of the fuel.

At assembly, the pellets are stacked in the cladding to the required fuel height. The compression spring is then inserted into the top end of the fuel and the end plugs pressed into the ends of the tube and welded. A hold-down force of approximately four times the weight of the fuel is obtained by compression of the spring between the top end plug and the top of the fuel pellet stack. All fuel rods are internally pressurized with helium in order to minimize compressive clad stresses and creep due to coolant operating pressures.

The fuel pellets are in the form of a right circular cylinder and consist of slightly enriched uranium dioxide powder, which is compacted by cold pressing and sintering to the required density. The ends of each pellet are dished slightly to allow the greater axial expansion at the center of the pellets to be taken up within the pellets themselves and not in the overall fuel length. The 15x15 Upgraded fuel has mid-enriched annular (IFBA) and solid (non-IFBA) pellets in the axial blanket region of the fuel rod and optimized plenum spring to maximize the available plenum volume for increased burnup. The 15x15 Upgraded fuel has a longer fuel rod to allow higher fission gas release due to longer cycles. This is allowable due to the ZIRLO cladding, which has less rod growth on irradiation.

Each fuel rod is marked with a permanent traceability code. This aids in ensuring that rods of the proper enrichment will be loaded into each fuel assembly. The identification numbers on the fuel assembly top nozzles will then maintain the enrichment identity.

### 3.1.3.2 Rod Cluster Control Assemblies

The rod cluster control assemblies remain inserted in fuel assemblies stored in the Spent Fuel Pit (SFP) or Independent Spent Fuel Storage Installation (ISFSI).

The control rods or rod cluster control assemblies consist of a group of individual absorber rods fastened at the top end to a common hub or spider assembly. These assemblies, one of which is shown in Figure 3.1-1, were provided to control the reactivity of the core under operating conditions. The absorber material used in the control rods is silver-indium-cadmium alloy, which is essentially "black" to thermal neutrons and has sufficient additional resonance absorption to increase its worth significantly. The alloy is in the form of extruded single-length rods, which are sealed in stainless steel tubes to prevent the rods from coming in direct contact with the coolant. Additional information regarding the rod cluster control assemblies is provided in Table 3.1-1.

The overall control rod length is such that when the assembly has been withdrawn through its full travel, the tips of the absorber rods remain engaged in the guide thimbles so that alignment between rods and thimbles is always maintained. Since the rods are long and slender, they are relatively free to conform to any small misalignments with the guide thimble. Prototype tests have shown that the rod cluster control assemblies are very easily inserted and not subject to binding even under conditions of severe misalignment.

The spider assembly is in the form of a center hub with radial vanes supporting cylindrical fingers from which the absorber rods are suspended. Handling detents and detents for connection to the drive shaft are machined into the upper end of the hub. A spring pack is assembled into a skirt integral to the bottom of the hub to stop the rod cluster control assembly and absorb the energy

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from the impact at the end of a trip insertion. The radial vanes are joined to the hub and the fingers are joined to the vanes by furnace brazing. A center-post, which holds the spring pack and its retainer is threaded into the hub within the skirt and welded to prevent loosening in service. All components of the spider assembly are made from type 304 stainless steel except for the springs, which are Inconel X-750 alloy and the retainer, which is of 17-4 PH material.

The absorber rods are secured to the spider so as to ensure trouble free service. The rods are first threaded into the spider fingers and then pinned to maintain joint tightness, after which the pins are welded in place. The end plug below the pin position is designed with a reduced section to permit flexing of the rods to correct for small operating or assembly misalignments.

In construction, the silver-indium-cadmium rods are inserted into cold-worked stainless steel tubing, which is then sealed at the bottom and the top by welded end plugs. Sufficient diametral and end clearance are provided to accommodate relative thermal expansions and to limit the internal pressure to acceptable levels.

The bottom plugs are made bullet-nosed to reduce the hydraulic drag during a reactor trip and to guide smoothly into the dashpot section of the fuel assembly guide thimbles. The upper plug is threaded for assembly to the spider and has a reduced end section to make the joint more flexible. Stainless steel clad silver-indium-cadmium alloy absorber rods are resistant to radiation and thermal damage thereby ensuring their effectiveness under all operating conditions. Rods of similar design have been successfully used in a number of operating nuclear plants.

### 3.1.3.3 Neutron Source Assemblies

Neutron source assemblies remain inserted in fuel assemblies stored in the SFP or ISFSI.

Six neutron source assemblies were utilized in the first cycle core. These consisted of two assemblies with four secondary source rods each, and four assemblies with one secondary source rod and one primary source rod each. The rods in each assembly were fastened to a spider at the top end. The spider for the four secondary source rod assemblies was similar to the rod cluster control assembly spiders, while the primary source assembly spider was similar to that of the burnable poison and plugging device assemblies. Various source assembly designs are used in the reload cycles.

In the first cycle core, the neutron source assemblies were inserted into the rod cluster control guide thimbles in fuel assemblies at unrodded locations. The location and orientation of each of the assemblies in the core is shown in [Historical Figure 3.1-17].

The primary and secondary source rods both utilize the same type of cladding material as the absorber rods (cold-worked type 304 stainless steel tubing). The secondary source rods contain Sb-Be pellets. The primary source rods contained capsules of Pu-Be source material in the initial core loading; for reload cores, this material may vary. Design criteria similar to that for the fuel rods is used for the design of the source rods; i.e., the cladding is free standing, internal pressures are always less than reactor operating pressure, and internal gaps and clearances are provided to allow for differential expansions between the source material and cladding.

Starting in Cycle 22, secondary sources were removed from the core.

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### 3.1.3.4 Plugging Devices

Plugging devices remain inserted in fuel assemblies stored in the SFP or ISFSI.

In order to limit bypass flow through the rod cluster control guide thimbles in fuel assemblies, which do not contain either control rods, source assemblies or burnable absorber rods, the fuel assemblies at those locations were fitted with plugging devices. The plugging devices consist of a flat plate with short rods suspended from the bottom surface and a spring pack assembly. The plugging devices fit within the fuel assembly top nozzles and rest on the adaptor plate. The short rods project into the upper ends of the thimble tubes to reduce the bypass flow area. The spring pack is compressed by the upper core plate when the upper internals package is lowered into place. Similar short rods are also used on the source assemblies to fill the ends of all vacant fuel assembly guide thimbles. All components in the plugging device, except for the springs, are constructed from type 304 stainless steel. The springs are wound from an age hardenable nickel base alloy to obtain higher strength.

Coincident with implementation of the Indian Point Unit 2 OFA transition, removal of thimble plugging devices from the core was allowed. This included the removal of the thimble plugs from the OFA assemblies, previously installed LOPAR assemblies, and all new core component clusters (burnable absorbers and sources).

Thimble plugs were reinstalled for all assemblies without a designated insert (e.g. RCCA, WABA, or secondary source) in Cycles 17, 18, 19 and 20 to satisfy stretch power uprate conditions. Starting in Cycle 21, thimble plugs were again removed from the core.

### 3.1.3.5 Burnable Absorber Rods

Burnable absorber rods remain inserted in fuel assemblies stored in the SFP or ISFSI.

The burnable absorber rods are statically suspended and positioned in vacant rod cluster control thimble tubes within the fuel assemblies at nonrodded core locations. The absorber rods in each fuel assembly are grouped and attached together at the top end of the rods by a flat plate, which fits with the fuel assembly top nozzle and rests on the top adaptor plate.

The plate with the absorber rods is held down and restrained against vertical motion with a spring pack, which is attached to the plate and is compressed by the upper core plate when the reactor upper internals package is lowered into the reactor. Historically, this ensured that the absorber rods cannot be lifted out of the core by flow forces.

The absorber rods used during Cycles 1 through 7 consisted of borated Pyrex glass tubes contained within type 304 stainless steel tubular cladding, which was plugged and seal welded at the ends to encapsulate the glass (Figures 3.1-18 and 3.1-19). The glass was also supported along the length of its inside diameter by a thin-wall type 304 stainless steel tubular inner liner.

Starting in Cycle 8, Wet Annular Burnable Absorber (WABA) rods were used. As shown in Figures 3.1-20 and 3.1-21, WABA rods are composed of annular pellets containing aluminum oxide-boron carbide ( $\text{Al}_2\text{O}_3$  -  $\text{B}_4\text{C}$ ) burnable absorber material contained within two concentric Zircaloy-4 tubes. The Zircaloy-4 tubes are plugged and seal welded at the ends to enclose the annular stack of absorber material. The tubes are also the inner and outer cladding of the annular burnable absorber rod. A hold-down device is placed on top of the pellet stack to hold the stack in position and to allow for pellet stack growth. The hold-down device is a C-shape Zircaloy polygonal ring

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clip. Within the rod is an annular plenum to allow for helium gas release from the absorber material during boron depletion. During operation, reactor coolant flowed through the inner tube and outside the outer tube of the annular rod. The annular rods are grouped and attached at the top end to a hold-down assembly and retaining plate in the same way as the borosilicate glass absorber rod. WABA rods are used in preference to standard BPRAs to provide smaller residual burnup penalty.

Starting with Cycle 11, Integral Fuel Burnable Absorbers (IFBA) were used in conjunction with the WABA rods. The IFBA features a zirconium diboride coating on the fuel pellet surface on the central portion of the enriched  $\text{UO}_2$  pellets. IFBA provided power peaking and moderator temperature coefficient control.

Additional information regarding the burnable absorber rods is provided in Table 3.1-1.

### 3.1.4 Evaluation

#### 3.1.4.1 Fuel Evaluation

To assure that manufactured fuel rods meet a high standard of excellence from the standpoint of functional requirements, many inspections and tests are performed both on the raw material and the finished product. These tests and inspections include chemical analysis, tensile testing of fuel tubes, dimensional inspection, X-ray of both end plug welds, ultrasonic testing, and helium leak tests.

#### 3.1.4.2 Fuel Assembly and Rod Cluster Control Assembly Mechanical Evaluation

Axial and lateral bending tests have been performed in order to simulate mechanical loading of the assembly. Although the maximum column load expected to be experienced in service is approximately 1000 lb., the fuel assembly was successfully loaded to 2200 lb. axially with no damage resulting. This information is also used in the design of fuel handling equipment to establish the limits for inadvertent axial loads during fuel handling.

### 3.1.5 Quality Assurance Program

The quality assurance program plan of the Westinghouse Nuclear Fuel Division is summarized in Reference 3.1-1.

The program provides for control over all activities affecting product quality, commencing with design and development and continuing through procurement, materials handling, fabrication, testing and inspection, storage, and transportation. The program also provides for the indoctrination and training of personnel and for the auditing of activities affecting product quality through a formal auditing program.

Westinghouse drawings and product, process, and materials specifications identify the inspections to be performed.

### 3.1.6 Quality Control

Quality control philosophy is generally based on the following inspections being performed to a 95-percent confidence that at least 95-percent of the product meets specification, unless otherwise noted.

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### 1. Fuel system components and parts

The characteristics inspected depend upon the component parts; the quality control program includes dimensional and visual examinations, check audits of test reports, material certification, and nondestructive examination, such as X-ray and ultrasonic.

All material used in the fuel assembly is accepted and released by quality control.

### 2. Pellets

Inspection is performed for dimensional characteristics such as diameter, density, length, and squareness of ends. Additional visual inspections are performed for cracks, chips, and surface conditions according to approved standards.

Density is determined in terms of weight per unit length and is plotted on zone charts used in controlling the process. Chemical analyses are performed on a specified sample basis throughout pellet production.

### 3. Rod inspection

The fuel rod inspection consists of the following nondestructive examination techniques and methods, as applicable:

- a. Each rod is leak tested using a calibrated mass spectrometer, with helium being the detectable gas.
- b. Rod welds are inspected by ultrasonic test or X-ray in accordance with a qualified technique and Westinghouse specification.
- c. All rods are dimensionally inspected prior to final release. The requirements include such items as length, camber, and visual appearance.
- d. All fuel rods are inspected by gamma scanning or other approved methods to ensure proper plenum dimensions.
- e. All fuel rods are inspected by gamma scanning, or other approved methods to ensure that no significant gaps exist between pellets.
- f. All fuel rods are active gamma scanned to verify enrichment control prior to acceptance for assembly loading.
- g. Traceability of rods and associated rod components is established by quality control.

### 4. Assemblies

Each fuel assembly is inspected for compliance with drawing and/or specification requirements.

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### 5. Other inspections

The following inspections are performed as part of the routine inspection operation:

- a. Tool and gauge inspection and control, including standardization to primary and/or secondary working standards. Tool inspection is performed at prescribed intervals on all serialized tools. Complete records are kept of calibration and conditions of tools.
- b. Audits are performed of inspection activities and records to ensure that prescribed methods are followed and that records are correct and properly maintained.
- c. Surveillance inspection, where appropriate, and audits of outside contractors are performed to ensure conformance with specified requirements.

### 6. Process control

To prevent the possibility of mixing enrichments during fuel manufacture and assembly, strict enrichment segregation and other process controls are exercised.

The uranium dioxide powder is kept in sealed containers. The contents are fully identified both by descriptive tagging and preselected color coding. A Westinghouse identification tag completely describing the contents is affixed to the containers before transfer to powder storage. Isotopic content is confirmed by analysis.

Powder withdrawal from storage can be made by only one authorized group, which directs the powder to the correct pellet production line. All pellet production lines are physically separated from each other and pellets of only a single nominal enrichment and density are produced in a given production line at any given time.

Finished pellets are placed on trays identified with the same color code as the powder containers and transferred to segregated storage racks within the confines of the pelleting area. Samples from each pellet lot are tested for isotopic content and impurity levels prior to acceptance by quality control. Physical barriers prevent mixing of pellets of different nominal densities and enrichments in this storage area. Unused powder and substandard pellets are returned to storage in the original color-coded containers.

Loading of pellets into the clad is performed in isolated production lines, and again only one enrichment and density loaded on a line at a time.

A serialized traceability code is placed on each fuel tube to provide unique identification. The end plugs are inserted and then inert-welded to seal the tube. The fuel tube remains coded and traceability identified until just prior to installation in the fuel assembly.

At the time of installation into an assembly, the traceability codes are removed and a matrix is generated to identify each rod in its position within a given assembly. The top nozzle is inscribed with a permanent identification number providing traceability to the fuel contained in the assembly.

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### REFERENCES FOR SECTION 3.1

1. J. Moore, "Nuclear Fuel Division Quality Assurance Program Plan;" WCAP-7800, Revision 5, Westinghouse Electric Corporation, November 1979.



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**TABLE 3.1-1**  
**Core Mechanical Design Parameters<sub>1</sub>**

<u>Fuel assemblies</u>	
Rod array	15 x 15
Rods per assembly	204 <sub>2</sub>
Rod pitch, in.	0.563
Overall dimensions	8.426 x 8.426 HIPAR/LOPAR 8.424 x 8.424 OFA 8.426 x 8.426 VANTAGE+/15x15 Upgraded
Number of grids per assembly (HIPAR/LOPAR/OFA)	9
(VANTAGE+)	12 or 13
(15x15 Upgraded)	13
Number of instrumentation thimbles	1
Number of guide thimbles	20
Diameter of guide thimbles, upper part, in., HIPAR	0.545 O.D. x 0.515 I.D.
Diameter of guide thimbles, lower part, in., HIPAR	0.484 O.D. x 0.454 I.D.
Diameter of guide thimbles, upper part, in., LOPAR	0.546 O.D. x 0.512 I.D.
Diameter of guide thimbles, lower part, in., LOPAR	0.489 O.D. x 0.455 I.D.
Diameter of guide thimbles, upper part, in., OFA/V+	0.533 O.D. x 0.499 I.D.
Diameter of guide thimbles, lower part, in., OFA/V+	0.489 O.D. x 0.455 I.D.
Diameter of guide thimbles, upper part, in., 15x15 Upgraded	0.533 O.D. x 0.499 I.D.
Diameter of guide thimbles, lower part, in., 15x15 Upgraded	0.487 O.D. x 0.455 I.D.
<u>Fuel rods</u>	
Outside diameter, in.	0.422
Diametral gap, in.	0.0075
Clad thickness, in.	0.0243
Clad material	Zircaloy-4 (HIPAR/LOPAR/OFA) ZIRLO™ (VANTAGE+/15x15 Upgraded)
Overall length	148.6, HIPAR 151.9, LOPAR 152.17, OFA 152.55, VANTAGE+ 152.88, V+ w/P+ Enhancements/15x15 Upgraded
Length of end cap, overall, in.	0.688, HIPAR 0.265, LOPAR 0.357, OFA/V+ (TOP) 0.430, OFA/V+ (BOTTOM) 0.350, V+w/P+/15x15 Upgraded (TOP) 0.810, V+w/P+/15x15 Upgraded (BOTTOM)
Length of end cap, inserted in rod	0.250, HIPAR 0.200, LOPAR 0.130, OFA/V+/15x15 Upgraded
Active fuel length, in.	142, HIPAR 144, LOPAR 144, OFA/V+/15x15 Upgraded



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**TABLE 3.1-1 (Cont.)**  
**Core Mechanical Design Parameters<sub>1</sub>**

Rod cluster control assemblies

Neutron absorber	5-percent Cd, 15-percent In, 80-percent Ag
Cladding material	Type 304 SS - cold worked
Clad thickness, in.	0.019
Number of control rods per cluster	20
Length of rod control, in.	156.436 (overall)
Length of absorber section, in.	142.00

Wet Annular Burnable Absorber (WABA) Rods

Pellet Material	Al <sub>2</sub> O <sub>3</sub> -B <sub>4</sub> C
Boron Loading(Natural)	0.0243 g/cm
(B-10)	0.0060 g/cm
Pellet O.D. /I.D.	0.318"/0.278"
Tube material	Zircaloy-4
Outer tube O.D. /I.D.	0.3810"/0.3290"
Inner tube O.D. /I.D.	0.2670"/0.2250"

Notes:

1. All dimensions are for cold conditions. Data is for all fuel types unless otherwise stated.
2. Twenty-one rods are omitted: Twenty provide passage for control rods and one contains incore instrumentation.

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### 3.1 FIGURES

Figure No.	Title
Figure 3.1-1	Typical Rod Cluster Control Assembly
Figure 3.1-2	Rod Control Cluster Assembly Outline
Figure 3.1-3	Fuel Assembly and Control Cluster Cross Section - HIPAR, LOPAR, OFA, and VANTAGE+
Figure 3.1-4	HIPAR Fuel Assembly
Figure 3.1-5	LOPAR Fuel Assembly
Figure 3.1-6	OFA Fuel Assembly
Figure 3.1-7	VANTAGE+ Fuel Assembly
Figure 3.1-8	Guide Thimble to Bottom Nozzle Joint
Figure 3.1-9	LOPAR Top Grid to Nozzle Attachment
Figure 3.1-10	OFA And VANTAGE+ Top Grid to Nozzle Attachment
Figure 3.1-11	Spring Clip Grid Assembly
Figure 3.1-12	Mid-Grid Expansion Joint Design Plan View
Figure 3.1-13	Elevation View - LOPAR Grid to Thimble Attachment
Figure 3.1-14	Elevation View-VANTAGE+ Grid to Thimble Attachment
Figure 3.1-15	VANTAGE+ Fuel Assembly with Performance+ Enhancements
Figure 3.1-16	15x15 Upgraded Fuel Assembly
Figure 3.1-17	Cycle 1 - Neutron Source Locations [Historical]
Figure 3.1-18	HIPAR Burnable Poison Rod
Figure 3.1-19	LOPAR Burnable Poison Rod
Figure 3.1-20	Comparison of Borosilicate Glass Absorber Rod with WABA Rod
Figure 3.1-21	Wet Annular Burnable Absorber Rod

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### 3.2 Chemical and Volume Control System

Portions of the chemical and volume control system will continue to be utilized in the permanently defueled condition to process liquid radwaste.

#### 3.2.1 System Design and Operation

The following chemical and volume control system (CVCS) equipment will remain in service in the permanently defueled condition to support the processing of liquid radwaste:

- 21, 22, and 23 CVCS Hold Up Tanks (HUT)
- 21, 22, and 23 CVCS Hold Up Tank Transfer Pumps
- Hold Up Tank Recirculation Pump

In addition, the following equipment is part of the Waste Disposal System.

- Waste Hold Up Tank (WHUT)
- Spent Resin Storage Tank
- Waste Hold Up Transfer Pump

Table 3.2-1 defines the ASME code class for the CVCS. A recirculation pump is provided to transfer liquid from one holdup tank to another.

Liquid effluent in the holdup tanks is processed by demineralization or as radwaste.

The resin fill tank is used to process resin from the demineralizers. The tank is made of austenitic stainless steel.

Basic material of construction is stainless steel for all valves.

All chemical and volume control system piping handling radioactive liquid is austenitic stainless steel. All piping joints and connections are welded, except where flanged connections are required to facilitate equipment removal for maintenance and hydrostatic testing.

#### 3.2.2 Purpose

Portions of the system are used to collect effluents and transfer them to the waste disposal system. Effluents are initially collected in the chemical and volume control system holdup tanks.

A holdup tank low pressure interlock will trip the CVCS holdup tank transfer pumps upon low pressure in the holdup tank. This interlock reduces the potential for creating a negative pressure condition in the holdup tanks during drain down of the tank.

Three holdup tanks contain radioactive liquid. The contents of one tank are normally being processed while another tank is being filled. The third tank is normally kept empty to provide additional storage capacity when needed. The total liquid storage sizing basis for the holdup tanks is given in Table 3.2-2. The tanks are constructed of austenitic stainless steel. The three holdup tank transfer pumps are used to transfer water to waste collection tanks in unit 1. These centrifugal pumps are constructed of austenitic stainless steel.

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**TABLE 3.2-1**  
**Chemical and Volume Control System Code Requirements**

<u>Component</u>	<u>Code</u>
Holdup tanks	ASME III, Class C
Piping and valves	USAS B31.1 <sub>2</sub>

Notes:

1. ASME III – American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

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TABLE 3.2-2  
Chemical and Volume Control System Principal Component Design Data Summary

	<u>Quantity</u>	<u>Type</u>	<u>Capacity,</u> <u>gpm</u>	<u>Head, ft</u> <u>or psi</u>	<u>Design Pressure,</u> <u>psig</u>	<u>Design</u> <u>Temperature, °F</u>
<u>Pumps</u>						
Holdup tank recirculation	1	Centrifugal	500	100-ft	75	200
Primary water makeup	2	Centrifugal	150	210-ft	150	Ambient
Holdup Tank Transfer Pump 22	1	Centrifugal	25	63-ft	150	200
Holdup Tank Transfer Pump 21 & 23	2	Centrifugal	25	63-ft	150	200

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### 3.3 Auxiliary Coolant System

#### 3.3.1 Design Basis

The auxiliary coolant system consists of two loops as shown in Drawings 227781, 9321-2720, and 251783 [Formerly Figure 3.3-1, Sheets 1, 2, and 3] the component cooling loop and the spent fuel pit cooling loop.

##### 3.3.1.1 Performance Objectives

###### 3.3.1.1.1 Component Cooling Loop

The component cooling loop is designed to provide cooling to dissipate waste heat from various facility components. It also provides cooling for spent fuel pit components.

The loop design provides for detection of radioactivity entering the loop from the spent fuel pit and also provides means for isolation.

###### 3.3.1.1.2 Spent Fuel Pit Cooling Loop

The spent fuel pit cooling loop is designed to remove from the spent fuel pit the heat generated by stored spent fuel elements.

The loop design consists of two pumps, a heat exchanger, a filter, a demineralizer, piping, and associated valves and instrumentation. Alternate cooling capability can be made available under anticipated malfunctions or failures (expected fault conditions).

Loop piping is so arranged that the failure of any pipeline does not drain the spent fuel pit below the top of the stored fuel elements.

The thermal design basis for the loop provides for all fuel pool rack locations to be filled at the end of a full core discharge.

##### 3.3.1.2 Design Characteristics

###### 3.3.1.2.1 Component Cooling Loop

Normally one pump and at least one component heat exchangers are operated to provide cooling water for the components located in the auxiliary building. At elevated CCW supply temperatures two pumps may be required. The water is normally supplied to the SFP cooling system even though one of the components may be out of service.

Makeup water is taken from the primary water treatment plant, as required, and delivered to the surge tank. A backup source of water is provided from the primary water makeup transfer pumps.

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The operation of the loop is monitored with the following instrumentation:

1. A pressure indicator on the line between the component cooling pumps and the component cooling heat exchangers.
2. A temperature indicator, flow indicator, and radiation monitor in the outlet line from the heat exchangers.
3. A temperature indicator on the main inlet line to the component cooling pumps.

### 3.3.1.2.2 Spent Fuel Pit Cooling Loop

The spent fuel pit contains spent fuel discharged from the Unit 2 and Unit 3 reactors. Spent fuel cooling loop performance has been analyzed for operation at a core power level of 102% of 3216 MWt and at service water temperatures up to 95°F. When a refueling load of approximately 88 freshly discharged assemblies (plus previously discharged assemblies) are present, the pump and spent fuel heat exchanger will handle the load and maintain a bulk pit water temperature less than 140°F. When a full core of 193 assemblies is freshly discharged, the bulk pit water temperature is maintained below 180°F.

Two criteria must be met before spent fuel can be discharged to the spent fuel pit:

1. Spent fuel cannot be discharged to the spent fuel pit until at least 84 hours after shutdown to satisfy the assumptions of the spent fuel handling accident analysis as discussed in Section 6.2.1. This requirement will be met prior to the implementation of the original version of the Defueled Technical Specifications and Defueled Safety Analysis Report. Thus, it will essentially be a historical requirement, because the facility will be permanently shut down and defueled.
2. An additional delay time limit prior to spent fuel discharge is administratively controlled by operating procedures to ensure that the total spent fuel heat load is within the capacity of the spent fuel cooling loop to satisfy the bulk pit water temperature limits discussed above. This is a variable time limit primarily dependent upon service water temperature, and cooling capacity without supplemental cooling.

### 3.3.1.3 Codes and Classifications

All piping and components of the auxiliary coolant system are designed to the applicable codes and standards listed in Table 3.3-1. The component cooling loop water contains a corrosion inhibitor to protect the carbon steel piping. Austenitic stainless steel piping is used in the remaining piping systems that contain borated water without a corrosion inhibitor.

### 3.3.2 System Design and Operation

#### 3.3.2.1 Component Cooling Loop

Component cooling is provided for Spent Fuel Pit heat exchanger (auxiliary coolant system).

Typically, one component cooling pump and at least one component cooling heat exchanger can accommodate the heat removal loads. Two CCW pumps are in stand-by and at least one heat exchanger is utilized. At elevated CCW supply temperatures two CCW pumps may be required. Three pumps and two heat exchangers can be used to remove the residual and sensible heat. The surge tank accommodates expansion, contraction and inleakage of water, and ensures a

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continuous component cooling water supply until a leaking cooling line can be isolated. Makeup to the surge tank is provided from the primary water makeup system. The surge tank is normally vented to the atmosphere. In the unlikely event that the radiation level in the component cooling loop reaches a preset level above the normal background, the radiation monitor in the component cooling loop annunciates in the control room and closes a valve in the surge tank vent line. Parameters for components in the component cooling loop are presented in Table 3.3-2.

### 3.3.2.2 Spent Fuel Pit Cooling Loop

The spent fuel pit cooling loop removes residual heat from fuel placed in the pit for long term storage. The loop can safely accommodate the heat load from all of the assemblies for which there is storage space available.

The spent fuel pit cooling loop consists of two pumps, a heat exchanger, filter, demineralizer, piping and associated valves and instrumentation. One of the pumps draws water from the pit, circulates it through the heat exchanger and returns it to the pit. Component cooling water cools the heat exchanger. Redundancy of this equipment is not required because of the large heat capacity of the pit and the slow heatup rate.

The clarity and purity of the spent fuel pit water is maintained by passing approximately 5-percent of the loop flow through a filter and demineralizer. The spent fuel pit pump suction line, which is used to draw water from the pit, penetrates the spent fuel pit wall above the fuel assemblies. The penetration location prevents loss of water as a result of a possible suction line rupture.

Parameters for components in the spent fuel cooling loop are presented in Table 3.3-3.

### 3.3.2.3 Component Cooling Loop Components

#### 3.3.2.3.1 Component Cooling Heat Exchangers

The two component cooling heat exchangers are of the shell and straight tube type. Service water circulates through the tubes while component cooling water circulates through the shell side. Parameters are presented in Table 3.3-2.

#### 3.3.2.3.2 Component Cooling Pumps

The three component cooling pumps, which circulate component cooling water through the component cooling loop are horizontal, centrifugal units. The original pumps have casings made from cast iron (ASTM 48) based on the corrosion-erosion resistance and the ability to obtain sound castings. The material thickness indicates the high quality casting practice and the ability to withstand mechanical damage and, as such, is substantially overdesigned from a stress level standpoint. Carbon steel casing material (ASTM A216) has been evaluated and approved for replacement pumps. Parameters are presented in Table 3.3-2.

#### 3.3.2.3.3 Component Cooling Surge Tank

The component cooling surge tank, which accommodates changes in component cooling water volume is constructed of carbon steel. Parameters are presented in Table 3.3-2. In addition to piping connections, the tank has a flanged opening at the top for the addition of the chemical corrosion inhibitor to the component cooling loop.



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### 3.3.2.3.4 Component Cooling Valves

The valves used in the component cooling loop are standard commercial valves constructed of carbon steel with bronze or stainless steel trim. Since the component cooling water is not normally radioactive, special features to prevent leakage to the atmosphere are not provided.

Self-actuated spring-loaded relief valves are provided for lines and components that could be pressurized beyond their design pressure by improper operation or malfunction.

### 3.3.2.3.5 Component Cooling Piping

All component cooling loop piping is carbon steel with welded joints and connections except at components, which might need to be removed for maintenance. The piping has been evaluated for the most limiting component cooling water temperatures under loss of coolant accident conditions and found to be acceptable.

### 3.3.2.3.6 Primary Water Storage Tank

A single 165,000-gal primary water storage tank is provided to store the demineralized water used by the primary water makeup system shown in Drawing 9321-2724 [Formerly Figure 3.3-2]. The storage tank is constructed of type 304 stainless steel.

Chemical addition to the tank, if required, can be accomplished via a 3-in. blind flange connection located near the top of the tank, directly off the pressure-vacuum relief valve. A local sample point is provided on the bottom of the tank in addition to a tank drain and a loop seal overflow. This loop seal will prevent the entrance of air. To ensure that this loop seal is filled with water a valved line is provided from the tank drain to the loop seal.

Besides these lines into the primary water storage tank, there is a feed from the primary water makeup pump recirculation. Lines carrying heating steam to and from the tank also enter it near its bottom. All of these connections and lines entering the tank are heat traced to prevent them from freezing. A large inspection port is provided on the side of the tank.

#### 3.3.2.3.6.1 Primary Water Storage Tank Level Measurement

Level in the tank is measured and indicated locally and in the central control room. In addition, high level and low level are alarmed in the central control room.

#### 3.3.2.3.6.2 Primary Water Storage Tank Temperature Control

Temperature in the tank is indicated locally. An additional temperature measurement is made at the tank, on the suction line to the makeup pumps.

The temperature element will sense a representative fluid temperature. This temperature measurement is used to control steam flow to the coils located at the bottom of the storage tank. The steam coils will maintain the water in the storage tank at a sufficiently high temperature to prevent freezing of the tank contents. The walls of the tank are insulated and all lines connected to the tank and exposed to the environment are electrically heat traced to prevent freezing.

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In addition, the external instrument cabinet is heated and weatherproofed to help ensure a controlled temperature for the tank level instrumentation. Low temperature alarms alert the operator of any instrument heat trace failure or low temperatures in the instrument enclosure.

### 3.3.2.3.7 Primary Water Makeup Pumps

Two primary water makeup pumps are provided and normally take their suction from the primary water storage tank. The pumps are constructed of type 316 austenitic stainless steel. Each can supply 150 gpm of water at a total dynamic head of 210-ft.

Control of both pumps is provided from the central control room. No local control of the pump is provided.

Each pump is also provided with a discharge pressure gauge. Operation of the pumps without a suction head is prevented.

### 3.3.2.4 Spent Fuel Pit Loop Components

#### 3.3.2.4.1 Spent Fuel Pit Heat Exchanger

The spent fuel pit heat exchanger is of the shell and U-tube type with the tubes welded to the tube sheet. Component cooling water circulates through the shell, and spent fuel pit water circulates through the tubes. The tubes are austenitic stainless steel and the shell is carbon steel.

#### 3.3.2.4.2 Spent Fuel Pit Pumps

One of two spent fuel pit pumps circulates water in the spent fuel pit cooling loop. The second pump is on standby. All wetted surfaces of the pumps are austenitic stainless steel, or equivalent corrosion resistant material. The pumps are operated manually from a local station.

#### 3.3.2.4.3 Spent Fuel Pit Filter

The spent fuel pit filter removes particulate matter larger than 5  $\mu$  from the spent fuel pit water. The filter cartridge is synthetic fiber and the vessel shell is austenitic stainless steel.

#### 3.3.2.4.4 Spent Fuel Pit Strainer

A stainless steel strainer is located at the inlet of the spent fuel pit loop suction line for removal of relatively large particles, which might otherwise clog the spent fuel pit demineralizer.

#### 3.3.2.4.5 Spent Fuel Pit Demineralizer

The demineralizer is sized to pass 5-percent of the loop circulation flow, to provide adequate purification of the fuel pit water for unrestricted access to the working area, and to maintain optical clarity.

#### 3.3.2.4.6 Spent Fuel Pit Valves

Manual stop valves are used to isolate equipment and lines, and manual throttle valves provide flow control. Valves in contact with spent fuel pit water are austenitic stainless steel or equivalent corrosion resistant material.

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### 3.3.2.4.7 Spent Fuel Pit Piping

All piping in contact with spent fuel pit water is austenitic stainless steel. The piping is welded except where flanged connections are used at the pump, heat exchanger, and filter to facilitate maintenance.

### 3.3.3 System Evaluation

System performance has been evaluated for service water temperatures up to 95°F for normal conditions and loss of offsite power.

#### 3.3.3.1 Availability and Reliability

##### 3.3.3.1.1 Component Cooling Loop

The portion of the component cooling loop supplying containment components (reactor coolant pumps, excess letdown heat exchanger, and residual heat removal heat exchangers) is permanently isolated and made non-functional following plant shutdown and final removal of the fuel from the core to the spent fuel pit.

The portion of the component cooling loop outside containment, which includes the spent fuel pit heat exchanger, the component cooling water pumps and heat exchangers, and associated valves, piping, and instrumentation, is maintained functional following permanent plant shutdown and de-fueling for the purposes of providing a cooling method for the fuel in the spent fuel pit. The components of this loop section are capable of being repaired or replaced as necessary. The wetted surfaces of the component cooling loop are fabricated from carbon steel. The component cooling water contains a corrosion inhibitor to protect the carbon steel. Welded joints and connections are used except where flanged closures are employed to facilitate maintenance. This loop section was originally designed to Seismic Class I criteria and maintains the robustness of the design although no credit is taken for this in the permanently defueled condition. The loop components continue to be housed in structures that were also originally designed to meet Seismic Class I requirements. The components are designed to the codes given in Table 3.3-1 and the design pressures given in Table 3.3-2. In addition, the components are not subjected to any high pressures or stresses. Hence, a rupture or failure of the system is very unlikely. Should cooling of the stored spent fuel by the component cooling loop be lost for some reason, other means of covering and cooling the fuel are available.

The Component Cooling Water Pumps are powered from appropriate electrical sources that are deemed to be reliable. These sources include primary and alternate off-site power supplies as well as available diesel generators.

##### 3.3.3.1.2 Spent Fuel Pit Cooling Loop

This manually controlled loop may be shut down safely for time periods, as shown in Section 3.3.3.2.2, for maintenance or replacement of malfunctioning components.

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### 3.3.3.2 Leakage Provisions

#### 3.3.3.2.1 Component Cooling Loop

With respect to water leakage from piping, valves, and equipment serving auxiliary coolant systems, welded construction is used where possible to minimize the possibility of leakage. The component cooling water could become contaminated with radioactive water due to a leak in any heat exchanger tube in the auxiliary coolant systems.

Tube or coil leaks in components being cooled would be detected during normal facility operations by the leak detection system described in Section 3.12. Such leaks are also detected at any time by a radiation monitor that samples the component cooling pump discharge downstream of the component cooling heat exchangers.

Leakage from the component cooling loop can be detected by a falling level in the component cooling surge tank. The rate of water level fall and the area of the water surface in the tank permit determination of the leakage rate. To assure accurate determinations, the site personnel would check that temperatures are stable.

The component, which is leaking can be located by sequential isolation or inspection of equipment in the loop. If the leak is in one of the component cooling water heat exchangers it can be isolated and repaired.

The atmospheric vent on the component cooling surge tank is automatically closed in the event of high radiation level in the component cooling loop. If the inflow completely fills the surge tank, the relief valve on the surge tank lifts. The discharge of this relief valve is routed to the auxiliary building waste holdup tank.

The relief valves on the cooling water lines downstream from the spent fuel pit, exchangers are sized to relieve the volumetric expansion occurring if the exchanger shell side is isolated when cool, and high temperature coolant flows through the tube side. The set pressure equals the design pressure of the shell side of the heat exchangers.

The relief valve on the component cooling surge tank is sized to relieve the maximum flow rate of water. Historically, the over-pressurization incident resulted in a maximum component cooling water pressure of 185 psig from an event that is no longer possible. This pressure is allowed in the component cooling water system in accordance with its design code of B31.1, 1967 edition, par 102.2.4(2), addressing permissible variation and allowable stress value for a limited time.

#### 3.3.3.2.2 Spent Fuel Pit Cooling Loop

A leaking fuel assembly in the spent fuel pit can result in a small quantity of fission products may enter the spent fuel cooling water. A bypass purification loop is provided for removing these fission products and other contaminants from the water.

The probability of inadvertently draining the water from the cooling loop of the spent fuel pit is exceedingly low. The only mode would be from such actions as opening a valve on the cooling line and leaving it open when the pump is operating. In the unlikely event of the cooling loop of the spent fuel pit being drained, the spent fuel storage pit itself cannot be drained and no spent fuel is uncovered since the spent fuel pit cooling connections enter near the top of the pit. With no heat removal the time for the spent fuel pit water to rise from 180°F to 212°F with a full core in

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storage is at least 1.8 hr. Makeup water can be supplied within this time from the primary water storage tank and/or the fire protection system. The maximum required makeup rate for boiloff is 62 gpm (for a full core). Spent fuel pit temperature and level instrumentation would warn the operator of an impending loss of cooling. A local flow indicator is available to support operation of the Spent Fuel Pit Pumps.

### 3.3.3.3 Incident Control

#### 3.3.3.3.1 Component Cooling Loop

In the unlikely event of a pipe severance in the component cooling loop, the leak could either be isolated by valving or the broken line could be repaired, depending on the location in the loop at which the break occurred.

Once the leak is isolated or the break has been repaired, makeup water is supplied from the primary water storage tank by one of the primary water pumps. If the loop drains completely before the leakage is stopped, it can be refilled by a primary makeup water pump in less than 2 hr.

Except for the normally closed makeup line the primary water and city water emergency cooling lines, and equipment vent and drain lines, there are no direct connections between the cooling water and other systems. The primary water make-up has manual valves that are normally closed unless required for their design function or testing. The city water emergency cooling line contains two normally closed isolation valves with an open tell-tale connection between them. The tell-tale prevents the potential contamination of a potable water source with component cooling water corrosion inhibitor chemicals. The equipment vent and drain lines outside the containment have manual valves, which are normally closed unless the equipment is being vented or drained for maintenance or repair operations.

#### 3.3.3.3.2 Spent Fuel Pit Cooling Loop

The most serious failure of this loop is complete loss-of-water in the SFP. To protect against this possibility, the SFP cooling connections enter near the water level so that the SFP cannot be either gravity drained or inadvertently drained. The water in the SFP below the cooling loop connections could be removed by using a portable pump.

Instrumentation is provided that will activate an alarm in the control room if the level in the spent fuel pit is at a preset level deviation above or below normal. Operators normally observe the level in the SFP on a regular basis.

### 3.3.3.4 Malfunction Analysis

A failure analysis of pumps, heat exchangers and valves is presented in Table 3.3-4.

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**TABLE 3.3-1**  
**Auxiliary Coolant System Code Requirements**

<u>Component</u>	<u>Code</u>
Component cooling heat exchangers	ASME VIII
Component cooling surge tank	ASME VIII
Component cooling loop piping and valves	USAS B31.1
Spent fuel pit filter	ASME III, Class C
Spent fuel heat exchanger side ASME VIII, shell side	ASME III, Class C, tube
Spent fuel pit loop piping and valves	USAS B31.1

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**TABLE 3.3-2 (Sheet 1 of 2)**  
**Component Cooling Loop Component Data**

<u>Component Cooling Pumps</u>	<u>Parameters</u>
Quantity	3
Type	Horizontal centrifugal
Rated capacity (each), gpm	3600
Rated head, ft H <sub>2</sub> O	220
Motor horsepower, hp	250
Material (pump casing)	Cast iron or Carbon steel
Design pressure, psig	150
Design temperature, °F	200
<u>Component Cooling Heat Exchangers</u>	
Quantity	2
Type	Shell and straight tube
Design heat transfer, Btu/hr	31.4 x 10 <sup>6</sup>
Shell side (component cooling water)	
Operating inlet temperature, °F	100.1
Operating outlet temperature, °F	88.2
Design flow rate, lb/hr	2.66 x 10 <sup>6</sup>
Design temperature, °F	200
Design pressure, psig	150
Material	Aluminum-bronze
Tube side (service water)	
Operating inlet temperature, °F	75 <sub>1</sub>
Operating outlet temperature, °F	81.9
Design flow rate, lb/hr	4.55 x 10 <sup>6</sup>
Design temperature, °F	200
Design pressure, psig	150
Material	Copper-nickel (90-10)

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**TABLE 3.3-2 (Sheet 2 of 2)**  
**Component Cooling Loop Component Data**

Component Cooling Surge Tank

Quantity	1
Volume, gal	2000
Normal water volume, gal	1000
Design pressure, psig	100
Design temperature, °F	200
Construction material	Carbon steel
Relief valve setpoint, psig	52

Component Cooling Loop Piping and Valves

Design pressure, psig	150
Design temperature, °F	200

Notes:

1. Operation is acceptable up to 95°F.



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**TABLE 3.3-3 (Sheet 1 of 3)**  
**Spent Fuel Cooling Loop Component Data**

Spent fuel pit heat exchanger	
Quantity	1
Type	Shell and U-tube
Design heat transfer, Btu/hrs <sub>1</sub>	7.96 x 10 <sup>6</sup>
Shell side (component cooling water)	
Normal operating inlet temperature, °F <sub>1</sub>	100
Normal operating outlet temperature, °F <sub>1</sub>	105.7
Design flow rate, lb/hr	1.4 x 10 <sup>6</sup>
Design temperature, °F	200
Design pressure, psig	150
Material	Carbon steel
Tube side (spent fuel pit water)	
Normal operating inlet temperature, °F <sub>1</sub>	120
Normal operating outlet temperature, °F <sub>1</sub>	112.8
Design flow rate, lb/hr	1.1 x 10 <sup>6</sup>
Design temperature, °F	200
Design pressure, psig	150
Material	Stainless steel
<u>Spent fuel pit skimmer pump</u>	Retired in place

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**TABLE 3.3-3 (Sheet 2 of 3)**  
**Spent Fuel Cooling Loop Component Data**

Spent fuel pit cooling loop piping and valves

Design pressure, psig	150
Design temperature, °F	200

Spent fuel pit skimmer loop piping and valves

Retired in place

Refueling water purification loop piping and valves

Design pressure, psig	150
Design temperature, °F	200

Spent fuel pit pump

Quantity	2
Type	Horizontal centrifugal
Material	Stainless steel
Rated capacity, gpm	2,300
Rated head, ft H <sub>2</sub> O	125
Motor, hp	100
Design pressure, psig	150
Design temperature, °F	200

Spent fuel pit

Volume ft <sup>3</sup>	37,300
Typical Boron concentration, ppm boron	>2,000 min
Tech Spec Boron concentration, ppm boron	>2,000 min

Spent fuel pit filter

Quantity	1
Internal design pressure of housing, psig	200
Design temperature, °F	250
Rated flow, gpm	100
Maximum differential pressure across filter element at rated flow (clean cartridge), psi	5
Maximum differential pressure across filter element prior to removing, psi	20
Filtration requirement	98-percent retention of particles down to 5 μ

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**TABLE 3.3-3 (Sheet 3 of 3)**  
**Spent Fuel Cooling Loop Component Data**

Spent fuel pit strainer

Quantity	1
Rated flow, gpm	2,300
Maximum differential pressure across the strainer element at rated flow (clean), psi	1
Perforation, in.	~0.2

Spent fuel pit demineralizer

Quantity	1
Type	Flushable
Design pressure, psig	200
Design temperature, °F	250
Flow rate, gpm	100
Resin volume, ft <sup>3</sup>	30

Notes:

1. Original design.

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**TABLE 3.3-4**  
**Failure Analysis of Pumps, Heat Exchangers, and Valves**

<u>Components</u>	<u>Malfunction</u>	<u>Comments and Consequences</u>
1. Component cooling water pumps	Rupture of a pump casing	The casing and shell are designed for 150 psi and 200°F, which exceeds maximum operating conditions. Pump is inspectable and protected against credible missiles. Rupture is not considered credible. However, each unit is isolable.
2. Component cooling water pumps	Pump fails to start	One operating pump supplies sufficient cooling water for SFP cooling.
3. Component cooling water pumps	Manual valve on a pump suction line	This is prevented by pre-startup and operational checks. Further, during normal operation, each pump is checked on a periodic basis, which would show if a valve is closed.
4. Component cooling water valve	Normally open valve	The valve is checked open during periodic operation of the pumps during normal operation.
5. Component cooling heat exchanger	Tube or shell rupture	Rupture is considered improbable because of low operating pressures. Each unit is isolable. Both units may be required to carry total heat load for normal operation at 95°F Service Water.
6. Demineralized water makeup line check valve	Sticks open	The check valve is backed up by the manually-operated valve. Manual valve is normally closed.
7. Component cooling heat exchanger vent or drain valve	Left open	This is prevented by pre-startup and operational checks. On the operating unit such a situation is readily assessed by makeup requirements to system. On the second unit such a situation is ascertained during periodic testing.
8. Component cooling water outlet valve to residual heat exchanger	Fails to open	There is one valve on each outlet line from each heat exchanger. One heat exchanger remains in service and provides adequate heat removal to support safe storage of spent fuel in the spent fuel pit.

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### 3.3 FIGURES

Figure No.	Title
Figure 3.3-1 Sh. 1	Auxiliary Coolant System - Flow Diagram, Sheet 1, Replaced with Drawing 227781
Figure 3.3-1 Sh. 2	Auxiliary Coolant System - Flow Diagram, Sheet 2, Replaced with Drawing 9321-2720
Figure 3.3-1 Sh. 3	Auxiliary Coolant System - Flow Diagram, Sheet 3, Replaced with Drawing 251783
Figure 3.3-2	Primary Water Makeup System - Flow Diagram, Replaced with Drawing 9321-2724

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### 3.4 Sampling System

#### 3.4.1 Design Basis

##### 3.4.1.1 Performance Requirements

This system provides for analysis of liquid samples obtained during normal conditions. Sampling of the following systems is discussed below:

1. Holdup tanks.
2. CVCS holdup tank transfer pumps discharge.
3. Chemical drain pump 21 discharge.

These samples are obtained at the high-radiation sampling system panels and evaluated by the online analysis systems or manual analysis.

Sampling system discharge flows are limited under normal and anticipated fault conditions (malfunctions or failure) to preclude any fission product releases beyond the limits of 10 CFR 20. Shielding has been provided to minimize site personnel exposure to any radiation during the sampling procedures.

##### 3.4.1.2 Design Characteristics

The design characteristics of the high-radiation sampling system include the following:

1. Control of background radiation and site personnel exposure to radiation.
2. Rapid sampling and analysis.
3. Sampling and transfer of undiluted samples.

In addition, the system is capable of the following:

1. The system can be used for routine sampling.
2. Additional sample connections are available for more flexibility in selecting sample points; redundant sample connections allow for further expansion if needed to ensure sample acquisition.

Flow paths are also provided for boron concentration, and isotopic inline analysis.

Sampling of other process coolants, such as tanks in the waste disposal system, is accomplished locally. Leakage and drainage resulting from the sampling operations are collected and drained to tanks located in the waste disposal system.

##### 3.4.1.3 Primary Sampling

Low temperature-low pressure samples are obtained by the primary sampling system from the chemical and volume control and auxiliary coolant systems.

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### 3.4.1.4 Codes and Standards

System code requirements are given in Table 3.4-1. In addition, the system meets the following provisions:

1. Provide capability to obtain and analyze a sample without radiation exposure to any individual exceeding the criteria of GDC 19 (10 CFR Part 50, Appendix A).
2. Provide means of measuring pH, conductivity, chlorides, dissolved hydrogen, dissolved oxygen, inline isotopic analysis, and boron analysis.
3. Provide means of safely obtaining diluted and undiluted samples for laboratory analysis.
4. Safely store the sampled fluid until its disposal is determined.
5. Provide the capability to use the system on a continuous day-to-day basis.
6. Provide the capability to flush the sampled lines.

### 3.4.2 System Design and Operation

#### 3.4.2.1 Primary Sampling System

The primary sampling system consists of the high-radiation sampling system, which is shown in Drawing 9321-2745 (Figure 3.4-1). The high-radiation sampling system provides the representative samples for inline monitoring and laboratory analysis under normal conditions. Analytical results provide guidance in the operation of the auxiliary coolant and chemical and volume control systems. Analyses show both chemical and radiochemical conditions. Typical information obtained includes fission product radioactivity level, hydrogen, oxygen, and fission gas content, corrosion product concentration, and chemical additive concentration.

Local instrumentation is provided to permit manual control of sampling operations and to ensure that the samples are at suitable temperatures and pressures before diverting flow to the sample sink.

#### 3.4.2.1.1 Components

##### 3.4.2.1.1.1 Liquid Sampling Panel

The liquid sampling panel valves and components are arranged in two modules installed in a common panel shield:

1. Module 2 - Demineralizer sampling module (DM).
2. Module 3 - Radwaste sampling module (RW).

Sample tubing and components are mounted behind the shielded panel within a plenum. Any gas leakage is vented to a local prefilter and HEPA filters and finally to existing ventilation ducts containing charcoal filters. A vessel at the bottom of the plenum collects any minor liquid leakage, which is pumped to radwaste. This provides containment of radioactivity during sampling operations.

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As a safety measure, the liquid sampling panel has a hooded splash box to contain any accidental liquid spill or gaseous release during reactor coolant or liquid grab sampling from the modules.

Each system can be purged through the sample lines and panel to ensure representative samples will be obtained. The purge flow can be directed back to chemical drain tank 21 and the associated waste disposal system or to the shielded high-radiation sampling system waste collection tank.

All lines of the liquid sampling panel can be flushed with demineralized water following each sampling operation. Provisions are included for eliminating water from the gas expansion vessel and drying the gas lines of the panel.

After sampling, the shielded casks can be removed to provide samples for backup in-house analyses or stored for subsequent offsite analysis. The viewing window and sampling compartment for alignment of the cart and cask are located in the lower right section of the liquid sampling panel.

The types of samples that can be obtained from the liquid sampling panel during normal conditions are undiluted, depressurized liquid grab samples from the demineralizer, and radwaste modules.

An additional function of the liquid sampling panel during normal conditions is the purging of lines with sample to ensure representative samples will be obtained.

### 3.4.2.1.1.2 Isotopic Analyzer

Isotopic analyses may be performed on the following samples obtained from the liquid sampling panel:

1. Undiluted grab samples from the demineralizer and radwaste modules of the liquid sampling panel for normal sampling.
2. Diluted liquid samples from the radwaste modules of the liquid sampling panel.
3. Undiluted liquid samples from the radwaste modules of the liquid sampling panel for offsite analyses.

### 3.4.2.1.1.3 Boron Analyzer

Backup boron analyses may be performed on undiluted grab samples from the demineralizer and radwaste modules of the liquid sampling panel for normal sampling for analysis in the onsite laboratory.

The primary sampling system provides that the routine sample analyses of undiluted samples are performed using a mannitol titration boron analyzer. It periodically samples an identical line from the chemical analysis panel from which conductivity, dissolved oxygen, and pH are measured.

### 3.4.2.1.1.4 Chemical Drain Tank

During normal operation the liquid and gaseous samples are routed to the chemical drain tank. This tank is then pumped to the Unit 2 waste holdup tank. A sample can be directed to the radwaste module, if analysis is required prior to transfer.



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### 3.4.2.1.1.5 Piping and Fittings

All liquid and gas sample lines are austenitic stainless steel tubing and are designed for high pressure service. With the exception of the sample pressure vessel quick-disconnect couplings and compression fittings at the sample sink and at the liquid sampling panel sump and pump connections, socket-welded joints are used throughout the sampling system. Lines are so located as to protect them from accidental damage during routine operation and maintenance.

### 3.4.2.1.1.6 Valves

Manual or motor-operated stop valves are provided for component isolation and flow path control at all normally accessible sampling system locations. Manual throttle valves are provided to adjust the sample flow rate.

All valves in the system are constructed of austenitic stainless steel or equivalent corrosion resistant material.

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**TABLE 3.4-1**  
**Sampling System Code Requirements**

Code

Piping and valves

USAS B31.1<sub>1</sub>

Notes:

1. USAS B31.1 - Code for pressure piping and special nuclear cases where applicable.

### 3.4 FIGURES

Figure No.	Title
Figure 3.4-1 Sh. 1	Primary Sampling System - Flow Diagram, Sheet 1, Replaced with Drawing 9321-2745
Figure 3.4-1 Sh. 2	Primary Sampling System - Flow Diagram, Sheet 2, Replaced with Drawing 227178

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### 3.5 Fuel Handling System

The fuel handling system provides a safe, effective means of transporting and handling fuel until it leaves the facility after post-irradiation cooling.

The system is designed to minimize the possibility of mishandling or maloperations that could cause fuel damage and potential fission product release.

The fuel handling system consists of the spent fuel pit, which is kept full of water and is always accessible to personnel.

#### 3.5.1 Design Basis

##### 3.5.1.1 Prevention of Fuel Storage Criticality

Criterion: Criticality in the new and spent fuel storage pits shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls. (GDC 66)

The spent fuel storage racks are designed so that it is impossible to insert assemblies in other than the prescribed locations. The spent fuel storage pit has accommodations as defined in Table 3.5-1. In addition, the spent fuel pit has the required spent fuel shipping area. The spent fuel storage pit is filled with borated water. The fuel is stored vertically in an array with sufficient center-to-center distance between assemblies to assure  $K_{eff} < 1.0$  even if unborated water was used to fill the pit and  $\leq 0.95$  when filled with water borated  $\geq 2000$  ppm boron. Limits on enrichment and burnup of fuel in the spent fuel storage pit are given in the Technical Specifications.

Both IP2 and IP3 irradiated fuel assemblies may be handled and stored in the IP2 spent fuel storage pit. The above stated spent fuel storage requirements are being applied to both IP2 and IP3 irradiated fuel assemblies in the IP2 spent fuel storage pit. Detailed instructions are available for use by personnel handling irradiated fuel assemblies. These instructions, the minimum operating conditions, and the design of the fuel handling equipment incorporating built in interlocks and safety features, provide assurance that no incident could occur during the irradiated fuel handling operations that would result in a hazard to public health and safety.

In lieu of maintaining a monitoring system capable of detecting a criticality as described in 10CFR70.24, IP2 has chosen to comply with the seven criteria of 10CFR50.68(b).

##### 3.5.1.2 Fuel and Waste Storage Decay Heat

Criterion: Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities and to waste storage tanks that could result in radioactivity release, which would result in undue risk to the health and safety of the public. (GDC 67)

The spent fuel pit cooling water provides a reliable and adequate cooling medium for spent fuel transfer and heat removal from the spent fuel pit. Overall this is provided by an auxiliary cooling system. Natural radiation and convection is adequate for cooling the holdup tanks.

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### 3.5.1.3 Fuel and Waste Storage Radiation Shielding

Criterion: Adequate shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities. (GDC 68)

Adequate shielding for radiation protection is provided by conducting all spent fuel transfer and storage operations underwater. This permits visual control of the operation at all times while maintaining radiation levels as low as reasonably achievable for the period of occupancy of the area by personnel. Pit water level is indicated, and water removed from the pit must be pumped out since there are no gravity drains. Shielding is provided for waste handling and storage facilities to permit operation within requirements of 10 CFR 20.

Gamma radiation is continuously monitored in the auxiliary building. A high level signal is alarmed locally and is annunciated in the control room.

### 3.5.1.4 Protection Against Radioactivity Release from Spent Fuel and Waste Storage

Criterion: Provisions shall be made in the design of fuel and waste storage facilities such that no undue risk to the health and safety of the public could result from an accidental release of radioactivity. (GDC 69)

All fuel and waste storage facilities are contained and equipment designed so that accidental releases of radioactivity directly to the atmosphere are monitored and do not exceed the applicable limits.

The spent fuel storage pit is a reinforced concrete structure with a seam-welded stainless steel plate liner. This structure is designed to withstand the anticipated earthquake loadings as a seismic Class I structure so that the liner prevents leakage even in the event the reinforced concrete develops cracks.

All vessels in the waste disposal system, which are used for waste storage are designed as seismic Class III equipment.

## 3.5.2 System Design and Operation

The spent fuel pit is kept full of water and is always accessible to personnel.

### 3.5.2.1 Major Structures Required for Fuel Handling

#### 3.5.2.1.1 Spent Fuel Storage Pit

The spent fuel storage pit is designed for the underwater storage of spent fuel assemblies, failed fuel cans if required, control rods and other non-fuel hardware inserts after their removal from both the Unit 2 reactor and the Unit 3 reactor.

The pit accommodations are listed in Table 3.5-1.

Spent fuel assemblies are handled by a long-handled tool suspended from an overhead hoist and manipulated by an operator standing on the movable bridge over the pit.

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The spent fuel storage pit is constructed of reinforced concrete and is seismic Class I design. This structure was analyzed to determine compliance with ACI-318(77), and SRP 3.8 of NUREG-0800. In addition to the mechanical loadings, the pool structure was also analyzed to include the temperature induced loadings. For this purpose, the thermal boundary conditions were conservatively specified as 180°F pool water temperature and 0°F outside ambient. The thermal moments computed by the finite element analyses were combined with those due to mechanical loads. The results of these analyses show that there are large margins between the factored loads and corresponding design strengths.

The pit is lined with a leak-proof stainless steel liner. All welds were vacuum-box tested during construction to assure a leaktight membrane. The effect of a thermal gradient would be to compress the liner. A review of the stress factors resulting from the finite element analyses demonstrates that an adequate design margin exists for the spent fuel pit liner walls and basemat.

Storage racks are provided to hold spent fuel assemblies and are erected on the pit floor. Fuel assemblies are held in a square array, and placed in vertical cells. Fuel inserts are stored in place inside the spent fuel assemblies from both Units 2 and 3.

### 3.5.2.1.2 Storage Rack

High density fuel storage racks have been designed to provide a maximum storage capacity of 1374 locations. The arrangement of the fuel storage racks in the SFP is shown in Figure 3.5-1.

The fuel storage rack arrangement contains two types of storage rack arrays.

Region 1, consisting of three racks that use the flux trap design, can store 269 irradiated fuel assemblies. The flux trap design used in Region 1 uses spacer plates in the axial direction to separate the cells. Boraflex absorber panels are held in place adjacent to each side of the cell by picture-frame sheathing. The spacer plates between cells form a flux trap between the boraflex absorber panels. Note: Boraflex is no longer credited for neutron absorption, but is still physically present in a degraded state.

Each Region I storage cell, as shown in Figure 3.5-2, is a square box with an 8.75 inch inside dimension. Boraflex poison is held in place adjacent to each side of the box by "picture-frame" sheathing. The boxes are assembled into racks with an east-west pitch of 10.765 inches (center-to-center) and a north-south pitch of 10.545 inches, as shown in Figure 3.5-3. A 1/2-inch thick base plate is provided at the bottom of the rack. Adjustable leg supports are welded to the underside of the base plate. A six-inch diameter flow hole is provided in the base plate for each storage cell, and two one-inch holes are provided for cross flow at the bottom of each cell.

Region 2, consisting of nine racks that use the egg-crate design, can store 1105 fuel assemblies and two failed fuel canisters. Region 2 racks consist of boxes welded into a "checkerboard" array with a storage location in each square. One Boraflex absorber panel is held to one side of each cell wall by picture frame sheathing.

The storage racks are positioned on the floor so that adequate clearances are provided between racks and between the rack and pool structure to avoid impacting of the sliding racks during seismic events. The horizontal seismic loads transmitted from the rack structure to the SFP floor are only those associated with friction between the rack structure and the pool liner. The vertical deadweight and seismic loads are transmitted directly to the SFP floor by the support feet.

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### 3.5.2.2 Major Equipment Required for Fuel Handling

#### 3.5.2.2.1 FSB Fuel Handling Bridge Crane

The PaR Systems, Inc. Crane is a wheel-mounted platform, which spans the East-West (E-W) direction of the Spent Fuel Pit (SFP) and travels in the North-South (N-S) direction. The PaR Crane is secured to the crane rails on the FSB El. 95'-0", via seismic hold-down brackets and associated bolting. An Encoder Tracking Device is mounted on the FSB West Walkway, El. 96'-6", which positions the crane in the N-S direction. The crane mounted computer will position the Crane Trolley-Tower Structure in the E-W direction. This equipment will position the Crane Motorized Hoist over a pre-assigned Spent Fuel Assembly (SFA) within the SFP. In addition, the PaR Crane controls interface with the existing FSB Up-Enders Control Console No. 21 (PK1). This computerized control feature provides assurance that the PaR Crane will not interfere with the FSB Up-Enders Assembly, which is located in the Fuel Transfer Canal.

The Motorized Hoist-Sheave Assembly is attached to a Trolley Structure, which is located on the wheel-mounted platform. The Motorized Hoist design incorporates a single lifting cable, which has a safety factor of 11.49:1. This safety factor exceeds the design criteria (10:1) for single lifting cables, as outlined in NUREG-0612. The Tower Structure is mounted on a Motorized Trolley, which travels in the E-W direction on the wheel-mounted work platform. The Motorized Hoist-Sheave Assembly, which has a 1-Ton rated capacity, will transfer SFAs within the SFP via long-handled tools suspended from the hoist hook. The hoist travel and tool length are designed to limit the maximum lift of a SFA and maintain a safe shield depth below the water surface of the SFP. A load weighing system will sense overload and underload conditions. This system will stop the upward movement of a SFA when it senses a load greater than a programmed set-point. In addition, this system will stop the downward movement of a SFA when it senses a slack cable condition.

A 480V, 3-phase, 50 AMP power feed (normal supply) is provided from Distribution Panel No. EP57 to the PaR Crane. In addition, a 480V, 3-phase, 100 AMP power feed (alternate supply) is provided from MCC27 to the PaR Crane. Transfer Switch No. EDA57 is provided so that the reliable power feeds can be provided by Distribution Panel No. EP57 or MCC27.

#### 3.5.2.2.2 Shield Transfer Canister (STC) and HI-TRAC Transfer Cask

The NRC has issued Amendment 268 for the inter-unit transfer of spent fuel from Unit 3 to Unit 2 (Reference 3.5-1). The Amendment is based on evaluations conducted for each aspect of the inter-unit transfer of fuel as documented in the Licensing Report (Reference 3.5-2). The non-proprietary version of the Licensing Report is "incorporated by reference" in the DSAR.

The STC is a thick-walled vessel with a removable top lid capable of transferring up to twelve spent fuel assemblies and associated non-fuel hardware. For inter-unit spent fuel transfer operations between the Unit 3 SFP and the Unit 2 SFP, the STC is used in conjunction with the HI-TRAC transfer cask. During STC closure activities and spent fuel transfer operations, the STC shielding is supplemented with the HI-TRAC shielding (steel, lead and water) and the water contained in the annulus space located between the STC and the HI-TRAC. For inter-unit spent fuel transfer operations, the HI-TRAC uses a solid lid and a centering assembly that keeps the STC centered inside the HI-TRAC cavity. The centering assembly forms an annular region inside the HI-TRAC which remains mostly full of water during loading and transfer operations. An air space is left in the HI-TRAC above the STC top flange to allow the STC lid operations to occur unhindered by water and provide an expansion volume for the water inside the HI-TRAC cavity.

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During spent fuel transfer operations the STC is mostly full of borated water and is steam blanketed to remove air from the STC. The STC includes a removable bolted lid with vent and drain ports for steam blanketing and water filling/draining purposes. The STC lid is coated on the top and sides to protect the carbon steel surfaces from corrosion. Should the coating system be damaged during wet fuel transfer operations, the damaged coating is removed and replaced with N-5000 or vacuum grease to prevent corrosion. The STC lid has lifting devices that can be remotely or manually actuated to engage trunnions on the STC body to lift the STC body when the STC lid bolting is removed. The STC lid also has threaded lid lifting points which provide a means to attach the STC and lid to overhead cranes.

THE STC is moved between Units 3 and 2 vertically in the HI-TRAC. Neither the HI-TRAC nor the STC are handled in the horizontal orientation when loaded with spent fuel assemblies and associated non-fuel hardware. In addition to the water in the STC cavity and the water in the annulus space between the STC and the HI-TRAC's inner shell, the HI-TRAC's water jacket is also filled with water. These three discrete zones of water provide shielding and aid in heat transfer.

### 3.5.3 System Evaluation

Underwater transfer of spent fuel provides essential ease and corresponding safety in handling operations. Water is an effective, economic, and transparent radiation shield and a reliable cooling medium for removal of decay heat.

Gamma radiation levels in the fuel storage area are continuously monitored. These monitors provide an audible alarm at the initiating detector indicating an unsafe condition.

An analysis evaluating the environmental consequences of a fuel handling incident is presented in Section 6.2.1.

Inadvertently locating an unirradiated fuel assembly of 5.0-percent enrichment in a region II storage location has been analyzed. The analysis shows that the array would be subcritical even with no soluble boron poison in the water in the SFP. With a boron concentration of 350 ppm the shutdown margin would be more than 5-percent. The technical specifications require that the boron concentration be maintained at 2000 ppm or more at all times.

### 3.5.4 Minimum Operating Conditions

Minimum operating conditions are specified in the facility Technical Specifications.

### 3.5.5 Control of Heavy Loads

#### 3.5.5.1 Introduction / Licensing Background

A generic letter dated December 22, 1980, required responses to the guidelines of NUREG-0612 "Control of Heavy Loads at Nuclear Power Plants." In response, the IP2 provisions for handling and control of heavy loads at Indian Point Unit 2 were addressed by letters June 22, 1981, September 30, 1982, January 31, 1983, and January 20, 1984. The NRC Safety Evaluation Report in letter dated February 19, 1985, concluded that the guidelines of NUREG-0612, Sections 5.1.1 and 5.3 have been satisfied and the Phase I of this issue for Indian Point Unit 2 is acceptable.



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The NRC Safety Evaluation Report in letter dated November 21, 2005 authorized the use of a single-failure-proof gantry crane for spent fuel cask handling operations up to 110 tons in weight.

Additional information was provided in letter dated July 12, 1996 in response to NRC Bulletin 96-02.

NEI-08-05 R0, "Industry Initiative on Control of Heavy Loads" documented the industry initiative to address NRC staff concerns regarding the interpretation and implementation of regulatory guidance associated with heavy load lifts, was endorsed in Regulatory Issues Summary 2008-28 and has been addressed at Indian Point 2. This supersedes prior head drop analyses.

### 3.5.5.2 Safety Basis

NUREG 0612 has two basic approaches available to demonstrate compliance: demonstrate adequate load handling reliability, or demonstrate that load drop consequences are within the limits of Criteria I-IV listed in Section 5.1 of the NUREG. Both approaches have been utilized in performing the evaluations described in the following sections.

In situations where a demonstration of handling system reliability was employed, the guidelines of NUREG-0612, Section 5.1.6, "Single-Failure Proof Handling Systems," were utilized. The Ederer crane for cask handling was designed as a single failure proof crane.

In situations where a demonstration of limited load drop consequences was employed, a combination of system analyses and structural analyses was utilized. The specific approach chosen was based on the completeness of the available information, and a preliminary assessment of the likelihood of success of the possible approaches.

### 3.5.5.3 Scope of Heavy Load Handling Systems

The following cranes and hoists were determined to be capable of handling heavy loads based on the criteria of NUREG-0612:

- Fuel Handling crane (40/5-ton)
- 110t Ederer crane (110-ton)

The following discuss the results of our evaluations and submittals and are controlled using commitments A-873, A-887, A-1010, A-1015, A-1207, A-2467, A-2491, A-2492, A-2493, A-3174, A-3175, A-3176, A-3179, A-3180 and A-3465.

### 3.5.5.4 Response to NUREG 0612, Phase I Elements

A defense-in-depth approach was used to ensure that all load handling systems are designed and operated so that their probability of failure is appropriately small. The basis for the approach was the Staff guidelines tabulated in Section 5 of NUREG-0612 and the program initiated to ensure that these guidelines are implemented. These guidelines consist of the following criteria from Section 5.1.1 of NUREG-0612:

- Guideline 1 - Safe Load Paths
- Guideline 2 - Load Handling Procedures
- Guideline 3 - Crane Operator Training



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- Guideline 4 - Special Lifting Devices
- Guideline 5 - Lifting Devices (Not Specially Designed)
- Guideline 6 - Cranes (Inspection, Testing, and Maintenance)
- Guideline 7 - Crane Design

Satisfaction of these guidelines for the Fuel Handling Crane and the Ederer Crane is shown in Table 3.5-2.

### Guideline 1 - Safe Load Paths

To ensure that crane operators remain knowledgeable of load handling precautions, annual refresher training is conducted to identify exclusion areas and to review load handling procedures.

In addition to the above procedures, additional structural and systems analyses were performed to determine the consequences of a load drop indicate that suitable system redundancy and structural integrity exist so that the consequences of a load drop would not exceed the criteria of NUREG-0612, Section 5.1.

### Fuel Storage Building Ederer Crane

The Ederer 110-ton design rated gantry crane is used to move spent fuel casks up to 110 tons into and out of the spent fuel pit by lifting a fully loaded Holtec HI-TRAC® 100 or HI-TRAC Version MS spent fuel transfer cask and its associated components. The HI-STORM® cask system utilizes the HI-TRAC® 100 or HI-TRAC Version MS transfer cask for transporting a multi-purpose canister (MPC) from the spent fuel pit, and for inter-cask MPC transfers required for on-site storage. However, this crane is restricted from handling casks over spent fuel in the spent fuel pit and will only be utilized for other loading activities in the FSB.

Safe load paths have been determined, analyzed and documented in procedures for control of heavy loads handled by the Ederer gantry crane. Deviations from the safe load paths will require written alternative procedures reviewed and approved in accordance with IP2 procedures.

The Ederer gantry crane (by design) is unable to move spent fuel casks over any area of the spent fuel pit where the spent fuel is stored.

### Fuel Handling Crane

The Fuel handling Crane may be used to transport equipment, such as inspection rigs or electronics, to the Spent Fuel Pit area. For equipment handling, the crane is utilized to transport loads of no greater than 2000 lbs over the pool area. For fuel handling, the crane may carry a load no heavier than the weight of a fuel assembly containing a control rod assembly, plus the tool and small load block.

No object weighing more than 2,000 pounds may be moved over any region of the spent fuel pit when the pit contains spent fuel, unless a technical analysis has been performed consistent with the requirements of NUREG-0612 establishing the necessary controls to assure that a load drop accident could damage no more than a single fuel assembly. Administrative and procedural controls to protect fuel and fuel racks may include path selection to prevent loads from passing over or near fuel. For cases in which very heavy loads (>30,000 pounds) are transported over the spent fuel pit, the loads cannot under any circumstances pass over irradiated fuel. In all cases where loads >2,000 pounds are carried over the pit, the ventilation system must be functional.

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All standard modes of failure have been considered in the design of the Fuel Handling Crane. These modes of failure were provided for by utilization of a minimum safety factor of 5 based on the ultimate strength of the material used in the design of cables, shafts and keys, gear teeth and brakes.

All crane equipment was sized to handle the single heaviest load realized during facility operation. All lifts are made by qualified personnel. The equipment is properly maintained and periodically inspected by qualified personnel. An analysis of impact loading of the spent fuel cask into the spent fuel storage pool is provided in Section 3.5.3.

Mechanical stops incorporated on the bridge rails of the Fuel Handling Crane make it impossible for the bridge of the crane to travel further north than a point directly over the spot in the spent fuel pit that is reserved for the spent fuel cask. Therefore, it will be impossible to carry any object over the spent fuel storage areas north of the spot in the pit that is reserved for the cask with either the 40 or 5-ton hook of the Fuel Handling Crane. However, to further minimize the potential for a heavy load impacting irradiated fuel in the spent fuel pit, load paths will be defined in procedures and shown on equipment layout drawings.

The mechanical stops may be removed under administrative controls and the crane moved over spent fuel storage areas, provided that the fuel storage building ventilation system is functional, the spent fuel pit boron concentration is at least 2000 ppm and there is no heavy load carried. This allows operations over the spent fuel pit with the 5-ton hoist. The 40-ton hoist may not carry any load over the SFP since the load block is a one-ton load and has not been fully evaluated for heavy loads.

The existing 40-ton non-single-failure-proof Fuel Handling Crane does not have the capacity to handle the HI-TRAC® 100 or HI-TRAC Version MS spent fuel transfer cask and its associated components. Performance of the crane satisfies the objectives of NUREG-0612 and the intent of NUREG-0554 with regard to maintaining the potential for a load drop extremely small.

### Guideline 2 - Load Handling Procedures

A series of operating procedures have been developed for operation of load handling equipment at Indian Point Unit 2.

Load handling procedures provide for the movement of all heavy loads in the vicinity of irradiated fuel or systems and equipment required for decay heat removal, and that load designation was based on the generic load identified in Table 3-1 of NUREG-0612. Further, these procedures contain the precautionary information required by NUREG-0612, Guideline 2. These procedures comply with the commitments made for safe load handling.

The 110T Ederer gantry crane operating procedures utilized for cask and cask component lifts include: identification of required equipment; inspection and acceptance criteria required before load movement; the steps and proper sequence to be followed in handling the load; defining the safe load path; and other precautions. A specific cask loading and handling procedure will provide additional details for controlled movement during cask handling operations.

### Guideline 3 – Crane Operator Training

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A qualification program for the qualification and training of crane operators at Indian Point Unit 2 have been developed to meet the provisions of ANSI B30.2-1976, with no exceptions taken. Crane operator training and qualification is addressed in the qualification program and include precautions and instructions to assure proper operator conduct.

This qualification program meets the requirements of Chapter 2-3 of ANSI B 30.2-1967, "Operation – Overhead and Gantry Cranes", as developed by the American National Safety Code for Cranes, Derricks, Hoists, Jacks and Slings.

### Guideline 4 - Special Lifting Devices

The HI-TRAC® lifting yoke used with the Ederer crane is the only special lifting device that is required to meet the guidelines of ANSI N14.6-1993 and the additional guidelines of NUREG-0612, Section 5.1.6(1)(a).

### Guideline 5 - Lifting Devices (Not Specially Designed)

Facility procedures require that sling selection and use for all loads requiring sling lifting devices be in accordance with ANSI B30.9.

Other lift components utilized with the Ederer Crane and HI-STORM® 100 cask system meet ANSI B30.9-1971 requirements, including the additional guidelines of NUREG-0612, Section 5.1.6(1)(b).

### Guideline 6 - Cranes (Inspection, Testing, and Maintenance)

The 110T Ederer gantry crane is inspected, tested and maintained in accordance with Chapter 2-2 of ANSI B30.2-1976 and the additional guidance contained in NUREG-0612, Section 5.1.1(6) regarding frequency of inspections and test.

### Guideline 7 - Crane Design

A design analysis of each handling system using the design criteria of the applicable standards has been performed.

The 40-ton Fuel Handling Crane was built prior to the issuance of ANSI B30.2-1976 and CMAA-70. However, a detailed point-by-point comparison has been performed, comparing information from the manufacturer with the criteria of these standards. Analysis was performed for only those components that are load bearing or are necessary to prevent conditions which could lead to a load drop. This review indicates that the crane complies with all requirements with the exception of Specification 3.2 of CMAA-70 and Section 2.1.4.1 of ANSI B30.2-1976. These specifications require that welding be performed in accordance with AWS D1.1, 'Structural Welding Code', and AWS D14.1, 'Specifications for Welding Industrial and Mill Cranes'. The welding procedures used are equivalent to current welding criteria based on the following:

- a) welding was performed in accordance with the then-current code AWS D1.1, 'Structural Welding Code'
- b) practices and procedures used for welding are equivalent to those in AWS D14.1, which was not issued at the time

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- c) welders were qualified to existing AWS criteria
- d) all welds were visually inspected

Section 3.5.6, Fuel Storage Building (FSB) Dry Cask Storage (DCS) Operations, provides more detail on the FSB 110-Ton Ederer Single Failure Proof Gantry Crane and Section 3.5.2.2.1 provides more detail on the FSB Fuel Handling Bridge crane.

Additional specific information concerning design compliance with the more restrictive requirements of CMAA-70 is contained in the safety evaluation report.

The 110-ton Ederer gantry crane is installed on a crane rail system. The crane rail system for the Ederer crane consists of crane rail, rail pad, rail clip, sole plate assembly, and sole plate anchor embedments. The sole plate assembly consists of 2" thick steel plate which is held to the concrete slab with 1" diameter rod anchor embedments. The crane rail is attached to the sole plate assembly by rail clips with a rail pad between the crane rail and the sole plate assembly. The crane rail and the concrete slab of the reconstructed truck bay are designed and built to withstand seismic loads, as well as the static loads.

The Ederer gantry crane was designed with a telescopic tower and automated folding cantilever arms to avoid interference with either the existing overhead crane or the refueling bridge crane. During dry cask loading operations the gantry crane will be in its raised position and the existing overhead crane will remain in the south position and de-energized to prevent accidental movement. Once the cask loading operation is completed, the gantry crane will be stored in its far west position, with the tower lowered and the arms folded. This will allow unobstructed use of both the existing overhead and refueling bridge cranes.

The cantilevered girder for the main hoist trolley will extend over the spent fuel pit cask laydown area. The girders are equipped with a retraction mechanism, accomplished via lead screw actuators that allow them to be folded back in order to permit unobstructed use of the existing overhead and refueling bridge cranes. Because of the cantilevered design, the gantry crane requires provisions to ensure stability against overturning. This is accomplished via a floor anchorage system with fixed-in-place hold down features that oppose crane uplift forces. To provide a foundation system capable of resisting these uplift forces, the design includes a steel ballast box filled with steel plates that will act as a counterbalance. The ballast box foundation consists of a 2-foot thick reinforced concrete slab founded on bedrock, and its primary function is to transmit all bearing loads from the weight of the ballast box directly to the underlying bedrock.

The Ederer gantry crane movements are governed by a series of limit and proximity switches that are controlled by a programmable logic controller (PLC) which ensures that: (1) movement of the trolley towards the spent fuel pit is only permitted if turnbuckles are attached to the crane tie down points, cantilever arms are extended and locked in place, and the main transfer hoist is operating at an elevation that allows the HI-TRAC® to clear the south wall of the spent fuel pit; (2) limit switches on the trolley rails limit excessive movement of the trolley to the north and prohibit lowering of the load until a minimum northward travel is reached; and (3) main transfer hoist operation is prohibited until the Ederer gantry crane tower is in its raised position and pinned in place.

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### 3.5.5.5 Single Failure Proof Cranes for Spent Fuel Casks

Sections 3.5.5.4, Response to NUREG 0612, Phase I Elements, and 3.5.6, Fuel Storage Building (FSB) Dry Cask Storage (DCS) Operations, provide more detail on the FSB 110-Ton Ederer Single Failure Proof Gantry Crane.

### 3.5.5.6 Safety Evaluation

The controls implemented to address NUREG-0612 Phase 1 elements make the risk of a load drop very unlikely. The use of increased safety factors for load path elements makes the risk of a load drop extremely unlikely and acceptably low. In the event of a postulated load drop, the consequences are acceptable, as demonstrated by system analyses or the load drop analysis. Restrictions on load height, load weight, and medium under the load are reflected in facility procedures. The risk associated with the movement of heavy loads is evaluated and controlled by station procedures.

The design and use of the Ederer single-failure-proof gantry crane is in accordance with NUREG-0554 and satisfies the guidelines of NUREG-0612. The crane enables the use of the HI-TRAC® transfer cask and associated components with very low risk to irradiated fuel stored in the spent fuel pit. The use of the Ederer single-failure-proof gantry crane for cask handling operations for loads up to 110 tons is approved.

### 3.5.6 Fuel Storage Building (FSB) Dry Cask Storage (DCS) Operations

The 100-Ton Dry Cask Storage System (HI-TRAC, Multi-Purpose Canister (MPC) and HI-STORM Overpack), FSB 110-Ton Single Failure Proof Gantry Crane, FSB Low Profile Transporter (LPT) System, and Vertical Cask Transporter (VCT) facilitate removal of Spent Fuel Assemblies (SFAs) from the Spent Fuel Pool (SFP). During FSB Dry Cask Storage Operations, SFAs are transferred from the SFP with the HI-TRAC / MPC, inserted into the HI-STORM Overpack. The LPT System transfers the HI-STORM Overpack from the FSB to the east side of the PAB / MOB Crossover Walkway. The Vertical Cask Transporter transports the HI-STORM Overpack to the IPEC Independent Spent Fuel Storage Installation (ISFSI) Facility.

#### 3.5.6.1 FSB 110-Ton Ederer Single Failure Proof Gantry Crane

The 110-Ton Ederer Crane was designed to withstand normal operating loads, rated loads, seismic loads and extraordinary loads. The design of the 110-Ton Ederer Crane satisfies the design requirements and safety factors of CMAA-70, NUREG-0554 and Regulatory Guide 1.29. The 110-Ton Ederer Crane conforms to the single failure proof requirements addressed in NRC NUREG-0554 and will retain and control a suspended critical load during and following a Safe Shutdown Earthquake (SSE).

The 110-Ton Ederer Crane is normally located on the west side of the FSB Truck Bay Floor, EL 77'-6" in its lowered position and rests on rails that are installed within the FSB Truck Bay Floor. In order to lift and transfer loaded and unloaded spent fuel casks into and out of the SFP, the 110-Ton Ederer Crane is raised to its upper position. Once in the upper position, the East and West Crane Girder Assemblies, and the North Tie-End Crane Girder Assembly are fully-extended (cantilevered position) out over the SFP. The 110-Ton Ederer Crane is then in position for visually controlled lifting and transferring of loaded and unloaded spent fuel casks into and out of the SFP Cask Pit Area.



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The 110-Ton Ederer Crane is connected, via tie-down turnbuckles, to the 30-Ton Ballast Box, which is embedded within the FSB Truck Bay Floor. The Ballast Box is filled with ~200-Tons of counter weight steel plates. The Ballast Box and tie-down turnbuckles provide stability and restraints for the 110-Ton Ederer Crane during a seismic event. The 110-Ton Ederer Crane is provided with counter weight to reduce the uplift (tension) loads in the tie-down turnbuckles, when the Crane Trolley travels onto the East and West Crane Girder Assemblies towards the SFP Cask Pit Area.

The 110-Ton Ederer Crane is provided with numerous limit switches, which control movement of the 110-Ton Ederer Crane. The limit switches are mounted on the Trolley, Girder Assemblies, Jacking Screw System, and Gantry Crane Structure. The limit switches control the speed of the equipment, limit travel distances and provide interlock signals to defeat some crane functions, when the 110-Ton Ederer Crane is not in the correct configuration. In addition, a Seismic Accelerometer automatically de-energizes the power feed to the 110-Ton Ederer Crane during a seismic event.

### 3.5.6.2 FSB Low Profile Transporter (LPT) System

The FSB LPT System was designed to withstand normal operating loads, maximum vertical – lateral track loads, maximum static stack-up loads, maximum transit loads, and Safe Shutdown Earthquake (SSE) loads. The design of the FSB LPT System satisfies the design requirements and safety factors of AISC, IPEC and Hilman-Rollers.

The FSB Gantry Crane transports the fully-loaded HI-TRAC / MPC towards the east for stake-up onto the HI-STORM Overpack. The LPT Assembly, Chain Drive Assembly and Chain Drive Control Panel control internal FSB movements in the East-West direction.

The FSB LPT System transports the fully-loaded HI-STORM Overpack towards the east and south, so that, the HI-STORM Overpack can exit the FSB. The LPT Transporter Assembly, Transfer Table, Hydraulic Cylinders, and Transfer Table Hydraulic Cart control internal FSB movements in the North-South direction.

The FSB LPT System transports the fully-loaded HI-STORM Overpack through the FSB Truck Bay Roll-Up Door Opening, into the FSB Alleyway Trench and to the east side of the IP2 PAB / MOB Crossover Walkway. The HI-STORM Overpack is in position to be lifted and transported by the Vertical Cask Transporter to the IPEC ISFSI Facility. The LPT Assembly, Track Assemblies, Guide Bars and Aircraft Tugger control external FSB movements in the East-West direction. Empty HI-STORM Overpacks will be transported into the FSB in the reverse order from above.

### 3.5.7 Inter-Unit Spent Fuel Transfer Operations

The NRC has issued Amendment 268 for the inter-unit transfer of spent fuel from Unit 3 to Unit 2 (Reference 3.5-1). The Amendment is based on evaluations conducted for each aspect of the inter-unit transfer of fuel as documented in the Licensing Report (Reference 3.5-2). The non-proprietary version of the Licensing Report is “incorporated by reference” into the DSAR.

In preparation for inter-unit spent fuel transfer operations between the Spent Fuel Pool (SFP) in the IP3 Fuel Storage Building (FSB) and the SFP in the IP2 FSB, the HI-TRAC top lid is removed and the empty shielded transfer canister (STC) is placed inside the HI-TRAC transfer cask. The HI-TRAC / STC Centering Assembly centers the STC inside of the HI-TRAC. The HI-TRAC's

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solid top lid is installed to prevent any spilling of the water during the transfer process. Movement of the HI-TRAC (containing the STC) is performed using the Vertical Cask Transporter (VCT), and using the IP2 Low Profile Transporter (LPT), or using Air Pads at IP3.

THE VCT moves the HI-TRAC containing the empty STC outside the IP3 FSB truck bay door. The HI-TRAC is lowered onto Air Pads and the VCT releases the HI-TRAC. The IP3 FSB truck bay door is opened and the HI-TRAC is positioned inside the IP3 FSB truck bay beneath the FSB cask handling crane using the IP3 Air Pads. The HI-TRAC top lid is removed and the annulus between the STC and HI-TRAC is filled with demineralized water to the required level. The STC lid nuts and washers are removed and the STC is filled with SFP water.

The FSB cask handling crane is positioned over the STC and the STC Lift Lock is fastened to the STC lid and attached to the FSB cask handling crane. The STC is removed from the HI-TRAC and positioned over the cask loading area of the SFP. A set of remotely (or manually) actuated STC Lifting Devices attach the STC lid to the STC lifting trunnions. The STC is lowered into the cask loading area and the lid is removed.

For each fuel transfer cycle, up to twelve IP3 spent fuel assemblies including associated non-fuel hardware are loaded into the STC. The STC lid is positioned over the STC and installed. The STC Lifting Devices attach the lid to the STC lifting trunnions. After the Lifting Device arms are properly engaged to the lifting trunnions, the STC is raised to the surface of the SFP and any standing water on the lid is removed. A small amount of water is removed from the STC to avoid spilling during handling. Under the direction of Radiation Protection personnel radiological controls are established and surveys taken as the STC is raised and removed from the SFP, sprayed with demineralized water and placed directly into the HI-TRAC in the IP3 truck bay. The STC lid, nuts and washers are installed with the nuts left loose. The STC Lift Lock is disconnected from the STC top lid and removed. Free flow verification through the STC lid vent and drain lines is performed. The STC lid nuts are torqued and the STC seals are tested in accordance with ANSI N14.5 to assure that the STC is properly assembled for transfer operations. The required STC water level is established by blowing steam into, and water out of, the STC cavity thereby creating a compressible water vapor space. The STC top lid radiation level is measured to verify compliance with Technical Specification requirements. As required by the Technical Specifications the pressure inside the STC is monitored for a period of 24 hours to demonstrate that there is not a significant amount of air in the STC and that a fuel misload has not occurred. Following completion of the pressure test the STC lid vent and drain port cover plates are installed and the seals are testing in accordance with ANSI N14.5. The HI-TRAC top lid is installed and the bolts are tightened and the seal is tested in accordance with ANSI N14.5. The HI-TRAC side radiation levels are measured to verify compliance with Technical Specification requirements. The IP3 FSB truck bay door is opened and the loaded HI-TRAC is moved outside the IP3 FSB to the VCT on Air Pads using the Prime Mover.

The VCT travels inside the Protected Area on the approved haul route between IP3 and IP2. Prior to each transfer of spent fuel assemblies, the haul route is visually inspected and repaired as necessary.

The HI-TRAC containing the loaded STC is lowered from the VCT onto the IP2 LPT and moved into the IP2 FSB. Inside the IP2 FSB, the HI-TRAC is positioned beneath the 110-Ton Ederer Crane. A drain line containing a pressure gauge is connected to the HI-TRAC top lid vent port and opened relieving any internal pressure. The HI-TRAC top lid bolts are removed and the HI-TRAC top lid is removed. The drain line is then attached to the vent port connection located on

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the lid of the STC and opened relieving any internal STC pressure. STC lid nuts and washers are removed.

The Lift Cleats (with the Lift Cleat Adapter) are attached to the STC lid (the STC Lifting Devices already are installed on the STC lid). The 110-Ton Ederer Crane is attached to the STC through the Lift Cleat Adapter. The STC lifting device arms are engaged with the STC trunnions. Under the direction of Radiation Protection personnel the STC is raised out of the HI-TRAC and positioned directly over the SFP cask loading area and lowered into the pool. IP2 Technical Specification 3.7.12 requires that boron levels in the IP2 SFP have a concentration of greater than 2000 ppm which is also required for the STC spent fuel unloading activities.

With the STC in the SFP cask loading area, the STC Lifting Devices are released from the STC lifting trunnions and the STC lid is removed. The spent fuel assemblies and associated non-fuel hardware are removed from the STC and placed in the SFP racks in accordance with the requirements of IP2 Technical Specification 3.7.13. The STC lid is positioned over the STC and installed. The lid's STC Lifting Devices are attached to the STC lifting trunnions and the STC is raised to the surface of the SFP. Any standing water on the lid is removed. Under the direction of Radiation Protection personnel the STC is raised and removed from the SFP, sprayed with demineralized water, and the water inside the STC is lowered before the STC is placed into the HI-TRAC. The STC lid studs and nuts are installed and the lid studs and nuts are tightened. The Lift Cleats are disconnected from the STC top lid and the Lift Cleats and Lift Cleat Adapter are removed. The HI-TRAC top lid is installed, the bolts are tightened, and the HI-TRAC containing the empty STC is then ready to be returned to the IP3 FSB.

### REFERENCES FOR SECTION 3.5

1. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 268 to Facility Operating License No. DPR-26, July 13, 2012.
2. Holtec Report HI-2094289, Licensing Report on the Inter-Unit Transfer of Spent Nuclear Fuel at Indian Point Energy Center, Revision 10.



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**TABLE 3.5-1**  
**Fuel Handling System Data**

### SPENT FUEL STORAGE PIT

Equivalent fuel assemblies <sup>1</sup>	1374
Number of space accommodations for failed fuel cans	2
Number of space accommodations for spent fuel shipping cask	1
Center-to-center spacing of Region cells, in	10.545(N-S) 10.765(E-W)
Center-to-center spacing of Region cells, in	9.04
Maximum $K_{\text{eff}}$ with borated water (Region)	$\leq 0.95$
Maximum $K_{\text{eff}}$ with unborated water (Region)	$< 1.0$

### MISCELLANEOUS DETAILS

Wall thickness for spent fuel storage pit, ft	3 to 6
Weight of fuel assembly with rod cluster control (dry), lb	1,580

#### Notes:

- <sup>1.</sup> After re-racking.

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**TABLE 3.5-2**  
**NUREG-0612 Compliance Matrix**

Heavy Loads	Weight or Capacity (tons)	Guideline 1 Safe Load Paths	Guideline 2 Procedures	Guideline 3 Crane Operator Training	Guideline 4 Special Lifting Devices	Guideline 5 Slings	Guideline 6 Crane – Test and Inspection	Guideline 7 Crane Design	Interim Measure 1 Technical Specifications	Interim Measure 6 Special Attention
1. Fuel Handling Crane	40	R	R	C	++	C	C	R	R	++
2. 110t Ederer Crane	110	C	C	C	C	C	C	C	++	++

C - Action complies with NUREG-0612 Guideline.

R - Revisions/modifications designed to comply with NUREG-0612 Guideline.

++ - Not applicable.

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### 3.5 FIGURES

Figure No.	Title
Figure 3.5-1	Spent Fuel Storage Rack Layout
Figure 3.5-2	Spent Fuel Storage Cell Region 1
Figure 3.5-3	Region I Cell Cross-Section
Figure 3.5-4	Region II Cross-Section

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### 3.6 Facility Service Systems

#### 3.6.1 Service Water System

##### 3.6.1.1 Design Basis

The service water system is designed to supply cooling water from the Hudson River to various heat loads to support the storage and handling of spent fuel. The system also provides water required for cleaning the traveling screens.

##### 3.6.1.2 System Design and Operation

The service water system flow diagram is shown in Drawings 9321-2722 and 209762 [Formerly Figure 3.6-1, sheets 1 and 2]. Six identical vertical, centrifugal sump-type pumps, each having a capacity of at least 5000 gpm at 220-ft total design head, supply service water to the discharge headers. A rotary-type strainer is in the discharge of each pump, and is designed to remove solids down to 1/16-in. diameter. Each header is connected to an independent supply line. Either of the two supply lines can be used to supply the loads. The loads are those, which are supplied with cooling water from the designated service water header by manually starting a service water pump when required. The minimum flow requirements for the service water system are met by one or more pumps supplying at least 5000 gpm. This ensures that the following loads will be provided with sufficient cooling:

- Spent fuel cooling via the CCW heat exchangers
- TWS wash water and CWP bearing cooling
- 22 Standby Diesel Generator (referred to as 22 Emergency Diesel Generator in site documents)
- Condenser waterbox degassing pumps
- Appendix R/SBO Diesel Generator
- Zurn strainer blowdown
- 13 FWCHX for CENTAC cooling

Water is drawn from the river and passes under a debris wall, through two racks in parallel and finally two traveling screens. Each pump in the circulating water system is installed in an individual chamber while the service water pumps are in a common chamber with two intakes. Each intake is provided with a traveling screen. Openings are also provided between the main circulating water pump chambers and the service water pump chamber. These two openings will be left open.

The service water pumps can therefore obtain water through four separate intakes each equipped with means to prevent debris from entering the pumps, and each capable of supplying all the water required for the service water pumps.

The loads are normally supplied by one or more pumps provided.

The standby diesel-driven generator unit is supplied with cooling water from the service water header.

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The cooling water supply to the Appendix R Diesel Generator Heat Exchanger may be manually realigned from city water to service water.

### 3.6.1.3 Design Evaluation

Sufficient pump capacity is included to provide design service water flow to support the storage of spent fuel in the SFP.

### 3.6.1.4 Tests and Inspections

Each service water pump underwent a hydrostatic test in the shop in which all wetted parts were subjected to a hydrostatic pressure of one-and-one-half times the shutoff head of the pump. In addition, the normal capacity versus head tests were made on each pump.

Valves in the portions of the service water system essential to safety underwent a shop hydrostatic test of 250 psi on the body and 175 psi on the seat. The service water system design pressure is 150 psig.

All service water piping was hydrostatically tested in the field at 225 psig or one-and-one-half times design. The welds in shop-fabricated service water piping were liquid penetrant or magnetic particle inspected in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII.

### 3.6.2 Fire Protection System

Criterion: Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and the control room. Fire detection and protection systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Fire fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components. (GDC 3, Appendix A to 10 CFR 50)

License Condition 2.K of Facility License DPR-26 for IP2 regarding the Fire Protection Program was eliminated in License Amendment No. 294 to reflect the permanently defueled condition of the facility. After the certifications required by 10 CFR 50.82(a)(1) were docketed for IP2, the 10 CFR Part 50 license no longer authorizes operation of the reactor or placement or retention of fuel in the reactor vessel pursuant to 10 CFR 50.82(a)(2). As a result, the fire protection program was revised to take into account the decommissioning facility conditions and activities. IP2 continues to utilize the defense-in-depth concept, placing special emphasis on detection and suppression in order to minimize radiological releases to the environment.

License Condition 2.K, which was based on maintaining an operational fire protection program in accordance with 10 CFR 50.48, with the ability to achieve and maintain safe shut down of the reactor in the event of a fire, is no longer be applicable at IP2. In addition, Appendix R to 10 CFR 50 is no longer be applicable to IP2. However, many of the elements that are applicable for the operating plant fire protection program continue to be applicable during facility decommissioning.

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During the decommissioning process, a fire protection program is required by 10 CFR 50.48(f) to address the potential for fires that could result in a radiological hazard.

The Indian Point 2 Fire Protection Program description is provided separately in the following documents:

- IPEC Fire Protection Program Plan
- IP2 Fire Hazards Analysis Report

These documents provide a complete description of the Indian Point 2 Fire Protection Program including a description of fire areas, fire suppression and detection as well as other fire protection features credited to limit the effects of fires.

### 3.6.3 City Water System

The functions of the city water system are:

1. To provide the water supply for the fire protection system.
2. To provide makeup water to various systems.
3. To provide cooling water to various components.
4. To provide water to areas where hose connections are located for general usage.
5. To provide cooling water to the SBO / Appendix R Diesel Generator Heat Exchanger.

City water for the Indian Point Unit 2 comes from the city water main on Broadway via the Unit 1 mains and storage tanks and is under cathodic protection where the piping crosses the Algonquin Gas pipes. Unit 2 is tied to this system primarily through piping connections at two locations on the low-pressure header (see Drawings 192505, 192506, and 193183 [Formerly Figure 3.6-2]). One connection is in the vicinity of the Unit 1 superheater building on the south side of the header. This connection provides water for:

1. Makeup to the expansion tank of the conventional plant closed cooling system.
2. Cooling to the Appendix R Diesel Generator Heat Exchangers.

The second connection is at the north side of the header. This connection provides water for general usage via hose connections inside the primary auxiliary building and waste holdup tank pit.

A backup water supply is also provided for the circulating water pump seals and bearings.

### 3.6.4 Compressed Air Systems

#### 3.6.4.1 Instrument Air System

The instrument air system is designed such that the instrument air shall be available under all conditions. The system is shown in Drawing 9321-2036 [Formerly Figure 3.6-3] and is provided by the Unit 1 station air system. A connection has been provided in the station air system to allow a backup supply of air from portable compressed air equipment.

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### 3.6.4.2 Station Air System

The station air system shown in Drawing 9321-2035 [Formerly Figure 3.6-4] is supplied by the Unit 1 service air system through a manually operated valve interconnection to the Unit 2 air receiver. The size of the connection is equal to the Unit 2 supply pipe.

### 3.6.5 Heating System

The heating system for Unit 2 represents an extension of the heating system for the Indian Point Unit 1.

Package boilers have been installed to supply steam for Unit 2 and are interconnected with the distribution header of the boilers for Unit 1. The main steam header from these boilers links the existing steam header to Unit 2 and also to Unit 3, so that output from any of the package boilers may be made available for the heating requirements of Unit 1, Unit 2, or Unit 3.

With respect to Unit 2, there are separate piping circuits for the unit heater steam supply to the east side and the west side of the turbine hall, including the heater bay. An extension from the circuit to the east side of the turbine hall serves the turbine oil storage tanks for both clean and dirty oil storage. Other heating services extend to the fan room, the fuel storage building, the primary auxiliary building, and the primary water storage tank.

Provision is made for the following heating services:

1. Primary auxiliary building.
  - a. Electric strip heaters.
  - b. Steam unit heaters.
2. Purge system containment building.
  - a. Air makeup steam tempering units.
3. Fuel storage building.
  - a. Steam unit heaters for standby heating.
  - b. Air makeup steam tempering units. (Steam supply isolated)
4. Fan room.
  - a. One steam unit heater.

### REFERENCES FOR SECTION 3.6

1. Letter from Donald S. Brinkman, NRC, to Stephen B. Bram, Con Edison, Subject: Emergency Amendment to Increase the Service Water Temperature Limit to 90°F (TAC 73764), dated August 7, 1989.

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### 3.6 FIGURES

Figure No.	Title
Figure 3.6-1 Sh. 1	Service Water System - Flow Diagram, Sheet 1, Replaced with Drawing 9321-2722
Figure 3.6-1 Sh. 2	Service Water System - Flow Diagram, Sheet 2, Replaced with Drawing 209762
Figure 3.6-2 Sh. 1	City Water System - Flow Diagram, Sheet 1, Replaced with Drawing 192505
Figure 3.6-2 Sh. 2	City Water System - Flow Diagram, Sheet 2, Replaced with Drawing 192506
Figure 3.6-2 Sh. 3	City Water System - Flow Diagram, Sheet 3, Replaced with Drawing 193183
Figure 3.6-3	Instrument Air - Flow Diagram, Replaced with Drawing 9321-2036
Figure 3.6-4	Station Air - Flow Diagram, Replaced with Drawing 9321-2035



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### 3.7 Equipment And System Decontamination

#### 3.7.1 Design Basis

Activity outside the core can result from fission products from defective fuel elements, fission products from tramp uranium left on the cladding in small quantities during fabrication, products of n- $\gamma$  or n-p reactions on the water or impurities in the water, and activated corrosion products. Fission products in the reactor coolant and tramp uranium are generally removed with the coolant or in subsequent flushing of the system being decontaminated. The products of water activation are not long lived and may be removed by natural decay during subsequent flushing procedures. Activated corrosion products are the primary source of the remaining activity.

The corrosion products contain radioisotopes from the reactor coolant, which have been absorbed on or have diffused into the oxide film. The oxide film, essentially magnetite ( $\text{Fe}_3\text{O}_4$ ) with oxides of other metals including Cr and Ni, can be removed by chemical means presently used in industry.

Water from the primary coolant system and the spent fuel pit is the primary potential source of contamination outside of the corrosion film of the primary coolant system components. The contamination can be spread by various means when access is required. Contact while working on primary coolant system or SFP components can result in contamination of the equipment, tools and clothing of the personnel involved in the maintenance. Also, leakage or spillage from these systems can contaminate the immediate areas and contribute to the contamination of the equipment, tools, and clothing.

#### 3.7.2 Methods of Decontamination

Surface contaminates, which are found on equipment in the primary system and the spent fuel pit that are in contact with the water are removed by conventional techniques of flushing and scrubbing as required. Tools are decontaminated by flushing and scrubbing since the contaminates are generally on the surface only of nonporous materials. Personnel and their clothing are decontaminated according to the standard health physics requirements.

Those areas of the facility, which are susceptible to spillage of radioactive fluids are painted with a sealant to facilitate decontamination that may be required. Generally washing and flushing of the surface are sufficient to remove any radioactivity present.

The corrosion films generally are tightly adhering surface contaminates, and must be removed by chemical processes. The removal of these films is generally done with the aid of commercial vendors who provide both services and formulations. Since decontamination experience with reactors is continually being gained, specific procedures may change for each decontamination case.

Portable components and tools can be cleaned by the use of a liquid abrasive bead decontamination unit, an ultrasonic unit, a sandblast unit or a Freon degreaser unit installed in Unit 1.

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### 3.7.3 Decontamination Facilities

Decontamination facilities onsite consist of an equipment pit and a cask pit located adjacent to the spent fuel storage pit inside Unit 1 on the 70' Fuel Handling floor. In the stainless steel-lined equipment pit, fuel handling tools and other tools can be cleaned and decontaminated.

For the personnel, a decontamination shower and washroom is located on the 72' inside Unit 1 NSB. Personnel decontamination kits with instructions for their use are in the radiation control area decon room.

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### 3.8 Primary Auxiliary Building Ventilation System

#### 3.8.1 Design Basis

The primary auxiliary building ventilation system is designed to accomplish the following:

1. Provide sufficient circulation of air through the various rooms and compartments of the building to remove equipment heat and maintain safe ambient operating temperatures.
2. Control flow direction of airborne radioactivity from low activity areas toward higher activity areas and through monitored exhaust paths.
3. Provide purging of the building to the plant vent for dispersion to the environment.

The air exhausted by the system is monitored and diluted so that offsite dose will not exceed Offsite Dose Calculation Manual (ODCM).

#### 3.8.2 System Design and Operation

The primary auxiliary building ventilation system (See Drawing 9321-4022 is composed of the following systems:

1. Makeup air handling system complete with fan, heating coils, and supply ductwork.
2. Exhaust system complete with fans and ductwork.

Design parameters for the system components are given in Table 3.8-1.

Branch supply ducts direct makeup air to the various floors at the east end of the building, from where it flows to the rooms and compartments. Air is exhausted from each of the building compartments through ductwork designed to make the supply air sweep across the room as it travels to the room exhaust register. The air then flows to the exhaust fan inlet plenum, before discharge to the plant vent. The exhaust system has been designed to ensure that air flows from the "clean" end of the building through the "hot" areas.

Ventilating air exhausted from the waste storage tank pit is arranged to bypass the primary auxiliary building system and flow directly into the exhaust fan inlet plenum.

There are three fans in the primary auxiliary building ventilation system. The two exhaust fans (primary auxiliary building exhaust fans 21 and 22) and the supply fan.

The primary auxiliary building supply fan normally runs, along with either or both of the exhaust fans. The interlocking for the fans is such that in no event will the number of supply fans operating be greater than the number of exhaust fans operating. However, operation of an exhaust fan without a supply fan running is acceptable.

Fans are manually selected. All fans can be started and stopped by discrete control switches located on the fan room control panels. Each fan has indicating lights on the fan room control panel and in the main control room. An auto trip alarm is also provided. In addition, each of the fans have a "jog" pushbutton located on the fan room control panel for testing.

## IP2 DEFUELED SAFETY ANALYSIS REPORT

**TABLE 3.8-1**  
**Primary Auxiliary Building Ventilation System Component Data**

System	Units Installed	Units Capacity	Units Required for Normal Operation
<u>Exhaust<sup>1</sup></u>			
Fans, standard conditions	2	55,500 cfm	1
Fan pressure	-	10.3 in. H <sub>2</sub> O	-
Fan motors	2	125 hp	1
Plenums	2	55,500 cfm	1
<u>Supply Tempering Unit</u> <u>(Primary Auxiliary Building)</u>			
Fans, standard conditions	1	50,400 cfm	1
Fan pressure	1	2.5-in. H <sub>2</sub> O	-
Fan motor	1	50 hp	1
Coils	1	50,400 cfm	1

Notes:

1. These two exhaust fans are used interchangeably for the ventilation of primary auxiliary building.

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### 3.9 Control Room Ventilation System

#### 3.9.1 System Design and Operation

The Unit 2 control room ventilation system is composed of the following equipment:

1. A direct expansion air conditioning unit complete with fan, steam heating coil. The design capacity of the unit is 9200 cfm. A backup fan of the same design capacity has been installed in parallel with the air conditioning unit.
2. Duct system complete with dampers and controls.

The Unit 1 control room ventilation equipment for the central control room has been modified for recirculation mode only.

The control room ventilation systems are shown on Drawings 252665 and 138248 [Formerly Figure 3.9-1]. The Unit 2 control room ventilation system can be operated as follows:

#### 1. Normal Condition

With outside air makeup will supply cooling or heating for the control room atmosphere as required, using fresh outside air makeup.

#### 2. Incident Condition

On toxic gas and/or smoke signal, the outside makeup air will be isolated, the system will be in 100% recirculation mode.

These operations are performed manually from the control room. A redundant toxic chemical and radiation monitor for central control room air intakes has been installed.

### 3.9 FIGURES

Figure No.	Title
Figure 3.9-1	Central Control Room HVAC (Heating, Ventilation, and Air Conditioning), Replaced with Drawings 252665 & 138248

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### 3.10 Fuel Storage Building Ventilation System

#### 3.10.1 Design Basis

The fuel storage building ventilation system is designed to perform the following functions:

1. Maintain the fuel storage building at negative pressure so as to prevent unmonitored releases.
2. Provide sweep ventilation of the building, across the SFP, from areas of low potential contamination to areas of higher potential contamination.
3. Remove normal building heat.

#### 3.10.2 System Design and Operation

The fuel storage building ventilation system consists of an exhaust system. In addition, an axial spot cooling fan circulates 3000 cfm of air to the spent fuel pit heat exchanger room.

The exhaust system consists of registers, ductwork, a filter bank, and a fan. Three exhaust registers are located near the pool surface level, at the north end, and a fourth is near the ceiling at the north end of the building. The registers near the SFP surface are intended to provide a sweep flow over the SFP.

Air from the registers is ducted to a plenum chamber and then to the exhaust fan. Air from the exhaust fan is discharged to the plant vent.

The exhaust fan is the centrifugal type, belt-driven by 100 hp 480-V motor.

The system provides an air flow rate of nominally 20,000 cfm. The system is balanced to divide the exhaust air flow equally between the exhaust registers and to maintain the building at a slight negative pressure. The exhaust fan is operated and controlled from a single local control room.

#### 3.10.3 Limiting Conditions for Operation

The fuel storage building ventilation system is not required to be operating to mitigate the consequences of a Fuel Handling Accident (FHA) in the Fuel Storage Building. The ventilation system must be functional (able to be operated), whenever spent fuel movement is taking place within the spent fuel storage areas so that in the event of a FHA it can manually be put into operation to monitor the release through the plant vent, if not already operating.

#### 3.10.4 Surveillance Requirements

The fuel storage building ventilation system does not have to be demonstrated functional in the assumed configuration prior to handling fuel. The fuel storage building ventilation system shall be periodically tested to verify that the system maintains the spent fuel storage pool area at a pressure less than that of the outside atmosphere during system operation at least once each 24 months.

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### 3.11 Circulating Water System

The circulating water system provides dilution flow for liquid waste discharges.

Hudson River water is used for the condenser circulating water. River water flows under the floating debris skimmer wall, through traveling screens, and into two separate screenwells. The traveling screens, which operate continuously, are designed to reduce the potential for fish and debris from entering the circulating water pumps. Each screenwell is provided with stop logs to allow dewatering of any individual screenwell for maintenance purposes.

The water from each individual screenwell flows to a motor-driven, vertical, mixed flow condenser circulating water pump. Each of the two condenser circulating water pumps provides 140,000 gpm and 21-ft total dynamic head when operating at 254 rpm and 84,000 gpm and 15-ft total dynamic head when operating at 187 rpm. Each pump is located in an individual pump well, thus tying a section of the condenser to an individual pump. The circulating water is piped to the condensers and is discharged back into the river.

A surface-type, single-pass, radial flow condenser (#22) with a bolted divided water box at both ends is provided. Fabricated steel water box and shell construction is used. Condenser #22 uses titanium tubes and tube sheets. Water box manholes are provided for access. The design parameters for Condenser #22 are given in Table 3.11-1.

The pressure-retaining components or compartments of components comply, as a minimum, with the codes detailed in Table 3.11-2.

**TABLE 3.11-1**  
**Design Parameters for Condenser #22**

Type	Radial flow, single-pass, divided water box, deaerating
Number	1

**TABLE 3.11-2**  
**Codes and Classifications**

System pressure vessels and pump casing	ASME Boiler and Pressure Vessel Code, Section VIII
System valves, fittings, and piping	USAS Section B31.1 Power Piping Code (1955) ASA, USAS, ANSI
Pressure Testing of Repairs and Modifications	USAS Section B31.1 Power Piping Code (1992)

### 3.11 FIGURES

Figure No.	Title
Figure 3.11-1	Condenser Air Removal and Water Box Priming - Flow Diagram, Replaced with Drawing 9321-2025

## IP2 DEFUELED SAFETY ANALYSIS REPORT

### 3.12 LEAKAGE DETECTION AND PROVISIONS FOR THE AUXILIARY COOLANT LOOPS

#### 3.12.1 Design Bases

##### Monitoring Radioactivity Releases

Criterion: Means shall be provided for monitoring the containment atmosphere and the facility effluent discharge paths for radioactivity released from normal operations, from anticipated transients, and from accident conditions. An environmental monitoring program shall be maintained to confirm that radioactivity releases to the environs of the plant have not been excessive. (GDC 17)

IP2 has been permanently shut down and defueled. As a result, there are no abnormal operations, transient, or accidents that credited the containment for isolation. The component cooling loop liquid is monitored for radioactivity concentration during normal operation. The Offsite Dose Calculation Manual (ODCM) provides the methodology to calculate radiation dose rates and dose to individual persons in unrestricted areas in the vicinity of Indian Point due to the routine release of liquid effluents to the discharge canal. The ODCM also provides setpoint methodology that is applied to effluent monitors and optionally to other process monitors.

#### 3.12.2 Systems Design and Operation

For relevant systems located outside the containment, leakage is determined by one or more of the following methods:

1. For systems containing radioactive fluids, leakage to the atmosphere would result in an increase in local atmospheric activity levels and would be detected by either the plant vent monitors or by one of the area radiation monitors. Similarly, leakage to other systems that do not normally contain radioactive fluids would result in an increase in the activity level in that system.
2. For closed systems such as the component cooling system, leakage would result in a reduction in fluid inventory.
3. All leakage would collect in specific areas of the building for subsequent handling by the building drainage systems, e.g., leakage from the service water loop would collect in the sumps provided, and would result in the operation, or increased operation, of the associated sump pumps and increased inventory in the liquid waste processing system.

The relevant fluid systems for which no special leak detection outside containment is provided include the following:

1. Component cooling.
2. Service water.
3. Waste disposal.

Various methods are used to detect leakage from the auxiliary loops. Although described to some extent under each system description, all methods are included here for completeness.



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### 3.12.3 Components

#### 3.12.3.1 Component Cooling Liquid Monitor

This channel continuously monitors the component cooling loop of the auxiliary coolant system for activity indicative of a leak from the spent fuel pool cooling system. A scintillation detector is installed in the local radiation monitor skid assembly. This assembly is located in the primary auxiliary building and receives sample flow from the component cooling pump discharge downstream of the component cooling heat exchangers. The detector assembly output is amplified by a preamplifier, processed and transmitted to the radiation monitoring system console, the display console and a recorder in the control room. The activity is indicated on digital displays. High-activity alarm indications are displayed on the control board annunciator and the display console.

The measuring range of this monitor is  $10^{-5}$  to  $10^{-2}$   $\mu\text{Ci}/\text{cm}^3$ .

#### 3.12.3.2 Component Cooling Loop

Leakage outside containment depending on location will be diverted by floor drains to either the PAB sump tank or PAB sump where it will then be transferred to the Waste Holdup Tank.

#### 3.12.3.3 Service Water System

Leakage outside containment depending on location will be diverted by floor drains to either the PAB sump tank or PAB sump where it will then be transferred to the Waste Holdup Tank.

### 3.12.4 Leakage Provisions

Provisions are made for the isolation and containment of any leakage.

#### 3.12.4.1 Design Basis

The provisions made for leakage are designed to prevent uncontrolled leaking of auxiliary cooling water. This is accomplished by routing the leakage to various sumps and holdup tanks.

#### 3.12.4.2 Design and Operation

Various provisions for leakage avert uncontrolled leakage from the auxiliary coolant loops.

#### 3.12.4.3 Component Cooling Loop

Leakage outside containment depending on location will be diverted by floor drains to either the PAB sump tank or PAB sump where it will then be transferred to the Waste Holdup Tank.

Other provisions made for leakage from the component cooling loop are discussed in Section 3.3.

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### **3.12.4.4 Service Water System**

Leakage outside containment depending on location will be diverted by floor drains to either the PAB sump tank or PAB sump where it will then be transferred to the Waste Holdup Tank.

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### 3.13 Information Display and Recording

Alarms and annunciators in the central control room provide site personnel with warning of abnormal facility conditions that might lead to the damage of components, fuel, or other unsafe conditions. Other displays and recorders are provided for indication of routine conditions and for the maintenance of records.

Control and display equipment for station auxiliary systems are located on the control board.

The auxiliary electrical system controls required for manual switching between the various power sources described in Section 3.15.1.2 are provided to the left of the control board.

Controls and indications Primary Auxiliary Building and Fuel Service Building ventilation systems are located on CCR panel SL.

Audible alarms will be sounded in appropriate areas throughout the station if high-radiation conditions are present.

A process computer system is installed with color graphic displays in the central control room that monitors facility data as well as easily accessible sets of key facility safety parameters. It also provides data links with the technical support center, the emergency operations facility and the Alternate emergency operations facility. It has the capability of long-term data storage and retrieval.

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### 3.14 Communications

Facility communications are conducted via telephone, radio, and Public Address (paging) systems.

The facility telephone and radio communications systems include PBX electronic switches, backup phone lines and a UHF radio system.

The public address system for Indian Point Unit 2 consists of "Page" and "Party" communications, which are common to both the primary (nuclear) and secondary (conventional) portions of Units 1 and 2. The "Page" and "Party" communications are also monitored at a speaker panel located in the CCR. Radio channels are available at the Indian Point Unit 2 control room. These radio channels are as follows:

1. Radios provide the central control room with radio communication to site personnel.
2. Indian Point area radios provide the central control room with radio communication to the emergency response facilities and offsite monitoring teams.

If the control room were to become inaccessible, communications would be conducted with the use of portable radios. This in-house radio system is also provided for communicating with site personnel.

#### 3.14.1 Central Control Room Communication Facilities

The central control room is provided with telephone-radio-page/party communication consoles and page/party handset stations.

A State/County Radiological Emergency Communication System (RECS) hotline is available. The NRC Emergency Notification System (ENS) hotline is available in a separate location.

A separate printer and its telephone modem are also available for meteorological data reception.

#### 3.14.2 Radio Communication

The radio channels are available at the radio line consoles in the central control room.

Repeaters for the radio channels are located onsite. Wired audio/control pairs connect the transceivers with the communication consoles in the central control room for remote operation.

#### 3.14.3 Page/Party Line Communication

"Page" or "Party" line communication can be initiated in the CCR from either communication consoles or from handset stations.

An emergency alarm switch is provided in the CCR to connect and actuate the existing alarm oscillators to the "Page" system for the "Evacuation," "Fire," or "Air Raid" alert signals.

Another switch is provided on the central control room desk, which allows all outdoor speakers of the Indian Point 2 facility to be turned off at night.

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### 3.14.4 Backup Power for Communications

The facility radio and telephone communications systems are automatically supplied from a back-up power source, upon failure of the normal power source. In addition, each PBX is provided with back-up battery capability of eight (8) hours of operation. The page/party system is powered from the DC system (through an inverter) with backup power from a bus.

### 3.14.5 In-house Radio System

An in-house radio system provides communications between personnel at the facility. Field units are low-wattage, hand-held units, which are not to be used in areas containing equipment, which is potentially sensitive to radio-frequency interference.

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### 3.15 Electrical Systems

The function of the auxiliary electrical system is to provide reliable power to those auxiliaries required during any normal facility conditions. Sufficient independence and isolation between the various sources of electrical power is provided in order to guard against concurrent loss of all auxiliary power. The facility is supplied with normal and standby power sources. Facility power is provided by a 13.8-kV / 6.9-kV autotransformer.

A diesel-generator set supplies standby power to the facility in the event of a loss of AC auxiliary power. There are no automatic bus ties associated with these buses. The Station Blackout (SBO) / Appendix R diesel-generator is installed in the Unit 1 Turbine Building and is used to supply standby power for the facility.

The standby diesel-generator set is located in the Diesel Generator Building adjacent to the Primary Auxiliary Building and supplies a source of standby power. It may be started manually upon the occurrence of an undervoltage condition on any 480-V switchgear bus. The standby diesel is adequate to provide standby power to ensure the safe storage of spent fuel at the facility in lieu of the SBO / Appendix R diesel generator.

All electrical systems and components that were historically vital to plant safety, including the standby diesel generator, were classified as seismic Class I and were designed so that their integrity was not impaired by the design-basis earthquake, certain wind storms, floods, or disturbances on the external electrical system.

The 13.8-kV / 6.9-kV autotransformer are adequate to run all of the facility auxiliary loads.

The 125-V DC power supply consists of a battery charger and a battery panel. Under all conditions, the battery charger supplies all DC loads. DC control power for the 480-V Switchgear and the standby diesel generator is supplied via the 125-V DC system. This design ensures that adequate DC power is available.

The 6.9-kV system is arranged as six buses. The buses receive power from the offsite 13.8 kV source. Buses 2, 3, 5, and 6 each serve one of the four 6900-V / 480-V station service transformers. Normal and offsite power to the 480-V switchgear buses is supplied through these station service transformers.

The 480-V system is arranged as four switchgear buses. Each 480-V switchgear bus supplies several 480-V motor control center buses for power distribution throughout the facility. The 480-V switchgear buses are supplied from the 6.9-kV buses as follows: 2A from 2, 3A from 3, 5A from 5, and 6A from 6. Tie breakers are provided between 480-V switchgear buses 2A and 3A, 2A and 5A, and 3A and 6A.

Power for instrumentation and control is provided by four 118-V AC Instrument Supply Systems. Each system consists of one manual bypass switch, two 118-V AC buses, and associated interconnections. The four systems are powered from transformers supplied from 480-V MCCs.

Several sources of offsite power are available to Indian Point Unit 2. These consist of three separate underground feeders from the Buchanan 13.8-kV substation. The 13.8-kV line is rated 19.8 MVA at 13.8-kV. The 13.8-kV / 6.9-kV autotransformer is rated 20 MVA. No safety or emergency power is required from these sources for the retired Indian Point Unit 1.

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The source of offsite power from the 13.8-kV distribution system at Buchanan is available to 6.9-kV buses 5 and 6 through supply breakers GT-25 and GT-26. The transfer from the normal to the reserve supply (or vice versa) must be accomplished manually.

The diversity and redundancy inherent in the combination of offsite electrical systems minimize the probability of losing electric power from any of the remaining sources as a result of, or coincident with, the loss of power from the transmission network, or the loss of onsite power sources.

### 3.15.1 ELECTRICAL SYSTEM DESIGN

#### 3.15.1.1 Network Interconnections

The external transmission system provides auxiliary power as required by the facility.

The electrical one-line diagram for the Indian Point Station is presented in Drawing 250907 [Formerly Figure 3.15-1]. Power is supplied to the facility from the Buchanan 138-kV Substation. Several power flow paths exist to connect 13.8-kV buses through 13.8-kV / 6.9-kV autotransformers to Buses 5 and 6.

##### 3.15.1.1.1 Reliability Assurance

Two external sources of power are available to Indian Point Unit 2. They are the 138-kV tie from the Buchanan 345-kV / 138-kV autotransformer and the 138-kV Buchanan-Millwood ties. These 138-kV ties normally power the 13.8-kV sources via 138-kV / 13.8-kV transformers in the Buchanan switchyard. Loss of any of these 138-kV sources will not affect the other. Substantial flexibility and alternate paths exist within each source.

The 138-kV supply from the Buchanan substation with its connections to the Con Edison 345-kV system provides a dependable source of station auxiliary power. Upon loss of 345-kV / 138-kV auto-transformer supply at Buchanan, two 138-kV ties are designed to provide additional auxiliary power from the Millwood 138-kV substation. A further guarantee of reliable auxiliary power, independent of transmission system connections, is provided by the SBO / Appendix R Diesel. The SBO / Appendix R Diesel, associated switchgear and breakers minimum operating requirements are specified in Appendix B of the DSAR (the Unit 2 Technical Requirements Manual (TRM)). A minimum quantity of fuel for the SBO / Appendix R Diesel shall be available at all times the SBO / Appendix R Diesel is considered functional. Support systems for cooling include the City Water Storage Tank and the Service Water System (SWS) (first the city water and then a switch to the SWS). If these requirements cannot be met, then the SBO / Appendix R diesel generator is considered non-functional and the TRM requirements are followed.

The fuel supply consists of two onsite 30,000-gal fuel oil tanks. A minimum amount of fuel is maintained available and dedicated for the SBO / Appendix R Diesel. This minimum fuel inventory ensures that the SBO / Appendix R Diesel will be capable of supplying the maximum electrical load for the Indian Point Unit 2. Commercial oil supplies and trucking facilities exist to ensure deliveries of additional fuel within one day's notice.

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### 3.15.1.2 Distribution System

The auxiliary electrical system is designed to provide a simple arrangement of buses requiring a minimum of switching to restore power to a bus in the event that the normal supply is lost.

The basic components of the facility electrical system are shown on the electrical one-line diagrams (See Drawings 208377, 231592, 208088, 9321-3004, 249956, 9321-3005, 208507, 249955, 208241, 9321-3006, 248513, 208500, 208502, 208503, 208501, 9321-3008, and Figure 3.15-4 [Formerly Figures 3.15-3, and 3.15-5 through 3.15-16]), which include the 13.8-kV, the 6.9-kV, the 480-V, the 118-V AC instrument, and the 125-V DC systems.

#### 3.15.1.2.1 Gas Turbine Autotransformer and Station Service Transformers

Power to the auxiliaries on the 6.9-kV buses is supplied by a 13.8-kV / 6.9-kV two-winding autotransformer connected to an offsite supply. Power to the 480-V buses is supplied from four 6900-V / 480-V, air-insulated, dry-type station service transformers.

These transformers were designed and constructed in accordance with ANSI C57.11, as the applicable standard of record at the time of fabrication. During engineered safeguards loading and operation, these transformers are loaded within their ratings. Manufacturer shop tests of the transformers were conducted in accordance with the American Standard Test Code C 57.12.90. This series of tests consisted of the following:

1. Resistance measurements of all windings.
2. Ratio tests.
3. Polarity and phase relation tests.
4. No-load losses.
5. Exciting current.
6. Impedance and load loss.
7. Temperature test.
8. Applied potential tests.
9. Induced potential tests.

The normal source of power to the 480-V buses is supplied from the 138-kV switchyard.

#### 3.15.1.2.2 6.9-kV System

The 6.9-kV system is arranged as six buses that receive power from the 13.8-kV system by bus main breakers and the 13.8-kV / 6.9-kV autotransformer. Buses 2, 3, 5, and 6 each serve one 6900-V / 480-V station service transformer.

#### 3.15.1.2.3 480-Volt System

The 480-V system is arranged as Switchgear buses 2A, 3A, 5A, and 6A and numerous motor control center buses. The 480-V switchgear buses are supplied from the 6.9-kV buses as follows: 2A from 2, 3A from 3, 5A from 5, and 6A from 6 (buses 2A and 3A are within the same power train). Tie breakers are provided between 480-V Switchgear buses 2A and 3A, 2A and 5A, and 3A and 6A.



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All four 480-V switchgear buses supply power to systems and equipment. A single source of DC control power is provided for control of 480-V breakers, protective circuits and other devices.

The sources of DC control power for the breakers are:

Transfer Switch	Associated Bus	Source
EDD1	6A	DC PP #22
EDD2	2A	DC PP #22
EDD3	3A	DC PP #21
EDD4	5A	DC PP #21

### 3.15.1.2.4 125-V DC Systems

There is a single 125-V DC system serving the various DC loads throughout the facility. The system consists of one battery, one battery charger and one or more DC panels.

The battery charger is supplied from a 480-V motor control center bus. The battery charger supplies the DC loads.

### 3.15.1.2.5 118-V AC Instrument Supply Systems

There are four 118-V AC instrument supply systems serving the various instrumentation and control systems throughout the facility. Each system consists of one manual switch and two 118-V AC instrument buses (See Drawing 250970 [Formerly Figure 3.15-2] for system arrangement and connections to power sources). All four systems are supplied from transformers supplied from 480-V MCCs. The 118-V AC system manual transfer switch is mounted in a separate enclosure and will bypass the static transfer switch and provide power from the step-down transformers directly to the 118-V AC buses. Voltage drop calculations demonstrate that equipment supplied from Buses 21 and 21A are operable with the postulated minimum source voltage. This is typical of all instrument buses.

### 3.15.1.2.6 Evaluation of Layout and Load Distribution

Electrical distribution system equipment is located to minimize the exposure of relevant circuits to physical damage as a result of natural phenomena. To a certain extent the Diesel-Generator Building is protected from tornados and major tornado generated major missiles because it is situated between large buildings as shown in the site plot plan (Drawing 327152). The diesel-generator installation is considered redundant to other lines of power supply. As described in Section 3.15, there are alternate power supplies. In the case of a tornado, reliance is placed on power supply redundancy and not solely on the diesel installation.

The 6.9-kV buses are housed in two metal-clad switchgear units. The enclosures for switchgear 21 and 22 are located at elevation 15 ft in the turbine building. Each breaker is mounted in a separate compartment. Switchgear 21 and 22 have a solid top with cable penetrations and some openings on the side. The cable openings at the top are sealed to minimize bus exposure to fire, water, and other physical damage. An overcurrent condition on any of the 6.9-kV buses actuates the associated bus protection lockout relays, which isolate the bus by tripping and locking out both the normal supply breaker and the 6.9-kV tie breaker for that bus.

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The 480-V buses are housed in two metal-enclosed switchgear units located at the 15-ft elevation of the Indian Point Unit 2 control building. The switchgear structure provides protection to minimize exposure from mechanical, fire, and water damage. Buses 5A and 2A are contained in switchgear enclosure 21; Buses 6A and 3A constitute switchgear enclosure 22. The switchgear contains the buses, the bus supply breakers, the tie breakers, the load (feeder) breakers, the station service transformers, and the potential transformers for synchronizing and under-voltage relay protection. The normal 480-V switchgear supply breakers 52/2A, 52/3A, 52/5A, and 52/6A are tripped under the following conditions:

1. Loss of voltage (~46-percent) on bus 5A or 6A.
2. Actuation of manual trip pushbuttons on each breaker.
3. Actuation of control switches in the Central Control Room.
4. Actuation of control switches in the Diesel-Generator Building.
5. Individual breaker overcurrent protection.

A separate category alarm and bullet lights in the central control room will alert site personnel when any 480-V switchgear bus voltage falls to 94-percent. These are primarily intended to alert site personnel to sustained degraded voltages that result from problems on the offsite power system.

Remote manual and automatic control of the 480-V switchgear breakers and associated relays requires 125-V DC control power.

Each 480-V switchgear breaker is equipped with a Westinghouse "Amptector 1A"\* solid-state overcurrent trip unit to protect the auxiliary equipment supplied by the breaker (including cables) and the associated switchgear. The settings of the solid-state overcurrent trip unit are based on the supplied load. The solid-state trip unit is provided with an instantaneous and/or short-time setting(s) to protect against fault conditions, and long-time setting to protect against over-load conditions.

Each circuit breaker is tripped on overcurrent conditions (overload or short circuit) by the combined operations of three components:

1. Sensors
2. Amptector solid-state trip unit \*
3. Actuator

All necessary tripping energy (for a breaker trip on an overcurrent condition only) is derived from the load current flowing through the sensors; no separate power source is required. The tripping characteristics for a specific breaker rating, as established by the sensor rating, are determined by the continuously variable settings of the Amptector\* static trip unit. This unit supplies a pulse of tripping current (when preselected conditions of current magnitude and duration are exceeded) to the actuator, which produces a mechanical force to trip the breaker.

If an overcurrent condition occurs on one of the 480-V switchgear buses while the bus is supplied from the normal source, lockout relays trip (if required) and prevent the closing of the alternate supply breakers (standby diesel and bus ties) associated with the bus. These relays must be manually reset after the overcurrent condition is cleared to allow these breakers to close.

The 480-V motor control centers are located in the areas of electrical load concentration.

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\* Note that the “Amptector” may be replaced by the equivalent Westinghouse device, “Westector.”

The Indian Point Unit 2 Cable Raceway System is comprised of 4 raceway systems. 6.9kV cables are routed in their own raceway system independent of the other raceway systems. 480 VAC and 125 VDC cable 350 mcm and larger are routed in the heavy Power Raceway. Those cables smaller than 350 mcm and over 65VAC are routed in the Small Power and control Raceway. Instrument cables 65VAC and less are run in the Instrument Raceway. Instrument cables less than 65VAC are typically routed in the Instrument Raceway. On a case by case basis, cables have been routed in an alternate raceway however there is no mixing between the 6.9kV raceway and cables of lower voltages. Certain other cables such as thermocouple cable, public address, instrument power and fiber optics are routed in raceway as convenient.

The application and routing of control, instrumentation, and power cables minimize their exposure to damage from any source. All cables are designed using conservative margins with respect to their current carrying capacities, insulation properties, and mechanical construction. All cables are fire resistant.

Cable loading of trays and consequently heat dissipation of cable throughout the facility has been carefully studied and controlled to ensure that there is no overloading. The criteria for electrical loading were developed using IPCEA (now ICEA) Standard P-46-426, manufacturer recommendations, and good engineering practice.

Derating factors for cables in trays without maintained spacing are taken from Table VIII of the IPCEA publication. Derating factors for the maximum ambient temperature existing in any area of the facility are also taken from the IPCEA publication. These factors are applied against ampacities selected from appropriate tables in other portions of the standard.

For physical loading of trays, the following criteria are followed: for 6.9-kV power, one horizontal row of cables is allowed in a tray; for heavy power, two horizontal rows of cables are allowed; for medium power, small power & control or instrumentation, 70-percent of the cross-sectional area of a tray is the maximum fill, with the heavy power cables limited to two horizontal rows. During initial plant construction, a computer program monitored the loading and prevented the routing of anything greater than this amount.

To ensure that only fire-retardant cables are used throughout the facility, a careful study of cable insulation systems was undertaken early in the design of the plant. Insulation systems that appeared to have superior flame-retardant capability were selected and manufacturers were invited to submit cable samples for testing. An extensive flame testing program was conducted including ASTM vertical flame and Con Edison vertical flame and bonfire tests. A report summarizing the testing was prepared by Con Edison. These tests were used as one of the means of qualifying cables, and the specifications were written on the basis of the results.

The following tests were made to determine the flame-retardant qualities of the covering and insulations of various types of cables for Indian Point Unit 2:

1. Standard Vertical Flame Test - made in accordance with ASTM-D-470-59T, “Tests for Rubber and Thermoplastic Insulated Wire and Cable.”

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2. Five-Minute Vertical Flame Test - made with cable held in vertical position and 1750°F flame applied for 5 min.
3. Bonfire Test - consisted of exposing bundles of three or six cables to flame produced by igniting transformer oil in a 12-in. pail for 5 min. The cable bundles were supported horizontally over the center of the pail with the lowest cable 3 in. above the top of the pail. The time required to ignite the cable and the time the cable continued to flame after the fire was extinguished were noted.

In those areas where the compressed instrument air system is near the essential 480-V switchgear, the following provision is incorporated to shield this switchgear and cabling from potential missiles or pipe whip:

1. The compressed instrument air lines in the vicinity of the switchgear are supported at the piping bends. This will resist any step loading of PA (which could occur in the event of an instantaneous circumferential rupture) without occurrence of a "plastic hinge." The possibility of pipe whip is eliminated.

These provisions ensure that no missile or whipping pipe originating from postulated failures in the compressed instrument air system will strike the switchgear.

### 3.15.1.3 Standby Power

#### 3.15.1.3.1 Source Descriptions

The source of offsite power is the Con Edison 138-kV system. A diesel-generator set provides a source of onsite standby power that can be used in lieu of the SBO / Appendix R diesel generator. The diesel generator set is an Alco Model 16-251-E engine coupled to a Westinghouse 900 rpm, 3-phase, 60-cycle, 480-V generator. The unit has a capability of 1750 kW (continuous), 2300 kW for 1/2 hour in any 24-hour period, and 2100 kW for 2 hours in any 24-hour period.

The standby diesel unit, which is a backup to the normal standby AC power supply, is capable of supplying the power requirement for facility equipment. The unit is installed in a historically classified seismic Class I structure located near the Primary Auxiliary Building.

The standby diesel is manually started by two redundant air motors, each unit having a complete 53-ft<sup>3</sup> air storage tank and compressor system powered by the 480-V distribution system. Each air receiver has sufficient storage for four normal starts. However, the standby diesel will consume only enough air for one automatic start during any particular power failure. Additionally, the engine control system is designed to shut down and lock out any engine that did not start during the initial try.

To ensure rapid start, the unit is equipped with water jacket and lube-oil heating. A pre-lube pump circulates the oil when a unit is not running. The unit is located in a heated room.

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Audible and visual alarms are located in the control room and in the diesel generator building. Alarms on the electrical annunciator panel in the control room are:

1. Diesel-generator trouble.
2. Diesel-generator oil storage tank low level.
3. Diesel-Generator Trouble.
4. Diesel-Generator Service Water Flow Low.

The activation of the standby diesel generator trouble alarm in the control room will be caused by the initiation of any of the following alarms in the diesel generator building:

1. Low oil pressure.
2. Differential fuel strainer, secondary.
3. Over-crank.
4. High differential lube-oil strainer.
5. High water temperature.
6. High differential pressure lube-oil filter.
7. High-high jacket water temperature.
8. Overspeed.
9. Overcurrent.
10. Low fuel oil level, day tank.
11. Reverse power.
12. Low start air pressure.
13. Exciter field shutdown.
14. High/Low lube-oil temperature.
15. High differential pressure primary filter.

The diesel-generator oil storage tank low level alarm will be energized on a low level in the fuel-oil storage tank.

The alarm "Diesel-Generator Trouble" located on Panel SG in the Central Control Room will be activated respectively by the following conditions at the diesel general local control panel:

1. Loss of DC control power.
2. Engine control switch position (Off or Manual).
3. Breaker control switch position pulled-out [Note - the breaker control switch in the CCR will activate the "Safeguards Equipment Locked Open" alarm (Window 1-8 on Panel SB-1) in the CCR].
4. Engine stop solenoid energized.
5. Day tank level low, primary and backup fuel pump fails to start.

There are six electrical contacts, each of which when activated will energize a diesel-generator lockout relay. This lockout relay will, in turn, cause the standby diesel to shut down if it is operating. These contacts are activated by one of the following conditions:

1. Activation of the standby diesel emergency stop push-button in the diesel-generator building.
2. Activation of the overcurrent relay. A phase-to-phase fault or excessive loads on the standby diesel generator will operate this relay.

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3. Activation of the reverse power relay.
4. Activation of the over-crank relay. If the standby diesel engine fails to attain speed within 13 sec, this relay will be energized.
5. Activation of the overspeed relay. When the mechanical governor senses 1070 rpm, this relay will be energized.
6. Activation of the low oil pressure relay. This relay is energized by the coincident sensing of lube-oil pressure below 60 psi by two of the three oil pressure switches for the standby diesel. An oil pressure timer is set to allow 20 sec to pass before tripping the standby diesel engine lockout relay. This circuit is designed to provide sufficient time for the oil pressure to build up following an engine start.

Shutdown permits corrective action to be taken before the engine is damaged, and the standby diesel generator can then be returned to normal operation. Once any of these six electrical contacts has been activated causing the standby diesel engine lockout relay to energize, the lockout relay must be manually reset locally before the diesel can be started.

### 3.15.1.3.2 Standby Fuel Supply

The standby diesel generator has a 175-gal fuel-oil day tank plus an underground bulk storage supply tank and uses diesel oil Specification Number 2. The day tank is located within the diesel-generator building and supplies its engine-mounted fuel-oil pump. The day tank is automatically filled during engine operation from its separate underground storage tank located outside adjacent to the diesel-generator building. The storage tank has a capacity of 7700 gal and is provided with a motor-driven transfer pump mounted in a manhole opening above oil level. If a low level is detected in the day tank for the standby diesel generator, its transfer pump will automatically start to refill the tank to approximately 158 gal.

The diesel oil transfer pump stops automatically when 15.5-in. of oil remains in the underground tank which equates to a maximum of approximately 7000-gal of available fuel oil.

As previously mentioned in Section 3.15.1.1.1, commercial oil supplies and trucking facilities exist to ensure deliveries on one day's notice.

### 3.15.1.3.3 Standby Diesel Generator Location

The standby diesel generator is located in a sheet metal, steel-framed building immediately South of the Primary Auxiliary Building. The engine foundation is surrounded by a 1-foot-high concrete curb containing sufficient volume to hold all the lube-oil or fuel released from a single engine in the event of an inadvertent spill or line break.

Diesel generator fire protection features necessary to meet the criteria of 10 CFR 50.48(f) are described in the document under separate cover entitled, "IP2 Fire Hazards Analysis." A control panel, which contains relays and metering equipment for the standby diesel generator is located on the west end of the building. A reinforced-concrete wall separates the standby diesel generator from the control panel.



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### 3.15.1.3.4 Loading Description

The standby diesel-generator is manually started on the occurrence of an undervoltage on any 480-V switchgear bus.

On undervoltage on any bus, the engine is manually started and run at idle and can be connected to deenergized buses by site personnel from the control room or locally.

### 3.15.1.3.5 Batteries and Battery Chargers

The battery installation is composed of individual lead-calcium storage cells connected to provide a nominal terminal voltage of 125-V DC. The battery is fed from a charger that is fed from a 480-V motor control center. The battery bus is equipped with a sensitive-type undervoltage relay, which provides alarm / indication of an undervoltage condition. Ground alarms are also provided on the charger's board. Improved status indication of the battery charger and the direct current system has been provided by a DC bus trouble alarm. Loads on the DC system are shown on Drawings 208501 and 9321-3008 [Formerly Figures 3.15-15 and 3.15-16].

### 3.15.1.3.6 Reliability Assurance

The 480-V equipment is arranged on four buses. The 6.9-kV equipment is supplied from six buses.

The outside source of power is adequate to run all normal operating equipment. The 13.8-kV / 6.9-kV autotransformer can supply all the auxiliary loads.

A diesel generator (SBO / Appendix R diesel generator or standby diesel generator) has enough capacity to provide standby power to the facility.

A total loss of DC feed to the switchgear and associated equipment will not cause a loss of offsite power through an inadvertent tripping of the Indian Point Unit 2 light and power supply circuit breakers, because DC power is required to trip a breaker. Loss of DC feed to protective relaying will cause an alarm condition rather than initiation of a protective action. If necessary, the light and power circuit breakers in the Buchanan substation may be tripped manually at the breaker mechanisms.

The equipment arrangement in the Indian Point Unit 2 Central Control Room is discussed in Section 3.13.

## 3.15.2 TESTS AND INSPECTIONS

The fuel supply and starting circuits and controls are continuously monitored and any faults are alarm indicated. An abnormal condition in these systems would be signaled without having to place the standby diesel generator on test. The standby diesel generator will be inspected in accordance with a licensee-controlled maintenance program. The maintenance program will require inspection in accordance with the manufacturer's recommendation for this class of standby service.

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The SBO / Appendix R diesel and support systems shall be tested and have surveillances in accordance with the TRM. These tests and surveillances are designed to assure that the SBO / Appendix R diesel will be available to provide power for operation of equipment, if required.

### 3.15 FIGURES

Figure No.	Title
Figure 3.15-1	Electrical One-Line Diagram, Replaced with Drawing 250907
Figure 3.15-2	Electrical Power System Diagram, Replaced with Drawing 250907
Figure 3.15-3	Main One-Line Diagram, Replaced with Drawing 208377
Figure 3.15-4	345-KV Installation at Buchanan
Figure 3.15-5	6900-V One-Line Diagram, Replaced with Drawing 231592
Figure 3.15-6	480-V One-Line Diagram, Replaced with Drawing 208088
Figure 3.15-7	Single Line Diagram 480-V Motor Control Centers 21, 22, 23,25, 25A, Replaced with Drawing 9321-3004
Figure 3.15-7a	Single Line Diagram - 480-V Motor Control Centers 24 and 24A, Replaced with Drawing 249956
Figure 3.15-8	Single Line Diagram - 480-V Motor Control Centers 27 and 27A, Replaced with Drawing 9321-3005
Figure 3.15-9	Single Line Diagram - 480-V Motor Control Centers 28 and 210, Replaced with Drawing 208507
Figure 3.15-9a	Single Line Diagram - 480-V Motor Control Centers 29 and 29A, Replaced with Drawing 249955
Figure 3.15-10	Single Line Diagram - 480-V Motor Control Centers 28A and 211, Replaced with Drawing 208241
Figure 3.15-11	Single Line Diagram - 480-V Motor Control Centers 26A and 26B, Replaced with Drawing 9321-3006
Figure 3.15-11a	Single Line Diagram - 480-V Motor Control Center 26C, Replaced with Drawing 248513
Figure 3.15-12	Single Line Diagram - 480-V Motor Control Centers 26AA and 26BB and 120-V AC Panels No. 1 and 2, Replaced with Drawing 208500
Figure 3.15-13	Single Line Diagram - 118-VAC Instrument Buses No. 21 thru 24, Replaced with Drawing 208502
Figure 3.15-14	Single Line Diagram - 118-VAC Instrument Buses No. 21A thru 24A, Replaced with Drawing 208503
Figure 3.15-15	Single Line Diagram - DC System Distribution Panels No. 21, 21A, 21B, 22, and 22A, Replaced with Drawing 208501
Figure 3.15-16	Single Line Diagram - DC System Power Panels No. 21 thru 24, Replaced with Drawing 9321-3008
Figure 3.15-17	Single Line Diagram of Unit Safeguard Channeling and Control Train Development, Replaced with Drawing 208376
Figure 3.15-18	Cable Tray Separations, Functions, and Routing, Replaced with Drawing 208761



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### 3.16 CONTAINMENT STRUCTURES

#### 3.16.1 Design Basis

Historically, the reactor containment completely enclosed the entire reactor and reactor coolant system and ensured that essentially no leakage of radioactive materials to the environment would result even if gross failure of the reactor coolant system were to occur. The liner and penetrations were designed to prevent any leakage through the containment. The structure provided biological shielding for both normal and accident situations. In the permanently shut down and defueled condition, the reactor containment performs no active function. However, it must remain capable of withstanding natural phenomenon, so that it does not damage any Class I SSC.

The reactor containment is designed to safely withstand several conditions of loading and their credible combinations. The major loading conditions are an earthquake or wind.

##### 3.16.1.1 Principal Design Criteria

###### 3.16.1.1.1 Quality Standards

Criterion: Those systems and components of reactor facilities, which are essential to the prevention, or the mitigation of the consequences, of nuclear accidents, which could cause undue risk to the health and safety of the public shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes and standards pertaining to design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance criteria to be used shall be identified. An indication of the applicability of codes, standards, quality assurance programs, test procedures and inspection acceptance criteria used is required. Where such items are not covered by applicable codes and standards, a showing of adequacy is required. (GDC 1)

The containment system structure is of primary importance with respect to its safety function by maintaining its structural integrity in the event of a natural phenomenon event, so it does not fail and cause damage to Class I SSCs.

Quality standards of material selection, design, fabrication, and inspection governing the above features conforms to the applicable provisions of recognized codes and good nuclear practice. The concrete structure of the reactor containment conforms to the applicable portions of ACI-318-63. Further elaboration on quality standards of the reactor containment is given in Section 3.16.1.3.

###### 3.16.1.1.2 Performance Standards

Criterion: Those systems and components of reactor facilities, which are essential to the prevention or to the mitigation of the consequences of nuclear accidents, which cause undue risk to the health and safety of the public shall be designed, fabricated, and erected to performance standards that enable such systems and

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components to withstand, without undue risk to the health and safety of the public, the forces that might reasonably be imposed by the occurrence of an extraordinary natural phenomenon such as earthquake, tornado, flooding condition, high wind or heavy ice. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been officially recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design. (GDC 2)

All components and supporting structures of the reactor containment are designed so that there is no loss of function of such equipment in the event of maximum potential ground acceleration acting in the horizontal and vertical directions simultaneously. The dynamic response of the structure to ground acceleration, based on the site characteristics and on the structural damping, is included in the design analysis. Historically, the reactor containment was defined as a Class I structure for purposes of seismic design. Following the permanent shut down and defueling of the reactor, it was re-classified as a Class III structure (Section 1.7). Its structural members have sufficient capacity to accept, without exceeding specified stress limits, a combination of normal operating loads, functional loads due to a loss-of-coolant accident, and the loadings imposed by the maximum potential earthquake.

### 3.16.1.1.3 Fire Protection

Criterion: A reactor facility shall be designed to ensure that the probability of events such as fires and explosions and the potential consequences of such events will not result in undue risk to the health and safety of the public. Noncombustible and fire resistant materials shall be used throughout the facility wherever necessary to preclude such risk, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features. (GDC 3)

Fire protection in all areas of the nuclear electric plant is provided by structure and component design that optimizes the containment of combustible materials and maintains exposed combustible material below the ignition temperature. The station is designed on the basis of limiting the use of combustible materials in construction by using fire-resistant materials to the greatest extent practical.

### 3.16.1.1.4 Records Requirement

Criterion: The reactor licensee shall be responsible for assuring the maintenance throughout the life of the reactor of records of the design, fabrication, and construction of major components of the plant essential to avoid undue risk to the health and safety of the public. (GDC 5)

Records of the design, fabrication, construction, and testing of the reactor containment are maintained throughout the life of the reactor, as modified in accordance with approved exemptions.

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### 3.16.1.1.5 Reactor Containment

Criterion: The containment structure shall be designed (a) to sustain, without undue risk to the health and safety of the public, the initial effects of gross equipment failures, such as a large reactor coolant pipe break, without loss of required integrity, and (b) together with other engineered safety features as may be necessary, to retain for as long as the situation requires, the functional capability of the containment to the extent necessary to avoid undue risk to the health and safety of the public. (GDC 10).

The containment structure and all penetrations are designed to withstand, within design limits, the combined loadings of the historical design-basis accident associated with containment performance and design and maximum potential seismic conditions.

### 3.16.1.2 Loadings

The following loadings are considered in the design of the containment:

1. Structure dead load.
2. Live loads.
3. Equipment loads.
4. Internal test pressure.
5. Earthquake.
6. Wind.

### 3.16.1.3 Codes and Standards

The following is historical information. The design, materials, fabrication, inspection, and proof testing of the containment vessel complies with the applicable parts of the following codes and standards.

Code	Title
1. ASTM A-333, Gr. 1	Specification for Seamless and Welded Steel Pipe for Low Temperature Service
2. ASTM A-181	Forged or Rolled Steel Pipe Flanges, Forged Fittings, and Valves and Parts for General Service
3. ASTM A-300, Cl. 1, Firebox	Specification for Notch Toughness Requirements for Normalized Steel Plates for Pressure Vessels
4. ASTM A-201, Gr. B	Specification for Carbon Silicon Steel Plates of Intermediate Tensile Ranges for Fusion Welded Boilers and other Pressure Vessels
5. ASTM A-36	Specification for Structural Steel
6. ASTM A-131, Gr. C	Specification for Structural Steel for Ships
7. ASTM A-240	Specification for Heat-Resisting Chromium and Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Fusion-Welded Unfired Pressure Vessels

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Code	Title
8. ASTM A-312	Specification for Seamless and Welded Austenitic Stainless Steel Pipe
9. ASTM A442, Grade 60	Specifications for Pressure Vessel Plates, Carbon Steel, Improved Transition Properties
10. ASME Boiler and Pressure Nuclear Vessels Vessel Code-Section III	Nuclear Vessels
11. ASME Boiler and Pressure Unfired Pressure Vessels Vessel Code-Section VIII	Unfired Pressure Vessels
12. ASME Boiler and Pressure Welding Qualifications Vessel Code-Section IX	Welding Qualifications
13. ASTM C-33	Standard Specifications for Concrete Aggregates
14. ASTM C-150	Standard Specifications for Portland Cement
15. ASTM C-172	Standard Method of Sampling Fresh Concrete
16. ASTM C-31	Standard Method of Making and Curing Concrete Compression and Flexure Test Specimens in the Field
17. ASTM C-39	Standard Method of Test for Compressive Strength of Molded Concrete Cylinders
18. ASTM-C-350	Specifications for Fly Ash for Use as an Admixture in Portland Cement Concrete
19. ASTM C-94	Specifications for Ready Mixed Concrete
20. ASTM C-42	Standard Methods of Securing, Preparing, and Testing Specimens from Hardened Concrete for Compressive and Flexural Strengths
21. ASTM C-494	Specifications for Chemical Admixtures for Concrete
22. ASTM A-305	Specifications for Minimum Requirements for Deformations of Deformed Steel Bars for Concrete Reinforcement
23. ASTM A-408	Specifications for Special Large Size Deformed Billet-Steel Bars for Concrete Reinforcement
24. ASTM A-432	Specification for Deformed Billet Steel Bars for Concrete Reinforcement with 60,000 psi Minimum Yield Strength
25. Research Council of Riveted and Bolted Structural Joints of the Engineering Foundation	Specification for Structural Joints Using ASTM A-325 Bolts
26. ACI-613	Recommended Practice for Selecting Proportions for Concrete
27. ACI-306	Recommended Practice for Winter Concreting

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Code	Title
28. ACI-318, Part IV-B	Structural Analysis and Proportioning of Members-Ultimate Strength Design
29. ACI-318	Building Code Requirements for Reinforced Concrete
30. ACI- 505	Specification for the Design and Construction of Reinforced Concrete Chimneys
31. ACI-315	Manual of Standard Practice for Detailing Reinforced Concrete Structures
32. ASA N6.2	Safety Standards for the Design, Fabrication and Maintenance of Steel Containment Structures for Stationary Nuclear Power Reactors
33. ASA A58.1	American Standard Code Requirements for Minimum Design Loads in Buildings and Other Structures
34.	State Building and Construction Code for the State of New York
35. SSPC-SP-6	Commercial Blast Cleaning

### 3.16.2 Containment Structure Design

#### 3.16.2.1 General Description

The reactor containment structure is a reinforced concrete vertical right cylinder with a flat base and hemispherical dome. A welded steel liner with a minimum thickness of 0.25-in. is attached to the inside face of the concrete shell to ensure a high degree of leaktightness. The design objective of the containment structure is to retain its structural integrity during normal conditions and natural phenomenon events.

The structure, as shown on Drawings 9321-2501, 9321-2502, 9321-2503, 9321-2506, 9321-2507, 9321-2508, and Figure 3.16-1 consists of side walls measuring 148-ft from the liner on the base to the springline of the dome, and has an inside diameter of 135-ft. The side walls for the cylinder and the dome are 4-ft 6-in. and 3-ft 6-in. thick respectively. The inside radius of the dome is equal to the inside radius of the cylinder so that the discontinuity at the springline due to the change in thickness is on the outer surface. The cylindrical part of the liner is substantially round. The difference between the minimum and maximum inside diameters at any selected cross section does not generally exceed 0.25-percent of the nominal diameter at that elevation. Between elevations 43-ft and 95-ft, the maximum diameter of any cross section is 135-ft 2-in., and the minimum diameter is 134-ft 10-in. except at the liner closing the temporary opening in the northwest quadrant where a minimum diameter of 134-ft 8-5/8-in. was measured. This portion of the liner was erected after all exterior concrete work was completed and is within the local buckle allowance of the liner plates. Above elevation 95 ft the tolerance on inside diameter does not exceed 0.50-percent of the nominal diameter of the selected cross section. The liner is erected true and plumb so that the deviation does not exceed 1/500 of the height at the selected cross section (allowing for 2-in. local buckling of the liner plates).

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Particular care is taken in matching edges of cylindrical and hemispherical sections to ensure that all joints are properly aligned. Maximum permissible offset of completed joints is 25 percent of nominal plate thickness. Plates buckled beyond acceptable limits are cut out and replaced with new plates.

The flat concrete base mat is 9-ft thick with the bottom liner plate located on top of this mat. The bottom liner plate is covered with 3-ft of concrete, the top of which forms the floor of the containment.

Where uplift from pressure occurs at the outer areas of the mat, the 9-ft thick mat has sufficient flexural capacity to resist the uplift.

No hydraulic uplift exists since the bottom elevation of the mat is considerably higher than that of the high water level.

The large mass of the containment including interior concrete and equipment makes the structure inherently stable from overturning due to seismic motion.

In addition, keying action from the reactor pit and sumps, plus friction between the concrete and rock, prevents a sliding of the structure from horizontal ground motion.

The basic structural elements considered in the design of the containment structure are the base slab, side walls, and dome acting as one structure under all possible loading conditions. The liner is anchored to the concrete shell by means of stud anchors. The lower portions of the cylindrical liner are insulated to avoid thermal deformation of the liner under accident conditions.

The containment structure is inherently safe with regard to common hazards such as fire, flood, and electrical storm. The thick concrete walls are invulnerable to fire and only an insignificant amount of combustible material, such as lubricating oil in pump and motor bearings, is present in the containment.

Internal structures consisted of equipment supports, shielding, reactor cavity and canal for fuel transfer, and miscellaneous concrete and steel for floors and stairs. All internal structures are supported on the mat with the exception of equipment supports secured to the intermediate floors.

A 3-ft thick concrete ring wall serving as a missile and partial radiation shield surrounds the reactor coolant system components and supports the polar-type reactor containment crane. A 2-ft thick reinforced concrete floor covers the reactor coolant system with removable gratings in the floor provided for crane access to the reactor coolant pumps. The four steam generators, pressurizer, and various piping penetrate the floor. Spiral stairs provide access to the areas below the floor.

The refueling canal connects the reactor cavity with the fuel transport tube to the spent fuel pool. The floor and walls of the canal are concrete, with wall and shielding water providing the equivalent of 6-ft of concrete.

The refueling canal floor is 5-ft thick. The concrete walls and floor are lined with 0.25-in. thick stainless steel plate. The linings provide a leakproof membrane that is resistant to abrasion and damage during fuel handling operation.

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Waterproofing is provided in the areas of the containment in contact with backfill to prevent ground-water seepage. This consists of a coat of bitumastic No. 50, a 0.625-in.-thick layer of hardboard insulation, and a second coat of bitumastic No. 50. Fill for innermost 5-ft from containment walls is crushed rock of maximum size of 6-in. and minimum amount of fines. All fill is free of vegetable matter.

### 3.16.2.2 Design Load Criteria

The following loads are considered to act upon the containment structure creating stresses within the component parts.

1. Dead load consists of the weight of the concrete wall, dome, liner, insulation, base slab, and the internal concrete. Weights used for dead load calculations are as follows:
  - a. Concrete 150 lb/ft<sup>3</sup>
  - b. Reinforcing steel 490 lb/ft<sup>3</sup> using nominal cross-sectional areas of reinforcing as defined in ASTM for bar sizes.
  - c. Steel lining 490 lb/ft<sup>3</sup> using nominal cross-sectional area.
  - d. Insulation 6 lb/ft<sup>3</sup> including stainless steel jacket.
2. Live load consists of snow and construction loads on the dome and major components of equipment in the containment. Snow and ice loads are assumed to be applied uniformly to the top surface of the dome at an estimated value of 20 lb/ft<sup>2</sup> of horizontal projection of the dome. This loading represents approximately 2-ft of snow, which is considered to be a conservative amount since the slope of the dome will tend to cause much of the snow that falls on it to slide off. A construction live load of 50 lb/ft<sup>2</sup> has been used on the dome, but will not be considered to act concurrently with the snow load. Equipment loads are considered as specified on the drawings supplied by the manufacturers of the various pieces of equipment.

Design live loads inside the containment building are as follows:

- a. Elevation 68-ft-0-in. 10-ft strip adjacent to crane wall = 600 psf  
Remaining strip = 100 psf
- b. Elevation 95-ft-0-in. Concrete slab = 500 psf Grating areas = 100 psf
3. For the free volume of 2,610,000-ft<sup>3</sup> within the containment, the design pressure is 47 psig.
4. The ground acceleration for the design earthquake has been determined to be 0.1g applied horizontally and 0.05g applied vertically. These values have been resolved as conservative numbers based upon recommendations from Dr. Lynch, Director of Seismic Observatory, Fordham University.

A dynamic analysis is used to arrive at equivalent design loads. Additionally, a hypothetical ground acceleration of 0.15 g horizontal and 0.10 g vertical is used to



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analyze for the no-loss-of-function. This is discussed in Section 3.16.3.11, Seismic Design.

Due to symmetry of the containment structure, torsional loads generated by an earthquake are insignificant and have not been considered.

Tornado loads have not been considered in the design of the Unit 2 containment; however, the seismic bars provide a more than adequate mechanism to withstand the torsional effect if it were to occur. An evaluation of the effect of tornado loads on the containment structure is presented in Appendix B of the Containment Design Report.

5. The American Standards Association "American Standard Code Requirements for Minimum Design Loads in Buildings and Other Structures" (A58.1-1955) designates the site as being in a 25 psf zone for wind loads. In this code, for height zones between 100 and 499-ft, the recommended wind pressure on a flat surface is 40 psf. Correcting for the shape of the containment by using a shape factor of 0.60, the recommended pressure becomes 24 psf. The state building and construction code for the State of New York stipulates a wind pressure up to 30 psf on a flat surface for heights up to 300 feet. For design, a 30 psf basic wind load has been used from ground level up.
6. Internal pressure was applied to test the structural integrity of the containment shell up to 115-percent of the design pressure. For this structure, the test pressure is 54 psig. The containment is also structurally designed to withstand an external pressure 2.5 psig higher than the internal pressure.

### 3.16.2.3 Material Specifications

Basically, five materials are used for the construction of the containment structure.

These are:

1. Concrete.
2. Reinforcing steel.
3. Plate steel liner.
4. Insulation.
5. Protective Coating.

Basic specifications for these materials are as follows:

1. Concrete is a dense, durable mixture of sound coarse aggregate, fine aggregate, cement, and water. Cement conforms to ASTM, Specification C-150-65 "Standard Specification for Portland Cement," Type I (Normal), or Type II (moderate heat of hydration) requirements. Whenever high early strength is required, Type III Cement is used. Water is free from any injurious amounts of acid, alkali, salts, oil, sediment, or organic matter. The concrete has a minimum density of 150 lb/ft<sup>3</sup>. The 28-day standard compressive strength of the concrete is 3000 psi. Adequate means of control are used in the manufacture of the concrete. To ensure the values of compressive strength are attained as a minimum, concrete samples are tested in accordance with the following ASTM Standards:



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ASTM C-172 - Standard Method of Sampling Fresh Concrete

ASTM C-31 - Standard Method of Making and Curing Concrete Compression and Flexure Test Specimens in Field

ASTM C-39 - Standard Method of Test for Compressive Strength of Molded Concrete Cylinders

All making and testing of concrete samples have been performed by Vacca Testing Laboratory and Research Company, Inc.

At certain specifically evaluated locations, non-structural surface type cracks and delaminations in the containment concrete have been repaired by injection of engineering approved epoxy grout. Although non-structural in nature, these repairs were performed in accordance with the requirements of IWL-4210 of the 1992 ASME Boiler and Pressure Vessel Code, Section XI, as applicable.

2. Reinforcing steel for the dome, cylindrical walls and base mat is high-strength, deformed billet steel bars conforming to ASTM Designation A432-65 "Specification for Deformed Billet Steel Bars for Concrete Reinforcement with 60,000 psi Minimum Yield Strength." This steel has a minimum yield strength of 60,000 psi, a minimum tensile strength of 90,000 psi, and a minimum elongation of 7-percent in an 8-in. specimen. Reinforcing bars No. 11 and smaller in diameter are lapped spliced in the mat for flexural loadings and spliced by the Cadweld process in the walls and dome for tension loading. Bars No. 14S and 18S are spliced by the Cadweld process only. A certification of physical properties and chemical content of each heat of reinforcing steel delivered to the job site has been issued from the steel supplier. The splices used to join reinforcing bars have been tested to ensure that they will develop at least 125-percent of the minimum yield point stress of the bar. The test program required cutting out, at random, approximately 3-percent, completed splices and testing to determine their breaking strength.
3. The plate steel liner is carbon steel conforming to ASTM Designation A442-65 "Standard Specification for Carbon Steel Plates with Improved Transition Properties," Grade 60. This steel has a minimum yield strength of 32,000 psi and a minimum tensile strength of 60,000 psi with an elongation of 22-percent in an 8-in. gauge length at failure.

The liner is 0.25-in. thick at the bottom, 0.50-in. thick in the first three courses, except 0.75-in. thick at penetrations, a minimum of 0.34-in. in the general area at elevation 46-ft. due to past corrosion, and 0.375-in. thick for remaining portion of the cylindrical walls and 0.50-in. thick in the dome. The 0.34-in. minimum thickness affects the calculated stress levels presented in the Containment Design Report and the Containment Liner Stress Analysis Report. However, evaluation of the reduced minimum thickness has concluded that no design criteria are exceeded. The liner material has been tested to ensure an NDTT more than 30°F lower than the minimum operating temperature of the liner material.

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Impact testing has been done in accordance with Section N331 of Section III of the ASME Boiler and Pressure Vessel Code. A 100-percent visual inspection of liner anchors was made prior to pouring concrete.

4. The material used for the original insulation of the liner plate was polyvinylchloride with stainless steel jacket. The carbon steel liner with an inorganic zinc protective coating makes contact with the polyvinylchloride insulation, the stainless steel, and the sealant. However, these materials do not react with each other.

Manufacturer's tests on the polyvinylchloride insulation indicated that the insulation was capable of withstanding periodic compression at 60 psig at temperatures from 40°F to 120°F and a single compression under accident conditions without any detriment or change to the insulation properties. The manufacturer's analog transient analysis indicated only a 5°F rise in liner temperature 1000 sec after an exposure to 310°F for the entire duration of the analysis. This provides a factor of safety of approximately 15 on specified tolerable temperature rise in the liner. A factor of safety of 2 is provided on specified insulation performance versus tolerable temperature rise in liner.

5. One 3 mil shop coat of Carbozinc No. 11 primer and one 4 mil minimum finish coat of Phenoline No. 305 as manufactured by the Carboline Company have been applied to the liner, as well as essentially all painted surfaces in containment, in accordance with the manufacturer's recommendations.

The effect of the historical post-accident environment on protective coatings was conservatively evaluated for Indian Point Unit 2. The coatings showed no deterioration after a number of cycles. A more thorough discussion on the qualifications of the protective coatings applied during construction is presented in WCAP-7198-L.<sup>1</sup>

In addition, various areas inside containment have been repaired and recoated with other DBA qualified coatings approved for use at Indian Point 2. Protective coatings used inside the containment are procured, applied, and maintained in compliance with Regulatory Guide 1.54 (June 1973), "Quality Assurance Requirements for Protective Coatings Applied to Water Cooled Nuclear Power Plants." New quality requirements will be developed based on its provisions, but specific requirements, such as documented site meetings, field demonstrations, substrate priming, applicator reporting, inspection reporting and report forms will be considered on a job-by-job basis.

Quality of both materials and construction of the containment structure was ensured by a continuous program of quality control and inspection by Con Edison, and/or its field representatives, and Westinghouse Atomic Power Division, and United Engineers and Constructors Inc., as described in Section 3.16.2.5.

### 3.16.2.4 Design Stress Criteria

This analysis, including the consideration of accident pressure loads, is retained, because it is conservative with respect to the containment structure's function in the permanently shut down and defueled condition.

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The design is based upon limiting load factors that are used as the ratio by which loads will be multiplied for design purposes to ensure that the loading deformation behavior of the structure is one of elastic, tolerable strain behavior. The load factor approach is being used in this design as a means of making a rational evaluation of the isolated factors, which must be considered in ensuring an adequate safety margin for the structure. This approach permits the designer to place the greatest conservatism on those loads most subject to variation and which most directly control the overall safety of the structure. In the case of the containment structure, therefore, this approach places minimum emphasis on the fixed gravity loads and maximum emphasis on accident and earthquake or wind loads. The loads utilized to determine the required limiting capacity of any structural element on the containment structure are computed as follows:

1.  $C = 1.0D \pm 0.05D + 1.5 P + 1.0 (T + TL)$
2.  $C = 1.0D \pm 0.05D + 1.25 P + 1.0 (T' + TL') + 1.25E$
3.  $C = 1.0D \pm 0.05D + 1.0P + 1.0 (T'' + TL'') + 1.0E'$

Symbols used in these formulae are defined as follows:

C	=	Required load capacity of section.
D	=	Dead load of structure and equipment loads.
P	=	Historical Accident pressure load as shown on historical pressure-temperature transient curves.
T	=	Load due to maximum temperature gradient through the concrete shell and mat based upon temperature associated with 1.5 times historical accident pressure.
TL	=	Load exerted by the liner based upon temperatures associated with 1.5 times historical accident pressure.
T'	=	Load due to maximum temperature gradient through the concrete shell and mat based upon temperatures associated with 1.25 times historical accident pressure.
TL'	=	Load exerted by the liner based upon temperatures associated with 1.25 times historical accident pressure.
E	=	Load resulting from either design earthquake or wind, whichever is greater.
T''	=	Load due to maximum temperature gradient through the concrete shell and mat based upon temperatures associated with the historical accident pressure.
TL''	=	Load exerted by the liner based upon temperatures associated with the historical accident pressure.
E'	=	Load resulting from assumed hypothetical earthquake.

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A chart for allowable versus actual stresses has been included in the Containment Design Report.

Load condition (1) indicates that the containment will have the capacity to withstand loadings at least 50-percent greater than those calculated for the historical postulated loss-of-coolant accident alone. Results of analysis using load condition (1) are shown in Figure 3.16-2.

Load condition (2) indicates that the containment will have the capacity to withstand loadings at least 25-percent greater than those calculated for the historical postulated loss-of-coolant accident with a coincident design earthquake. Results of analysis using load condition (2) are shown in Figure 3.16-3.

Load condition (3) indicates that the containment will have the capacity to withstand loads at least equal to those calculated for the historical postulated loss-of-coolant accident with a coincident hypothetical earthquake defined in Section 3.16.2.2. Results of analysis using load condition (3) are shown in Figure 3.16-4.

The mat has been analyzed using load conditions (1), (2) and (3) as shown in Figures 3.16-5 through 3.16-7 and also for loads occurring only at operating and test pressure conditions. For loads, see Table 3.16-1, Flooded Weights-Containment Building.

The loads resulting from wind on any portion of the structure do not exceed those resulting from earthquake.

The capacity of all structural components, with the minor exceptions of outer rebar at large containment openings addressed in Section 3.4.4 of the Containment Design Report, exceeds or is equal to the capacity required by the most severe loading combination. The loads resulting from the use of these equations will hereafter be termed "factored loads."

The load factors used in these equations are based upon the load factor concept employed in Part IV-B, "Structural Analysis and Proportioning of Members Ultimate Strength Design" of ACI-318-63. Because of the refinement of the analysis and the restrictions on construction procedure, the load factors in the design primarily provide for a safety margin on the load assumptions.

The design includes the consideration of both primary and secondary stresses. The design limit for tension member (i.e., the capacity required for the design load) is based upon the yield stress of the reinforcing steel.

The theoretical load carrying capacity of steel reinforced concrete cross-sections are reduced by a capacity reduction factor " $\phi$ ", which provides for the possibility that small adverse variations in material strengths, workmanship, dimensions, and control, while individually within required tolerances and the limits of good practice, occasionally may combine to result in under-capacity. For tension members, the factor " $\phi$ " has been established as 0.95. The factor " $\phi$ " is 0.90 for flexure and 0.85 for diagonal tension, bond, and anchorage.

For principle compression and tension, the liner stresses are maintained below 0.95 specified minimum yield at normal operating temperature (i.e.,  $\phi=0.95$ ). For shear, the liner stresses are maintained below 0.6 yield.

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The liner is designed to assure that no strains greater than the strain at the guaranteed yield point will occur at the factored loads. In regions of local stress concentrations or stresses due to localized secondary load effects, the liner is permitted to yield but the maximum liner strain is limited to 0.5-percent. Sufficient anchorage is provided to ensure elastic stability of the liner. The basic design concept for the liner stud anchorage is the ductility of the anchorage that assures stud failure due to shear, tension or bending stress without the stud connection causing failure or tear of the liner plate. References 3.16-2 and 3.16-3 provide information on design of stud connection. The studs in the 0.50-in. plate are installed on 24-in. horizontal and 28-in. vertical grid and in the 0.375-in. plate on a 24-in. horizontal and 14-in. vertical grid. Studs are centered between vertical bars. In the dome, 5-ft by 5-ft panels are anchored in the center by studs and by T-bars at the edges. The 0.50-in. diameter bent welding studs are 9-in. long minimum and 9.50-in. long maximum with a 2-in. 90-degree hook at the end. An arc stud welding process was used on all bent welding studs. The arc stud welding process produces a circular weld around the 0.50-in. diameter stud with a diameter (outside to outside of weld) equal to 0.678-in. and a height equal to 0.157-in. The design considers the possibility of daily stress reversals due to ambient temperature changes for the life of the plant, and fatigue limit of the studs exceeds the design requirements. However, to accommodate possible fatigue failure in the plate-to-stud weldment, the depth of penetration to the liner plate is controlled to avoid impairment of liner integrity.

The boundary conditions in the cylinder are determined by assuming a buckling model (shown in Figures 3.16-8 through 3.16-10) in which the studs form the low points and the center of the panels form the high points of a series of peaks and valleys thus forming a set of panels whose edges represent points of inflection. The analytical procedure used is a simply supported plate under biaxial compression. A Mohr's circle analysis is used to find the normal and shear stresses on this simply-supported plate. The critical buckling stress is derived considering a plate whose length is equal to one-half of the diagonal distance between studs. This critical buckling load is 38.1 ksi for the 0.375-in. liner and 38.4 ksi for the 0.50-in. liner, which is higher than the yield strength of the liner, 32 ksi; therefore, the liner plate will begin to yield before the critical buckling stress is reached, and buckling failure does not control the design. Since shear reduces the stability of a plate subjected to compressive stresses, critical shear is considered and it was found that critical buckling is controlled by normal stresses rather than shear stresses. This is determined by considering the magnitude of both the normal and the shear stresses on the panel. The magnitude of the shear is so low that it shows no effect on the previously stated critical buckling stresses.

In the dome the liner will be considered clamped at the stiffeners forming a 5-ft by 5-ft grid panel pattern. The center of each panel is fixed by a stud. Assuming points of inflection at the one-quarter point a distance of 1-ft 3-in. occurs between points of simple support. The critical buckling load is 58.1 ksi, which is also higher than the yield strength of the liner.

At maximum strain in the liner, the studs will not fail. This maximum strain due to an unbalanced load would occur in a panel adjacent to a buckled panel. Since this adjacent stud will not fail, no zipper effect will occur and massive buckling of the liner and mass failure of anchors is not credible.

The anchorages can fail by failure of the studs in shear or tension, by studs pulling out from the concrete, or by studs separating from the liner plate. The most likely mode of failure is by tensile failure of the stud. The anchors are designed so that failure occurs in the anchor rather than the plate, thereby ensuring that the leaktight integrity of the containment liner will be maintained.

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If failure should develop, it would be a random stud failure due to poor workmanship during stud attachment. This failure would not impair the liner integrity nor would it cause progressive failure.

The anchor must resist tensile and shearing loads. Tests have indicated that the lateral load needed to prevent column buckling is 1-percent of the axial yield load. Conservatively doubling this value to account for uncertain field conditions, a value of 2-percent is used.<sup>4</sup> The total load per plate would be 24-in. x 0.50-in. x 32,000 psi = 384,000 lb. Therefore, the tensile load per anchor is  $384,000 \text{ lb} \times 0.02 = 7680 \text{ lb}$ , which yields a stress of  $7680/0.2 = 38,400 \text{ psi}$ .

This compares with a yield value of 50,000 psi and a tensile strength of 60,000 psi in the studs. This does not consider the internal pressure, which provides further stability against buckling.

The shear load on the anchor is due to the strain in the liner. Assuming the liner approaches its yield strain of 0.1-percent, the anchor deflection would be 28-in. x .001 = .028-in. Tests on the stud anchor have shown a maximum deflection of about 0.1-in. can be tolerated before failure of the stud.

### 3.16.2.5 Quality Control

This section is historical, and is retained for information only.

To ensure a high degree of confidence in plant design, construction, workmanship, materials, and performance, a quality control program has been in effect for this project in which the following principal organizations have their respective responsibilities:

1. Consolidated Edison Company of New York, Inc. as initial owner and operator of the plant.
2. Westinghouse Electric Corporation as the turnkey plant contractor and supplier of major equipment.
3. United Engineers and Constructors Inc. as architect-engineer, construction manager, and constructor.

The function and responsibility in the quality control program of each of the above organizations is as follows:

#### 3.16.2.5.1 Consolidated Edison Company of New York, Inc. (Con Edison) – Initial Licensee

A qualified field representative was assigned to the field during the construction period. His responsibilities included continuous inspection of the construction of the containment building to ensure that all materials used and work performed was strictly in accordance with the plans and specifications. The Con Edison representative, through instructions received from the home office, had the power to stop the construction until any discrepancies were corrected and the work once more was in compliance with the specifications and plans.

The Con Edison representative was in constant communication and consultation with the construction superintendent in matters regarding quality control. In addition, personnel from U.S.



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Testing Laboratories were assigned to this project to monitor the inspection of the construction and obtain samples of the materials for testing.

### 3.16.2.5.2 Westinghouse Electric Corporation

For the assurance of plant integrity and quality, Westinghouse performed the following functions regarding the containment building:

1. Reviewed and approved the containment design criteria, material specifications and detail design concepts before they were released for construction. This work was done by qualified structural engineers at the company's home office.
2. Reviewed the construction and inspection methods employed by United Engineers and Constructors Inc.

Westinghouse Pressurized Water Reactor Division, Nuclear Power Services Group had a field quality assurance representative in residence during the construction period. His function was the same as the Con Edison representative mentioned above. He reported discrepancies to the Westinghouse Construction and Services resident engineer who had the authority to stop the work until the discrepancy was resolved.

In addition to this, he audited the construction files, and verified that records were complete, accurate, and adequate for quality assurance.

Nuclear Power Service Headquarters quality assurance engineers also made trips to the site to audit, monitor, and review the project with regard to site quality assurance. Construction practices were observed for conformance to codes, specifications, and approved procedures.

### 3.16.2.5.3 United Engineers and Constructors Inc.

The responsibilities of United Engineers and Constructors Inc. in the quality control of the containment building were as follows:

1. They inspected all materials delivered to the job site, and examined the suppliers' certified test reports of physical and chemical properties for those components furnished by them.
2. They inspected fabrication of major components of the containment structure in the shop. Trip reports are available at the site.
3. They maintained an adequate force of qualified supervisory personnel at all times.
4. They supervised and were fully responsible for the quality of work performed by their subcontractors and for the craft labor employed and supervised by them.
5. They maintained as part of their field engineering force, qualified personnel who performed a thorough inspection of each construction operation.

No changes in design or specifications were allowed without the approval of the engineer in charge of design.

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### 3.16.3 Containment Stress Analysis

#### 3.16.3.1 General

The structural design of the containment meets the requirements established by 1961 edition of "The State Building and Construction Code for the State of New York" so far as these provisions are applicable. All concrete structures have been designed, detailed, and constructed in accordance with the provisions of "Building Code Requirements for Reinforced Concrete" (ACI 318-63) so far as these provisions are applicable.

#### 3.16.3.2 Method of Analysis

Basically, three separate structural components have been analyzed, each in equilibrium with loads applied to it and with constraints occurring at the juncture of the structures. The three components are:

1. The 135-ft ID hemispherical dome.
2. The 135-ft ID cylinder.
3. The base slab.

Mathematically, the dome and cylinder have been treated as thin-walled shell structures, which results in a membrane analysis. Since the thickness of the dome and cylinder is small in comparison with the radius of curvature ( $1/20$  and  $1/15$ ) and there are no discontinuities such as sharp bends in the meridional curves, the stresses due to pressure and wind or earthquake are calculated by assuming that they are uniformly distributed across the thickness.

Since the concrete is not assumed to resist any tensile or shear forces, radial shear reinforcing has been introduced in the lower portion of the wall in the form of hooked diagonal stirrups and diagonally bent bars as shown in Figure 3.16-1. Diagonal shear reinforcing, at  $45^\circ$  and  $135^\circ$  to the circumferential direction, are placed in the center of the cylinder wall for the full height of the wall and a distance above the springline into the dome to resist earthquake shears. The diagonal bars are discontinued in the upper area of the dome (beyond about 30 degrees above the springline), where the seismic shears are small and are carried by the dome reinforcing steel lying in the plane of principal tension.

The base slab has been treated as a flat circular plate supported on a rigid nonyielding foundation.

The limiting cases in the design of the wall for discontinuity moments and shears were considered. One case considered an uncracked wall and the other considered a cracked wall with the steel acting as a spring constant. The value of  $\mu_c$  varied from zero in the cracked case to .14 in the uncracked case. In the uncracked case, variations in  $E_c$  will have no effect on the answer since  $E_c$  appears in both the numerator and the denominator of the stiffness formulation. For the above variation in  $E_c$  and  $\mu_c$ , the values of discontinuity moment and shear vary by 14-percent and 7-percent respectively at the base. These are the maximum deviations of the wall forces since the wall will actually vary from uncracked to cracked with an increase in containment height rather than be cracked or uncracked for the total height.



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In the area of thermal stress, the entire wall section will be cracked and no variation in  $E_c$  or  $\mu_c$  need be considered. The liner stresses depend on the strains of the reinforcing steel and are not related to the concrete properties.

Shrinkage and creep effects will be relieved by cracking during the pressure test and will not be included in accident design considerations.

The finite element computer program has the capabilities of taking into account variations in  $\mu_c$  and  $E_c$  and axisymmetric loads. However, it is not necessary to take into account the variations in  $\mu_c$  and  $E_c$  for the reasons stated above.

The computer program used to study the general behavior of the structure and to generate boundary conditions was the axisymmetric shell structure program. This computer program, developed by Franklin Institute Research Laboratories, is designed to handle arbitrarily shaped shells of revolution subjected to axisymmetric as well as nonaxisymmetric loadings. The method of analysis consists of subdividing the shell into elements having continuous meridians with continuous first and second derivatives so that the first and second fundamental forms of the resulting shell elements are continuous throughout the element. By expanding the dependent variables in Fourier series in the circumferential direction, and assigning unspecified functions for the meridional variation, the independent variables are separated and a system of ordinary differential equations results for the dependent variables in terms of the meridional independent variable. Particular and complementary solutions of these ordinary differential equations are then found for each of the elements and each of the circumferential harmonics individually. The matching of the elements is achieved by writing the required boundary conditions.

The idealized section used with the axisymmetric shell structure program consists of five layers whose moment of inertia is equal to that of the actual section. The wall section is considered as cracked with the reinforcing carrying all loads.

A finite element program, with the capability to incorporate thermal loads, was used to analyze the containment shell considering the effect of the equipment hatch opening.

The shell was idealized into 10 layers with alternate layers of steel and concrete. Section 3.16.3.10 provides more information on the finite element analysis.

The computer program can handle the loads in the form of either surface traction or edge loads or both.

Analysis of the liner is presented in the Containment Liner Stress Analysis Report. The report also contains a description of analytical procedures arriving at forces, shears, and moments in the structural shell.

### 3.16.3.3 Dome Analysis

The analysis of the hemispherical dome has been performed by the super-position of membrane forces resulting from gravity, historical accident pressure, and historical accident thermal loads. In addition, earthquake or wind loading create both direct and shear stresses in the dome, and the historical operating temperature of the liner creates tension and compression. All of the combined direct stresses are developed in the reinforcing steel encased in the concrete. In the

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upper area of the dome (about 30 degrees above the springline), where the seismic shears are small, seismic shears are carried by dome reinforcing steel lying in the plane of the principal tension. The dome reinforcing is spliced to the vertical steel in the cylindrical concrete wall, so that a continuity between the dome and the cylinder is realized. See Figure 3.16-11 for a section of wall, dome and for reinforcing in the dome.

### 3.16.3.4 Cylinder Analysis

The analysis of the cylinder is by superposition of membrane forces resulting from gravity, historical pressure and thermal loads, overturning due to earthquake or wind and shears due to earthquake or wind. The concrete has been reinforced circumferentially using steel hoops and vertically by straight bars. Diagonal bars have been placed to resist the horizontal and vertical shears due to earthquake or wind. The required capacity of the diagonal bars has been designed so that the horizontal component per foot of the diagonals is equal to the maximum value of shear flow. A check was made to ensure that no net compressive force results in the diagonal bars because of the combination of seismic shear load and internal pressure load. Although, in the cylinder, the liner has some capacity available to resist the seismic shears, no credit is taken for this capacity.

For all of the cylinder and the lower areas of the dome, the diagonal reinforcing has been designed to accommodate all seismic shears. No credit has been taken for the dowel action of the vertical and horizontal bars in resisting seismic shear.

Only in the upper area of the dome (beyond about 30 degrees above the spring line) where the seismic shears are small is the liner counted on to resist shear. For all of the cylinder and the lower areas of the dome, the diagonal reinforcing has been designed to accommodate all seismic shears. No credit has been taken for the dowel action of the vertical and horizontal bars in resisting seismic shear.

### 3.16.3.5 Base Mat Analysis

The base slab was treated as a flat circular plate supported on a rigid non-yielding foundation. For loads applied uniformly around the slab, the analysis considers a 1-ft wide beam fixed at a point where the vertical shear is equal to zero. This is the point where the downward pressure on the mat and the dead weight overcome the uplift at the containment wall base mat juncture from pressure and earthquake loadings. Radial and circumferential reinforcing is provided at the top and bottom of the mat to resist moments in the areas where uplift occurs. Temperature steel was added in other areas to meet requirements of the (ACI-318) Code. Diagonal tension reinforcement was added to meet requirements of ACI-318 Code. See Figure 3.16-14 for base slab reinforcing detail.

Moments and shears were calculated by writing equations for moment and shear in terms of X using the containment wall-base slab juncture as the origin with X increasing toward the center of the containment building. The point along the circumference of the containment wall chosen as the end of the beam is a point where the maximum tension from the earthquake will exist. Since the containment structure is considered a beam in all earthquake analyses, the maximum uplift for which the mat is designed will occur at only one point on the circumference and will represent the worst possible uplift on the mat.

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All stresses were calculated using Part IV-B Structural Analysis and Proportioning of Members - Ultimate Strength Design of the Building Codes Requirements for Reinforced Concrete (ACI-318-63). No rebar stresses exceed  $0.90 f_y$ .

A gradient with an operating temperature of 120°F inside the containment (historical condition) and a 50°F temperature at the mat-rock interface was considered and stresses were negligible. Ambient accident temperatures have no appreciable effect on the base slab. The maximum operating temperature of the containment is 130°F. The effect of elevated operating temperature on the structural elements was evaluated in 1987 and was found acceptable.

It is not possible to show that the design on nonyielding rock is more conservative than assuming the rock to be elastic. However, due to the installation of temperature reinforcing, the design is conservative. Reinforcing and concrete stresses are very low when considering the rock to be elastic.

To substantiate the above statement, the following studies were performed:

1. The foundation modulus were determined using the expression:<sup>15</sup>

$$k_z = \frac{4Gr_o}{1 - \mu}$$

where:

$k_z$  = The vertical spring constant of a circular base supported on an elastic foundation

$$G = \frac{E}{2(1 + \mu)}$$

$r_o$  = Radius of Foundation

$\mu$  = Poisson's Ratio

To obtain the foundation modulus,  $k_z$  is divided by the area of the circular base to yield

$$k_o = \frac{k_z}{A} \times \frac{4G}{\pi r_o(1 - \mu)}$$

Substituting for G

$$k_o = \frac{2E}{\pi r_o(1 - \mu^2)}$$

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2. The first case examined was that of a rectangular strip loaded with 1.5 times historical design accident pressure plus dead load using conservative properties for the Dolomitic limestone:<sup>7,14</sup>

$$E = 6.0 \times 10^6 \text{ psi}$$

$$\mu = 0$$

Applying these values

$$k_o = 4370 \text{ lbs/in.}^3$$

The "characteristic"  $\lambda$  is defined as:<sup>6</sup>

$$\lambda = \left[ \frac{k}{4 EI} \right]^{1/4}$$

Where:

$E$  is the modulus of elasticity of the structural base (concrete),

$I$  is the moment of inertia of the structural base,

$$k = k_o b, \text{ (} b = \text{width of base)}$$

using base properties

$$\lambda = 7.56 \times 10^{-3} \text{-in.}^{-1}$$

Where  $\lambda \ell > \pi$  beams may be considered as infinite in length.<sup>6</sup>

Taking the length of beam as being the base diameter

$$\lambda \ell = 13.1 > \pi$$

The beam was then analyzed as a beam of unlimited length loaded over an area equal to the base diameter with an 80 psi uniform load.

The solution to this problem gives

$$y_c = \frac{q}{2k} (2 - D_{\lambda a} - D_{\lambda b})$$

$$M_c = \frac{q}{4\lambda^2} (B_{\lambda a} + B_{\lambda b})$$

$$Q_c = \frac{q}{4\lambda} (C_{\lambda a} - C_{\lambda b})$$

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where

$y_c$  is deflection of point being considered

$M_c$  is the moment at point being considered

$Q_c$  is shear at point being considered

$q$  is the uniform load

$a$  is the distance from point under consideration to end of load

$b$  is distance from point under consideration to other end of load.

$$B_{\lambda x} = e^{-\lambda x} \sin \lambda x$$

$$C_{\lambda x} = e^{-\lambda x} (\cos \lambda x - \sin \lambda x)$$

$$D = e^{-\lambda x} \cos \lambda x$$

Maximum moment occurs at mid-point of load and is equal to 352-in.-lbs/in.

For the area of the mat where there is only temperature reinforcing, the maximum moment would cause a stress of 30 psi in the reinforcing.

The maximum shear would occur at the ends and is equal to 2.64 kips/in. This shear would cause a shear stress in an unreinforced concrete section of 26.4 psi.

3. A second case examined was for the foundation material being less rigid than the concrete base. The model was the same for the first case:

$$\text{Assumed } E_{\text{rock}} = 2.6 \times 10^6 \text{ psi}$$

$$\mu = 0$$

For this case, the following were determined:

$$k_o = 1890 \text{ lb/in.}^3$$

$$\lambda = 6.2 \times 10^{-3} \text{-in.}^{-1}$$

$$M_{\text{max}} = 3.66 \text{-in.-kips/in.}$$

$$Q_{\text{max}} = 3.23 \text{ kips/in.}$$

$$S_{\text{rebar}} = 312 \text{ psi}$$

$$v_{\text{conc}} = 32.3 \text{ psi}$$

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As a final study, the maximum deflection as calculated in the first case was imposed as a settlement of the base mat for the outer portion and a section of the mat was analyzed for this settlement. A 30-ft section was used with fixity at the reactor pit, the remainder cantilevered from the pit.

The resulting moment and shear are as follows:

$$M = 142\text{-in.-kips/in.}$$

$$q = 396 \text{ lbs}$$

resulting in a rebar stress of 12.2 ksi and a shear stress of 4.0 psi.

From the above, it can be seen that the assumption that a foundation on rock is a rigid unyielding foundation is a valid assumption and that temperature reinforcing provides much greater resistance than required to accommodate the effects of any elastic deformation of the subgrade.

### 3.16.3.6 Analysis of Liner and Reinforcing Steel

Approximately 67-percent of the inclined bars, provided to resist radial shear at the base of the containment wall, are secondary vertical bars, which are inside the primary vertical bars on the outside face and inside face of the wall. These bars are continuous and are bent across the wall where reinforcing is required to resist the radial shear. The remaining 33-percent of the required steel area is provided by stirrups that are hooked around the vertical bars by means of a 90-degree hook. Only one-third of the shear reinforcing at a particular elevation is made up of these hooked bars, which occur at four elevations up the wall. See Figure 4.16 of the Containment Design Report.

Since the stud anchors are hooked around reinforcing bars, concrete stresses for pull out loads are negligible. For high shear loads, which would be caused if a stud anchor should fail or be missing, local crushing of the concrete occurs; however, integrity of the anchor and liner plate is not impaired. See Figures 3.16-12 and 3.16-13.

The lowest elevation at which these hooked bars are used is at a point where only 65-percent of the maximum shear at the base is present. The remaining three levels are in regions where the shear is less than 25-percent of maximum base shear. Since the large majority of the shear is resisted by continuous vertical bars, a minimal amount of load must be transmitted to the vertical bars. The hooked stirrups will mechanically transmit the small amount of shear, which they carry. The main function of the stirrups is to contain the formation of the diagonal tension crack. The mechanical anchorage of the stirrups is sufficient for this purpose.

There are no significant structural loadings, which must be transferred through the liner such as those required for crane brackets or machinery equipment mounts. Miscellaneous spray system piping, instrumentation, conduit, and insulation, which are attached to the liner can be supported by the free-standing liner without inducing significant stresses in the liner or liner anchorage.

Liner stress is imposed on the cylindrical penetration as a circular uniform load acting around the circumference of the penetration. The liner plate is locally thickened at the penetrations to take care of additional stresses.

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The liner can accommodate any shear it will see due to thermal expansion or earthquake.

An investigation was made on the thermal effects, based on the conservative assumptions that the base mat was fully fixed against any thermal movement thereby restraining the liner from movement. The 3-ft fill slab was then subjected to thermal growth. No excessive forces were introduced into the liner and the welds on the test channels were found to be sufficient to prevent any shear failure of the test channels from the liner due to movement of the 3-ft fill mat.

Seismic shear of the interior concrete is resisted by the keying action of the reactor pit and the sump for the recirculation pumps in addition to the weld channels. Considerable resistance is also provided by friction between the liner and the 3-ft slab.

Jet forces cannot remove the liner panels since the forces will be compressing the insulation panels against the liner and exterior wall. The panels are anchored to the liner with 3/16-in. diameter stainless steel studs. The consequence of an insulation panel being displaced from the liner during or as a consequence of an accident is that the exposed liner would tend to expand. The unequal strain between the exposed and unexposed portions of the liner causes a shear load on the liner anchor, and a local yielding in compression of the exposed portion of the liner. The liner anchor stud has the capacity to accommodate much greater strains than would be experienced at yield strain in the liner.

### 3.16.3.7 Containment Interior Structure

The interior structure may be separated into five main structural components. They are:

1. 3-ft thick fill slab.
2. 3-ft thick crane wall.
3. 4-ft to 6-ft thick refueling canal.
4. 2-ft thick operating floor slab.
5. Primary shield wall.

The method of design, stress analysis, critical stresses and locations are as follows:

1. 3-ft thick fill slab - The controlling loads on the 3-ft slab are the reactions are from the primary equipment supports due to various historical postulated pipe breaks. The slab was designed as a series of radial beams running under the equipment supports and spanning between the reactor support wall and the crane wall. Stresses in reinforcing were limited to 0.9 fy. Maximum stresses occur immediately below the primary equipment supports.
2. 3-ft thick crane wall - The crane wall was designed for a 7 psi differential pressure occurring immediately after a historical primary pipe break and prior to pressure equalization.

Although the stress levels associated with this pressure differential were sufficiently low to establish that the concrete could resist the pressure loading, sufficient reinforcing was provided to resist all membrane forces without any contribution from the concrete. Stresses were limited to 0.9 fy. The membrane hoop stress was 33 ksi and the axial vertical rebar stress was 14.3 ksi.

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A two dimensional finite element analysis was performed to determine the effect of the jet forces associated with the pipe break on the crane wall.

The jet force associated with a historical pipe break has been based on the static force  $PA$  where  $P$  is the primary system operating pressure and  $A$  is the cross sectional area of the coolant pipe. The analysis indicated that in local areas (at the application of the force) yielding of the crane wall rebar will occur. The load was assumed to act at the mid-height of the wall, thus causing maximum bending moment. The ability of the wall to support the dead load of the crane was checked, considering the yielded area indicated by the computer analysis as unable to carry load. A beam 12-ft long and 5-ft deep (the underside of the operating floor to the top of the potential yield portion of the crane wall) was found to provide more than twice the ultimate capacity required. This analysis was very conservative for three reasons:

1. A jet force load at this location would cause little yielding since it is not located at mid span.
2. The haunch at the underside of the operating floor was not considered.
3. The membrane effect of the circular crane wall was not taken into account.

Further stability of the crane wall was demonstrated by determining the ultimate failure load by means of a yield line analysis. This analysis indicated that the structure has the capacity, through strain energy of structural response, to resist the uniform jet force load of 1500 kips or 975 kips with the 7 psi pressure differential without failure.

The containment internal concrete is essentially rigid; (fundamental frequency 18.6 cps) therefore, seismic loads were calculated using the maximum ground acceleration (0.15g).

The crane wall was initially considered as a cantilever beam with a frequency of approximately 13 cps and the base shear was determined by the response spectrum approach. The base shear was distributed to the individual nodes by the formula:

$$F_x = \frac{W_x h_x V}{\sum W h}$$

Where

$$\begin{aligned} V &= \text{base shear} \\ W_x &= \text{weight of node under consideration} \\ h_x &= \text{distance from base to section under consideration.} \\ \sum W h &= \text{Summation of the product of weights and heights of all nodes} \end{aligned}$$

The moment at the base was determined and the uplift calculated by considering a circular ring of thickness equal to the area of steel per in. This maximum uplift, which occurs at one point at the base of the structure stresses the rebar to 5.2 ksi.

The crane wall was also designed to resist steam and feed water pipe break reactions of 340 kips and 200 kips where supports are connected to the wall. The extra steel provided for pipe break loads is available in the form of steel buttresses to resist pressure, jet force, and seismic loads; however, it was not considered in the analysis.



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3. 4-ft to 6-ft thick refueling canal - The refueling canal was designed for the 7 psi pressure differential. The wall resists the pressure by spanning vertically between the refueling floor and the operating floor. Stresses were limited to 0.9 fy.

An analysis was performed to check the effects of the jet force load the cross section was found to be sufficient to provide stability. A yield line analysis was performed and provided the basis for the above.

The seismic load was determined by the same procedure used for the crane wall. The average load in kips/ft was distributed over the wall and the vertical span was conservatively assumed to carry the entire load. The resulting bending moment was found to be well within the capacity of the wall.

4. 2-ft thick operating floor slab - Because of the many openings in the floor for equipment, the floor was designed as a series of beams. Principal loadings were D.L. + 500 psf live load and 7 psi upward pressure differential + D.L. The first loading (D.L. + 500 psf live load) was designed in accordance with Part IV-B of ACI 318. Stresses for the pressure differential case were limited to 0.9 fy.

The operating floor was investigated. The following is retained for its historical context. There appears to be very little area of the operating floor, which could be reached by the expanding jet of water from a break in the reactor coolant system. The jet will be greatly dispersed in the distance between the primary coolant piping and the underside of the operating floor. The only area of the floor, which could be struck by a jet spans between areas of the floor heavily reinforced as beams. The span cross section consists of a T-beam with the 2-ft thick floor acting as the flange and the 7-ft high biological shielding wall as the web. This section can resist the jet force load within 0.9 fy stress limit on the rebar.

5. Primary Shield Wall - This was designed for two loading conditions due to a split in the reactor. The stress in the reinforcing was limited to the tensile strength of the bars. The first load considered was a 1-ft wide longitudinal split along the length of the reactor. The vessel is assumed accelerated through a 6-in. distance against the support wall by the jet force caused by a 2200 psi pressure acting through a 26.4-ft long by 1-ft wide longitudinal vessel rupture, which results in an impact load of 650 k/ft. This load is imposed by considering an impact factor of two. The maximum rebar stress is 69.5 ksi. The second load considered a pressure buildup of 1000 psi inside the pit due to release of reactor contents. This produces a rebar stress of 86 ksi. The rebar used is ASTM A 432 with specified yield of 60 ksi and ultimate tensile strength of 90 ksi.

To protect the containment base liner, an average of 2-ft of concrete above the containment liner plus a 1-in. liner plate embedded on top of the concrete was provided at the bottom of the containment reactor cavity pit. Below the containment liner plate is 4.5-ft of structural concrete poured on rock.

The following is retained for its historical context. Temperature differential conditions as a result of a LOCA are considered to be of such short duration that the effects were not used in the design of interior structures for stress analysis. A sketch of the design conditions is given in Figure 3.16-15.

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The following is retained for its historical context. During normal operations, the only significant transient temperature gradients occur during startup. The minimum containment internal temperature is limited to 50°F. The maximum operating containment internal temperature is 130°F. Forced movement of containment air is used to limit the concrete temperature surrounding the reactor vessel. This forced air movement of the containment air as well as normal convection and radiation is expected to limit the concrete temperature differentials in the range of 5°F to 10°F. To demonstrate the large margin available in the concrete crane wall and the primary shield wall, a conservative assumption of a 30°F temperature gradient has been evaluated. The evaluation included the gradient effect through the crane wall, the 6-ft thick portion of the primary shield wall below the reactor coolant pipe nozzle, the 5-ft thick portion of the primary shield wall where the nozzles penetrate the wall, and the 4-ft thick wall above the shield wall.

The maximum rebar stress was found to be 4500 psi and occurs in the vertical rebar in the crane wall. The maximum compressive concrete stress was found to be 226 psi and occurs in the hoop direction in the 5-ft portion of the primary shield wall. These stresses are approximately 20-percent of the allowable working stress values and will have no significant effect on the design adequacy of the structures analyzed.

### 3.16.3.8 Soil Pressure

Portions of the containment structure are subjected to the effects of backfill bearing against the containment wall. The effects on the structure are:

1. Shear and overturning effects due to seismic response and interaction between the soil and structure.
2. Discontinuity effects caused by the soil restraining deformation of the structure under accident pressures.

To determine the shear and overturning effects two limiting cases were investigated. The first was the case where the structure and soil move out of phase. It was assumed that the structure was subjected to the passive pressure of the soil with the mass of soil, within the shear failure envelope, accelerated against the structure with ground acceleration. In the second case the soil and structure move in phase. For this case it was assumed that the structure was subjected to the active pressure of the soil with the mass of soil, within the shear failure envelope, accelerated with the structure at ground acceleration.

These loads were then treated as external loads on the structure. See Section 3.1.5 of the Containment Design Report for additional information.

To determine the discontinuity effects caused by soil restraint, the structure was analyzed for the passive pressure case. The restraint of the deformation of the structure due to the soil was calculated. Vertical and circumferential bending moments due to this restraint were then

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determined. Reinforcing bar stresses were calculated and found to be minor. This analysis was then verified by a finite element analysis.

In this analysis, full contribution of the backfill was assumed. During the course of construction it became necessary to build a retaining wall in a substantial area of the backfill, to facilitate construction. The retaining wall extends over 50-ft in plan and includes all of the high fill points assumed in the analysis and design. It can therefore be concluded that the analysis was conservative in that the backfill effects on the completed structure would be only a fraction of that assumed in the original design.

### 3.16.3.9 Thermal Stresses

Temperature effects on the containment structure are due to a thermal gradient through the wall. The reinforced-concrete wall restrains the liner from growing, resulting in compression in the liner and additional tension in the reinforcing.

Calculation of gradient stresses is based on method of analysis outlined in ACI 505-54, "Specification for the Design and Construction of Reinforced Concrete Chimneys."<sup>9</sup> The gradient used is linear with 120°F on the inside and 0°F exterior concrete temperature (-5°F ambient). The maximum operating temperature of the containment is 130°F. The effect of elevated operating temperature (up to 150°F) on the structural elements was evaluated in 1987 and was found to be acceptable.

The ACI method assumes a cracked section in which the concrete carries no tension. The neutral surface (surface at which no thermal stress exists) is determined. Stresses in the liner and reinforcing are calculated based on the assumption that there is no distortion of the wall; i.e., variation of strain through the wall thickness is linear.

### 3.16.3.10 Analysis of Openings

The methods followed in design of large openings are described in Section 3.4 of the Containment Design Report (CDR). Included are descriptions of the safety factors used in design. Sample calculations are provided, listing all the criteria and analyzing the effects of all pertinent factors, such as cracking. Also addressed in the CDR is how the existence of biaxial tension in concrete (cracking) has been taken care of in the design, and how the normal and shear stresses due to axial load, two-directional bending, two-directional shear, and torsion are combined. Additionally, the criteria for the design of the thickened part of the wall around the openings is stated.

The methods used to check the design of the thickened stiff part of the shell around large openings and its effect on the shell, torsional stresses, and shrinkage considerations are also addressed in Section 3.4 of the Containment Design Report. This section also describes how deformations and forces are handled around the large openings and in the transition zones into the main portion of the structure.

In the cylindrical section of the containment, where there are large openings for access hatchways and penetrations, the reinforcing bars (hoop, vertical and diagonal) are continued without interruption around the openings.

No bar terminates at any openings as illustrated around the penetration in Figure 3.16-1. Also, additional bars have been furnished locally to take the stresses developed around large openings.

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Concrete is locally thickened at the equipment access hatchway area to accommodate all the reinforcing bars required in this area.

A finite element analysis is performed on the large openings. Representation of the structure is by rectangular elements; each element consists of ten layers of orthotropic, elastic material to represent the reinforcement, concrete and the liner. About 1000 degrees of freedom are considered in the model. This analysis is used as a check on the adequacy of the large openings. Results appear in the Containment Design Report.

A finite element analysis of the equipment hatch area indicated local liner plastic deformations during the pressure test. For the order of magnitude and location of these stresses, see Section 3.4 of the Containment Design Report. These deformations have no influence on the structure during the pressure test due to the ductility of the studs and liner plate.

The limiting elastic liner deformations during test pressure will be from tensile stresses. During an accident loading they will be from compressive stresses. Therefore, a relationship between the pressure and accident loads cannot be determined directly. However, the test pressure demonstrates the ductile behavior of the liner.

Since the containment is not subject to accident temperatures during the testing, no direct correlation between test and accident conditions can be made in evaluating thermal stresses at large openings.

The liner is stressed beyond the yield point in very local areas adjacent to the transition from the thickened equipment hatch boss to the cylinder wall. The maximum stress is equal to 39.28 ksi for the 1.5P loading condition. The strain corresponding to this stress (0.17-percent) is below the limits (0.5-percent) stated in Section 2.2.4 of the Containment Design Report. The average liner stress in the cylinder for the 1.5P load combination is approximately -15 ksi in the vertical direction and -2.0 ksi in the horizontal direction.

The maximum rebar stress associated with the 1.5P load combination is approximately 66 ksi in the 4'-6" portion of the containment wall cylinder.

For a complete discussion of liner stresses, see the Containment Liner Stress Analysis Report. For a detailed discussion of liner stresses in the equipment hatch area and further justification of the stresses noted above, see Section 3.4.4 of the Containment Design Report.

All reinforcing is continuous around penetrations. Steps have been taken to ensure that no local crushing of concrete will occur. From Reference 3-16-16, it has been determined that in order to prevent local crushing of the concrete, a minimum bend diameter of 31 times the bar diameter is required when the reinforcing is stressed to yield. The angle of bend in the rebar determines the force that will be transmitted to the concrete in the event the bar tries to straighten out due to tension. For this reason most bars are bent at 10 degrees except at large penetrations including the equipment hatch, personnel lock, main steam and feedwater, and air purge penetrations, where the deviation of the bar from its centerline is too large to permit a 10-degree bend. In these cases the bars have been bent at 30 degrees but a tie-back system is used, which prevents a buildup of forces. To prevent this buildup, (in all cases except the equipment hatch penetration), the line of force makes an angle of one-half of the angle of bend, from a horizontal line from the vertical bars and from a vertical line for the horizontal bars and is tangent to the outside of the penetration.

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At the personnel and equipment hatches a large void will be carried since, due to the large offset of the bars from their centerline, it will take the bars longer to return to their centerline after passing the penetration. To prevent any cracking and spalling of concrete and to add lost strength to the cross-section, these voids have been filled with added rebar, which achieves bond by means of mechanical anchorage.

The same precautions mentioned above have been taken with the seismic bars. See Figure 3.16-16.

For penetrations between 9-in. and 18-in. in diameter, all the reinforcing bars including primary and secondary vertical bars and diagonal bars have been grouped around the penetrations. Due to the continuity of the bars and the relatively small opening size, no special provisions need be made to resist normal, shear, and bending stresses. The penetrations are keyed into the concrete, thus creating an edge loading, which will put torsion into the wall. The loads are small and the rebar will feel little effects from this torsional loading.

For penetrations greater than 18-in. up to 48-in. in diameter, the bars are continuous. Due to the large angle of bend of these bars, a tie-back system is used, which offers additional resisting strength to shear, bending, and torsional stresses.

### 3.16.3.11 Seismic and Wind Design

The design of the containment, which is a Class I structure (see Section 1.7), is based on a "response spectrum" approach in the analysis of the dynamic loads imparted by earthquake. The seismic design takes into account the acceleration response spectrum curves as developed by G. Housner. Seismic accelerations have been computed as outlined in TID-7024<sup>10</sup> and Portland Cement Association Publication.<sup>11</sup>

The following damping factors have been used:

<u>Component</u>		<u>Percent Critical Damping</u>
1.	Containment structure	2.0
2.	Concrete support structure of reactor vessel	2.0
3.	Steel assemblies:	
	a. Bolted or riveted	2.5
	b. Welded	1.0
4.	Vital piping systems	0.5
5.	Concrete structures above ground:	
	a. Shear wall	5.0
	b. Rigid frame	5.0

As indicated in Section 3.16.2.2, ground accelerations used for design purposes are 0.1g applied horizontally and 0.05g applied vertically. The natural period of vibration is computed by the Rayleigh method; in this method, the containment structure is analyzed as a simple cantilever intimately associated with the rock base and with broad base sections of adequate strength to

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assure full and continued elastic response during seismic motions. Further, both bending and shear deformations are considered.

The structure is divided into sections of equal length and loaded laterally by dead weight of the section and any equipment and live load occurring at the section. Deflections caused by shear and moments are then determined, and the end deflection is given the value  $\phi' = 1.0$  with corresponding values determined for other sections. The natural period of vibration for the structure is then determined by setting potential energy equal to kinetic energy and solving for the period.

$$T = 2\pi \sqrt{\frac{Y_0 \sum \phi^2 dm}{g \sum \phi dm}}$$

where

$Y_0$  = maximum actual deflection

$\phi = \frac{\text{deflection of section under consideration}}{\text{maximum actual deflection}}$

$g$  = acceleration due to gravity

$dm$  = weight of section under consideration

$T$  = period in sec.

Based on an uncracked concrete section, the period is determined to be 0.241 sec. A more realistic calculation for a cracked section, using reinforcing steel and liner as the resisting elements, yields a period 0.936 sec.

Using the derived period and entering the acceleration spectral curves, Figures 1.7-1 and 1.7-2 of Section 1.7, and applying a 2-percent critical damping, a spectral acceleration for the containment was selected. This value was derived to determine the base shear. The distribution of base shear is a triangular loading assumption.

This assumption yields a load distribution pattern with zero loading at the base to a maximum loading at the spring line of the dome. Above this line, the loading decreases due to a change in section and consequently change in weight. This load distribution allows the determination of shears and moments at any critical section through the containment from which the appropriate unit stresses are obtained.

Seismic shears are resisted by diagonal reinforcing except in the upper areas of the dome. No credit is taken for the reinforcing in compression.

From 30 degrees above the springline, where the seismic shears are small, the shears are carried by dome reinforcing steel lying in the plane of principal tension



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A finite element analysis was performed on the basemat using loads determined for the three basic loading conditions specified in the Containment Design Report. Maximum hoop moment caused by lack of symmetry of the seismic loading was found to be 454 in.-kips/in. This compares with a capacity of 690-in.-kips/in. for the in-place hoop reinforcing.

Tornado loads have not been considered in the design of the Indian Point Unit 2 Containment Building; however, similarity in design of Indian Point Unit 3 (where such loads are considered) indicates that seismic reinforcement bars provide a more than adequate mechanism to withstand the torsional effect of Tornado loads.

The torsional effect results from wind striking the containment building at an angle  $\alpha$  from the normal, as shown in Figure 3.16-17. The torsional force is due to the component of the wind tangential to the surface of the containment building and is equal to:

$$F_t = AC_D (q) (\sin \alpha)$$

Where

A = surface area of the containment

$C_D = 0.5$  from A.S.C.E. Paper 3269 - "Transactions of the A.S.C.E.," Vol. 126 Part II 1961, p. 1165 (coefficient of drag)

$q = 0.002558 V^2$  (wind pressure)

$\alpha = 45$  degrees

This assumption is conservative in that the actual tangential force would be the result of skin friction and the effects would be negligible.

This component of torsional force is computed from a direct wind loading as based on A.S.C.E. Paper 3269.

Torsional shear is a maximum at the juncture of the walls and base slab and varies to zero at the top of the dome.

The torsional effect can be converted to a shear per lineal foot around the circumference of the containment by distributing the shear over the circumference of the seismic reinforcing.

The seismic bars provide a more than adequate mechanism to withstand this torsional effect. The maximum stress in the bars under this loading is 17 ksi. See Figure 3.16-17.

### 3.16.3.12 Cathodic Protection (Historical Information)

During the initial Licensing process, a complete survey and tests to determine the need for cathodic protection on Indian Point Unit 2 was made by the A. V. Smith Engineering Company of Narberth, Pennsylvania. Electrical resistivity measurements and a visual inspection of the area away from the river, where the turbine generator building, reactor building, primary auxiliary building and associated facilities are located indicated that the environment is mostly rock with

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areas of dry sandy clay. The electrical resistivity of the soil ranged from 3,500 to 30,000 ohm-cm with the majority of the readings being above 10,000 ohm-cm. On this basis, it was determined that cathodic protection was not required on underground facilities in areas away from the river or the containment building liner, although a protective coating on pipes was recommended to eliminate any random localized corrosion attack. An analysis of Hudson River water data, obtained from the Con Edison plant chemist, showed the electrical resistivity of the water to vary over an extremely wide range due to salt intrusion from the ocean. The range of resistivity has been from 59 to 10,000 ohm-cm with a large number reading in the 300 ohm-cm area. This value was considered to be extremely corrosive and the following structures in the area near the river were placed under cathodic protection:

1. Circulating water lines.
2. Service water lines.
3. Bearing piles.
4. Sheet piling (earth and water side) and wing wall anchorage system.
5. Metallic structures inside intake structure (traveling screens, bar racks, circulating water pump suction, service water pump suction).

In 2008, a cathodic protection field survey and assessment of underground structures at Indian Point Unit 2 was performed by PCA Engineering of Pompton Lakes, New Jersey. A positive shift in pipe potential was found where the City Water supply piping from the City Water Tank crosses the Algonquin Gas pipes. As a result the City Water supply piping in the vicinity of the gas pipes was placed under cathodic protection.

In 2009, a guided wave assessment of buried piping at Indian Point Unit 2 was performed by Structural Integrity Associates, Inc. of Centennial, Colorado. The assessment identified minor corrosion indications on the Unit 2 CST Condensate supply and return piping in the vicinity of the AFW Pump Building. As a result this piping was placed under cathodic protection.

The cathodic protection system for the Circulating Water lines and the Service Water lines were found not to be functional and the rectifiers were removed. In order to assure the lines will perform their functions, the buried pipes are inspected as part of the Underground Piping and Tank Program. Inspections of buried piping are initially performed using Guided Wave (GW) ultrasonic inspection techniques to locate potential areas of degradation. If significant degradation is detected during the GW inspections, excavation is performed to uncover the affected sections of piping and a direct visual inspection and UT thickness measurements are performed. Repairs and / or replacements are implemented as required to restore degraded piping sections to within the required structural margins of safety.

In addition to the inspections performed as part of the Underground Piping and Tank Program, the historically categorized nuclear safety related portion of the service water piping is further subjected to pressure and / or flow testing as required by ASME XI, Subsection IWA-5244. Visual inspections on the inside surface of the SW piping are also performed under the GL 89-13, Service Water program. Based on the results of the inspections and testing, the Service Water system is structurally adequate to perform its required safety function.

The cathodic protection system for the Traveling Water Screens and Bar Racks were found not to be effective and the installed cathodic protective systems were retired. The original Traveling Water Screens which were carbon steel were upgraded to stainless steel frames, baskets, and chains. The splash housings are also stainless steel. The Bar Racks, replaced in the mid-1990's,



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are of carbon steel construction and are epoxy coated with a tar epoxy to provide corrosion protection. The guides for the Screens and Racks are carbon steel channels mounted in a concrete through. The rate of corrosion is slow and the Screens and Racks are on a regular PM cycle that checks for degraded conditions.

The Service Water and Circulating Water pumps suction are not cathodically protected. Rather, the Service Water Pumps suction is inspected and refurbished as part of the Service Water Pumps Preventing Maintenance (PM) activities. The Circulating Water Pumps are inspected and refurbished according to Preventive Maintenance program requirements.

### 3.16.3.13 Containment - Shear Crack

The arrangement of reinforcing bars in the containment shell is such that a reinforcing bar crosses any potential crack plane. Any cracks resulting from diagonal tension caused by shearing forces will be carried by reinforcing bars, which span across the crack. Thus all shears will be carried by the reinforcing bars and none by the concrete.

The reinforcing bars are almost all continuous throughout the containment structure; however, where a bar terminates this is accomplished by means of a 180-degree hooked bar. In no case are bars simply terminated without providing means for additional anchorage.

Throughout the cylinder, the meridional reinforcing is continuous. Beyond the springline, the bars extend radially toward the center of dome. As the bars reach a 6-in. spacing, which is one-half the required spacing, alternate bars have been dropped off by means of reinforcing splice plates. The splice piece consists of a plate with two Cadweld sleeves welded on the incoming side and one sleeve welded on the outgoing side. Thus, the number of bars present is halved and the spacing is increased to the required 12-in.

This is repeated to the top of the dome where a three-layered grid pattern has been used to maintain the continuity of the rebars. The bars in the grid pattern have been Cadwelded to the same type reinforcing splice plates described above, but the Cadweld is beveled to obtain the desired direction of the grid.

At the base in the area of high discontinuity stresses, additional No. 18S bars have been provided. At the point where they were no longer needed, they have been Cadwelded to a No. 11 bar, which is terminated with a 180-degree hook.

All seismic bars have been terminated in a 180-degree hook. In no case was a No. 18S bar terminated in this way since the minimum 180-degree hook could not be provided in a 4-ft 6-in. thick wall.

Radial shear reinforcing stirrups were terminated by hooking around vertical bars.

### 3.16.5 Longitudinal Splitting

The cavity wall is designed to withstand the forces and internal pressurization associated with a longitudinal split without gross damage. See Section 3.16.3.7 for a discussion of the analysis of this assumed historical accident condition.

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### 3.16.6 Containment Structure Design Evaluation

#### 3.16.6.1 Performance Capability Margin

The containment structure is designed based upon limiting load factors, which are used as the ratio by which historical postulated accident and earthquake loads are multiplied for design purposes to ensure that the load/deformation behavior of the structure is one of elastic, low strain behavior. This approach places minimum emphasis on fixed gravity loads and maximum emphasis on accident and earthquake loads. Because of the refinement of the analysis and the restrictions on construction procedures, the load factors primarily provide for a safety margin on the load assumptions. Load combinations and load factors used in the design, which provide an estimate of the margin with respect to all loads, are tabulated in Section 3.16.2.

#### 3.16.7 Preoperational Tests

After the containment building was complete with liner, concrete structures, and all electrical and piping penetrations, equipment hatch and personnel locks were in place, the following tests were performed.

#### 3.16.8 Strength Test

A pressure test was made on the completed building using air at 54 psig. This pressure was maintained on the building for a period of at least 1 hr. During this test, measurements and observations were made to verify the adequacy of the structural design. For a description of observations, cracks, strain gauges, etc., refer to Reference 3.16-18.

### REFERENCES FOR SECTION 3.16

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## IP2 DEFUELED SAFETY ANALYSIS REPORT

**TABLE 3.16-1**  
**Flooded Weights - Containment Building**

<u>Item</u>	<u>Flooded Operating Weight, lbs.</u>
Pressurizer - 1	346,000
Steam generators - 4	3,746,000
Reactor - 1	
(a) Vessel	868,000
(b) Internals	420,000
(c) Piping	1,000,000
Reactor pumps - 4	824,000
Accumulator tanks - 4	529,000
175-ton polar crane - 1	650,000
Ventilation fans - 5	656,000
Reactor coolant drain tank - 1	20,000
Pressure relief tank - 1	100,000
Other miscellaneous equipment	100,000
<u>Total</u>	9,259,000

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### 3.16 FIGURES

Figure No.	Title
Figure 3.16-1	Containment Structure
Figure 3.16-2	Cylinder and Dome-Load Condition (A) - 1.5P
Figure 3.16-3	Cylinder and Dome-Load Condition (B) - 1.25P
Figure 3.16-4	Cylinder and Dome-Load Condition (C) - 1.0P
Figure 3.16-5	Loading Diagram in Mat-Load Condition (A) - 1.5P
Figure 3.16-6	Loading Diagram in Mat-Load Condition (B) - 1.25P
Figure 3.16-7	Loading Diagram in Mat-Load Condition (C) - 1.0P
Figure 3.16-8	Weld Stud Connection at Panel Low Point
Figure 3.16-9	Weld Stud Connection at Panel Low Point
Figure 3.16-10	Weld Stud Connection at Panel Center
Figure 3.16-11	Wall Section
Figure 3.16-12	Cylinder Base Slab Liner Juncture
Figure 3.16-13	Typical Base Mat Liner Detail
Figure 3.16-14	Base Slab Reinforcing Detail
Figure 3.16-15	Reactor Cavity Pit
Figure 3.16-16	Equipment Hatch Personnel Lock, Main Steam and Feedwater, Air Purge - Rebar
Figure 3.16-17	Torsional Effects

# IP2 DEFUELED SAFETY ANALYSIS REPORT

## CHAPTER 4 WASTE DISPOSAL AND RADIATION PROTECTION SYSTEM

### 4.1 Waste Disposal System

#### 4.1.1 Design Bases

##### Control of Releases of Radioactivity to the Environment

Criterion: The facility design shall include those means necessary to maintain control over the plant radioactive effluents whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control must be justified (a) on the basis of 10 CFR 20 requirements, for normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10 CFR 100 dose level guidelines for potential reactor accidents of exceedingly low probability of occurrence (GDC 70).

Liquid, gaseous, and solid waste processing and handling facilities are designed so that the discharge of effluents and offsite disposal shipments are in accordance with applicable government regulations.

Radioactive fluids entering the waste disposal system are collected in sumps and tanks until determination of subsequent treatment can be made. They are sampled and analyzed to determine the concentration of radioactivity, with an isotopic breakdown if necessary. Before any attempt is made to discharge radioactive waste, it is processed as required. The processed water from waste disposal, from which most of the radioactive material has been removed, is discharged through a monitored line into the circulating water discharge. The system design and operation are characteristically directed toward minimizing releases to unrestricted areas. Discharge streams are appropriately monitored and safety features are incorporated to preclude releases in excess of the limits of 10 CFR 20.

The spent resins from the demineralizers and the filter cartridges are packaged and stored onsite until shipment offsite for disposal. Suitable containers are used to package these solids at the highest practical concentrations to minimize the number of containers shipped for burial.

All solid waste is placed in suitable containers and stored onsite until shipped offsite for disposal.

The application of the NUREG-1465 alternative source term methodology for Indian Point Unit 2 includes verification that the dose limits specified in 10 CFR 50.67 are met for low probability accidents.

#### 4.1.2 System Design and Operation

The waste disposal system process flow diagrams are shown in Figure 4.1-1, Sheets 1 and 2 (replaced with Drawings 9321-2719 and 9321-2730), and performance data are given in the Annual Radioactive Effluent Release Report.

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The waste disposal system collects and processes all potentially radioactive facility wastes for removal from the facility within limitations established by applicable government regulations. Fluid wastes are sampled and analyzed to determine the quantity of radioactivity, with an isotopic breakdown if necessary, before any attempt is made to discharge them. They are then released under controlled conditions. A radiation monitor is provided to maintain surveillance over the release operation, but the permanent record of activity release is provided by radiochemical analysis of known quantities of waste.

As secondary functions, system components provide facilities to transfer fluids from inside the containment to other systems outside the containment.

The Offsite Dose Calculation Manual (ODCM) provides the methodology to calculate radiation dose rates and dose to individual persons in unrestricted areas in the vicinity of Indian Point due to the routine release of liquid effluents to the discharge canal. The ODCM also provides setpoint methodology that is applied to effluent monitors and optionally to other process monitors.

Activity release due to tritium is given in the Annual Radioactive Effluent Release Report.

### 4.1.2.1 System Description

#### 4.1.2.1.1 Liquid Processing

The waste disposal system processes liquids from the following sources:

1. Equipment drains and leaks.
2. Chemical laboratory drains.
3. Decontamination drains.
4. Floor drains.

The reactor coolant drain tank collects and transfers liquid drained from the following sources:

1. Reactor coolant loops.
2. Refueling Canal Drain
3. Containment Spray Header Recirculation Lines

The fluid pumped by the reactor coolant drain pumps is sent to the waste holdup tank. The waste holdup tank serves as the collection point for liquid wastes. It collects fluid directly from the following sources:

1. Reactor coolant drain tank pumps
2. Containment sump pumps.
3. Holdup tank pit sump pump.
4. Sump tank pump (from primary auxiliary building).
5. Equipment drains.
6. Chemical drain tank pump.
7. Relief valve discharge from the component cooling surge tank and the chemical and volume control system holdup tanks.
8. Maintenance and Operation Building floor drains.
9. Primary Auxiliary Building sump pumps.

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Where facility layout permits, waste liquids drain to the waste holdup tank by gravity flow. Other waste liquids, including floor drains, drain to the sump tank or to the primary auxiliary building sump. The liquid wastes are pumped to the waste holdup tank. The liquid waste holdup tank is processed by sending its contents to the Unit 1 waste collection system.

The Indian Point Unit 1 waste collection system has four tanks with a capacity of 75,000 gal each. From there the liquid can also be processed by use of sluiceable demineralizer vessels.

A portable demineralization system is being used in the Unit 1 Chemical System Building. The system employs a number of in-line ion exchanger resin beds and filters to remove radionuclides and chemicals as required from the waste stream. The demineralization/filtration system processes liquid waste from the unit 1 waste collection tanks and discharges the clean water to the distillate storage tanks.

Spent resins from the portable system are sluiced from the vessels into a high integrity container, which is dewatered and then transported to the burial site without solidification. Spent filters can also be placed in the high integrity container.

The processed water produced by the demineralizer water processing is collected in two distillate storage tanks. Each storage tank is vented to the unit 1 ventilation system. When a distillate storage tank is ready for discharge, it is isolated and sampled to determine the allowable release rate. If the contents of the tank are not suitable for release, they are returned to waste collection tanks for reprocessing. If analysis confirms that the activity level is suitable for release, the distillate is discharged to the river. A radiation detector and high radiation trip valve are provided in the release line to prevent an inadvertent release of activity at concentrations in excess of the setpoint derived from the ODCM.

### 4.1.2.1.2 Gas Processing

Gas processing has been abandoned. The Vent Header and Gas Decay tanks were inerted and opened to atmosphere. There is no longer a use for Nitrogen as a cover gas and radiological/explosive gases are no longer generated.

### 4.1.2.1.3 Solids Processing

Solid waste processing is controlled by the Process Control Program in the ODCM.

Resin is normally stored in the spent resin storage tank for decay; this tank is described in Section 4.1.2.2.5. Resin is removed from the storage tank to a high integrity container, which is dewatered and prepared for transportation in accordance with the Process Control Program. Spent filters can be placed in the high integrity containers.

Miscellaneous solid wastes such as paper, rags and glassware, are processed in accordance with the Process Control Program. When possible, solid waste is sent to a licensed incineration and volume reduction center, or to a material recovery center. This process is controlled by the Process Control Program.

The unit 1 containment has been modified for use as an interim onsite storage facility for dry active waste.



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The Original Steam Generators (OSGs) are stored in the Original Steam Generator Storage Facility (OSGSF). Storage in this building is limited to the OSGs. The OSGSF is a reinforced concrete structure measuring approximately 150 feet by 54 feet (not including the labyrinth entryways). The building is located on the eastern side of the facility, between Electrical Tower 3 and the Buchanan Service Center access road. This location is within the Owner Controlled Area outside the Protected Area. The structure is constructed of cast-in-place concrete. Except for the South wall, which consists of pre-cast stackable concrete blocks. Use of pre-cast blocks provide access to install the OSGs and for removal of the OSGs at a later date. The roof is covered with a single-ply membrane roofing system.

The walls of the OSGSF are 3'-0" thick and the roof is tapered from 2'-6" in the center of the building to 2'-0" at the east and west walls. The slab is 3'-0" thick with a thickened perimeter that is 5'-0" thick. Personnel access doors with labyrinth entryways are provided at each end of the building to prevent radiation streaming through the door. The walls of the labyrinth entryway are 3'-0" thick with the roof over the labyrinth entryway tapered from 1'-2" to 1'-0". Two locked steel doors in each entryway will provide access to the building after the pre-cast concrete blocks are put in place, one in the exterior wall opening and one in the labyrinth wall.

The OSGSF is designed to contain contaminated materials and facilitate decontamination should such an action become necessary. Waterstops are used at all construction joints to prevent both the intrusion of water into the facility and the escape of contaminated water from the facility. The floor of the facility is sloped to provide adequate drainage to a sump. Protective coatings are applied to the floor slab and lower portion of the walls to ease decontamination, if required. A passive HEPA filter system is provided to allow venting of the OSGSF while containing any airborne contamination.

An electrical system provides interior and exterior lighting, 110-volt AC outlets, and a remote alarm system on each entryway. Two locked steel doors secure the building and a security fence is installed around the perimeter of the building.

### 4.1.2.2 Components

Codes applying to components of the waste disposal system are shown in Table 4.1-1. Component summary data is shown in Table 4.1-2. Waste disposal system components are located in the auxiliary building except for the reactor coolant drain tank, which is in the containment and the waste holdup tank, which is in the liquid holdup tank vault.

#### 4.1.2.2.1 Chemical Drain Tank

The chemical drain tank is a vertical cylinder of austenitic stainless steel and collects drainage from the chemistry sampling station. The tank contents are pumped to the waste holdup tanks.

#### 4.1.2.2.2 Reactor Coolant Drain Tank

The reactor coolant drain tank is a horizontal cylinder with spherically dished heads. The tank is all welded austenitic stainless steel. This tank serves as a drain surge tank for the reactor coolant system and other equipment located inside the reactor containment. The water collected in this tank is transferred to the chemical and volume control system holdup tanks, the refueling water storage tank, or the waste holdup tank.

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### 4.1.2.2.3 Waste Holdup Tank

The waste holdup tank is the central collection point for radioactive liquid waste. The tank is stainless steel of welded construction.

### 4.1.2.2.4 Sump Tank and Sump Tank Pumps

The sump tank serves as a collecting point for waste discharged to the basement level drain header. It is located at the lowest point in the auxiliary building. Floor drains enter this tank through a loop seal to prevent back flow of gas from the tank. Two horizontal centrifugal pumps transfer liquid waste to the waste holdup tank. All wetted parts of the pumps are stainless steel. The tank is all-welded austenitic stainless steel.

### 4.1.2.2.5 Spent Resin Storage Tank

The spent resin storage tank retains resin discharged from the facility demineralizers. Normally, resins are stored in the tank for decay of short-lived isotopes. However, the contents can be removed at any time, if sufficient shielding is provided for the spent resin shipping vessel. A layer of water is maintained over the resin surface as a precaution against resin degradation due to heat generation by radioactive decay. Resin is removed from the tank by first sparging with nitrogen to loosen the resin and then pressurizing the tank with nitrogen to approximately 60 psig to force the resin slurry out of the tank. If desired, the primary water supply can be used instead of nitrogen for agitating the resin before discharging it from the tank. The tank is all-welded austenitic stainless steel.

### 4.1.2.2.6 Gas Decay Tanks

The Gas Decay Tanks have been permanently abandoned.

### 4.1.2.2.7 Compressors

The compressors have been permanently abandoned.

### 4.1.2.2.8 Distillate Storage Tanks

Two distillate storage tanks are provided.

The tanks are horizontal, cylindrical type with standard flanged and dished heads. Each tank is provided with heaters for cold weather temperature control.

### 4.1.2.2.9 Nitrogen Manifold

The Nitrogen manifold has been permanently abandoned.

### 4.1.2.2.10 Gas Analyzer

The automatic gas analyzer has been permanently abandoned.

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### 4.1.2.2.11 Pumps

Pumps used throughout the system for draining tanks and transferring liquids are shown on Figure 4.1-1 sheets 1 and 2 (replaced with Drawings 9321-2719 and 9321-2730).

The wetted surfaces of all pumps are stainless steel.

### 4.1.2.2.12 Piping

Piping carrying liquid wastes is stainless steel while all gas piping is carbon steel. Piping connections are welded except where flanged connections are necessary to facilitate equipment maintenance.

### 4.1.2.2.13 Valves

All valves exposed to gases are carbon steel. All other valves are stainless steel.

Stop valves are provided to isolate each piece of equipment for maintenance, to direct the flow of waste through the system, and to isolate storage tanks for radioactive decay.

Relief valves are provided for tanks containing radioactive waste if the tanks might be over-pressurized by improper operation or component malfunction. Tanks containing wastes, which contain oxygen and are normally of low activity concentrations are vented into the auxiliary building exhaust system.

## 4.1.3 Design Evaluation

### 4.1.3.1 Liquid Wastes

Liquid wastes are primarily generated by facility operations. The Annual Radioactive Effluent Release Report provides the total liquid effluent activity released by isotope.

Appendix 4B presents the results of an original plant preoperational assessment of river water dilution factors between the Indian Point site and the nearest public drinking water intake and is retained for historical purposes.

### 4.1.3.2 Gaseous Wastes

The Gaseous Waste System was retired. The system has been permanently abandoned.

### 4.1.3.3 Solid Wastes

Solid wastes consist of sludges, spent resins and filters, and miscellaneous materials such as paper and glassware.

Spent resins and filters are packaged in liners, which are placed in waste casks for removal to a burial facility. Miscellaneous wastes are packaged in 52 or 55-gal drums. When possible, solid waste is sent to a licensed incinerator, volume reduction center, or material recovery center. Preparation of solid radwastes for shipment and offsite disposal is conducted in accordance with a process control program. Certain activities such as inspections and verifications are considered to be Quality Control activities.

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### 4.1.4 Minimum Operating Conditions

Minimum operating conditions for the waste disposal system are enumerated in the ODCM.

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**TABLE 4.1-1**  
**Waste Disposal System Components Code Requirements**

<u>Component</u>	<u>Code</u>
Chemical drain tank	No code
Reactor coolant drain tank	ASME III, <sub>1</sub> Class C
Sump tank	No code
Spent resin storage tanks	ASME III, <sub>1</sub> Class C
Gas decay tanks	ASME III, <sub>1</sub> Class C
Waste holdup tank	No code
Distillate storage tank	No code
Waste filter	No code
Piping and valves	USAS-B31.1, <sub>2</sub> Section 1

Notes:

1. ASME III, American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section IV, Nuclear Vessels.
2. USAS-B31.1, Code for pressure piping, U.S. American Standards Association and special nuclear cases where applicable.

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**TABLE 4.1-2 (Sheet 1 of 2)**  
**Component Summary Data**

<b>Tanks</b>	<b>Quantity</b>	<b>Type</b>	<b>Volume</b>	<b>Design Pressure</b>	<b>Design Temperature °F</b>	<b>Material</b>
Reactor Coolant drain	1	H	350 gal	25 psig	267	ss
Chemical drain	1	V	375 gal	Atm	180	ss
Sump	1	V	375 gal	Atm	150	ss
Waste holdup	1	H	3300-ft <sup>3</sup>	Atm	150	ss
Spent resin Storage	1	V	300-ft <sup>3</sup>	100 psig	150	ss
Distillate storage	2	H	25000 gal	17 psig	250	cs
Gas decay (large)	4	V	525-ft <sup>3</sup>	150 psig	150	cs
Gas decay (small)	6	V	40-ft <sup>3</sup>	150 psig	150	cs

<b>Pumps</b>	<b>Quantity</b>	<b>Type</b>	<b>Flow gpm</b>	<b>Head ft</b>	<b>Design Pressure psig</b>	<b>Design Temperature F°</b>	<b>Material<sub>1</sub></b>
Reactor coolant drain (A)	1	H, CC	50	175	100	267	ss
Reactor coolant drain (B)	1	H, CC	150	175	100	267	ss
Chemical drain	1	H, C <sub>2</sub>	20	100	100	180	ss

## IP2 DEFUELED SAFETY ANALYSIS REPORT

**TABLE 4.1-2 (Sheet 2 of 2)**  
**Component Summary Data**

Pumps	Quantity	Type	Flow gpm	Head Ft	Design Pressure psig	Design Temperature °F	Material <sub>1</sub>
Sump tank	2	H, C <sub>2</sub>	20	100	150	180	ss
Waste transfer	1	H, C <sub>2</sub>	30	215	105	70	ss
Distillate recirculation	2	H, C <sub>2</sub>	200	100	43 <sub>3</sub>	120 <sub>4</sub>	ss
Reactor cavity pit (2RCPP)	1	Sub-merge V, C	100	50	150	120	ss
Reactor cavity pit (1RCPP)	1	Sub-merge V, C	20	62	150	120	ss

Miscellaneous	Quantity	Capacity	Type
Waste gas compressors	2	48 f <sup>3</sup> /min	H, C <sub>2</sub>

**Key:**

H = Horizontal  
V = Vertical

C = Centrifugal  
CC = Centrifugal canned

CC = Carbon Steel  
SS = Stainless Steel

**Notes:**

1. Wetted surfaces only.
2. Mechanical seal provided.
3. 43 psig is the operating differential pressure of the pump.
4. 120°F is the maximum operating temperature of the pump

## IP2 DEFUELED SAFETY ANALYSIS REPORT

### 4.1 FIGURES

Figure No.	Title
Figure 4.1-1 Sh. 1	Waste Disposal System Process Flow Diagram, Sheet 1, Replaced with Drawing 9321-2719
Figure 4.1-1 Sh. 2	Waste Disposal System Process Flow Diagram, Sheet 2. Replaced with Drawing 9321-2730



## IP2 DEFUELED SAFETY ANALYSIS REPORT

### 4.2 Radiation Protection

#### 4.2.1 Design Bases

Radiation protection at Indian Point 2 incorporates a program for maintaining radiation exposures as low as reasonably achievable (ALARA). The ALARA program is part of all normal and special work processes. Procedures, designs, modifications, work packages, inspections, surveillances, maintenance activities and facility betterment activities are subjected to ALARA reviews to ensure dose reduction actions are taken. ALARA is taught in Radiation Worker Qualification courses

##### 4.2.1.1 Monitoring Radioactivity Releases

Criterion: Means shall be provided for monitoring the containment atmosphere and the facility effluent discharge paths for radioactivity released from normal operations, from anticipated transients, and from accident conditions. An environmental monitoring program shall be maintained to confirm that radioactivity releases to the environs of the plant have not been excessive. (GDC 17)

The plant vent and the waste disposal system liquid effluent are monitored for radioactivity during normal operations and accident conditions.

All gaseous effluent from possible sources of accidental releases of radioactivity (e.g., the spent-fuel pit and waste handling equipment) will be exhausted from the plant vent, which is monitored. Any contaminated liquid effluent discharged to the condenser circulating water canal is monitored. The details of the procedures and equipment to be used in the event of an accident are specified in Section 4.2.5, the procedures, and the emergency plan. The formulation of these details considers the requirements for notification of personnel, the utility load dispatcher, and local authorities.

##### 4.2.1.2 Monitoring Fuel and Waste Storage

Criterion: Monitoring and alarm instrumentation shall be provided for fuel and waste storage and associated handling areas for conditions that might result in loss of capability to remove decay heat and to detect excessive radiation levels. (GDC 18)

Monitoring and alarm instrumentation are provided for fuel and waste storage and handling areas to detect inadequate cooling and to detect excessive radiation levels. Radiation monitors are provided to maintain surveillance over the release operation, but the permanent record of activity releases is provided by radiochemical analysis of known quantities of waste.

The spent fuel pit temperature and level are monitored to assure proper operation, as discussed in Section 3.3.3.2.2.

A controlled ventilation system removes gaseous radioactivity from the atmosphere of the fuel storage and waste treating areas of the auxiliary building and discharges it to the atmosphere via the plant vent. Radiation monitors are in continuous service in these areas to actuate high-activity alarms on the control board annunciator, as described in Section 4.2.3.

##### 4.2.1.3 Fuel and Waste Storage Radiation Shielding

Criterion: Adequate shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities. (GDC 68)

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Auxiliary shielding for the waste disposal system and its storage components is designed to limit the dose rate to levels not exceeding 0.75 mrem/hr in normally occupied areas, to levels not exceeding 2.0 mrem/hr in intermittently occupied areas, and to levels not exceeding 15 mrem/hr in limited occupancy areas.

Gamma radiation is continuously monitored in the auxiliary building. A high-level signal is alarmed locally and annunciated in the control room.

### 4.2.1.4 Protection Against Radioactivity Release From Spent Fuel and Waste Storage

Criterion: Provisions shall be made in the design of fuel and waste storage facilities such that no undue risk to the health and safety of the public could result from an accidental release of radioactivity. (GDC 69)

All waste handling and storage facilities are contained and equipment designed so that accidental releases directly to the atmosphere are monitored and will not exceed applicable limits; refer also to Sections 4.1.2, 6.3, and 6.4. The components of the waste disposal system are designed to the pressures given in Table 4.1-2 and the codes given in Table 4.1-1. Hence, the probability of a rupture or failure of the system is exceedingly low.

### 4.2.2 Shielding

#### 4.2.2.1 Design Basis

Radiation shielding is designed to limit the normal radiation levels at the site boundary below those levels allowed for continuous non-occupational exposure.

Site personnel at the facility are protected by adequate shielding, monitoring, and procedures. When additional shielding is suggested, and permitted, it will be evaluated in the context of the station ALARA program and temporary shielding procedures. Modifications to existing structures or shields, which may alter personnel or equipment qualification dose will be evaluated in the design review process. The permanent large and significant shielding arrangement is shown on Figures 5.1-3, 5.1-4, 5.1-6 and 5.1-7. Shielding arrangements may be altered consistent with the radiation protection plan and the ALARA program station administration orders.

Detailed and periodic surveys of all restricted area radiation levels are performed. All high radiation areas are appropriately marked and access controlled in accordance with 10 CFR 20 and other applicable regulations and station procedures as well as the Technical Specifications.

The shielding is divided into the following categories according to function: (1) fuel transfer shielding; and (2) the auxiliary shielding.

#### 4.2.2.1.1 Fuel Handling Shield

The fuel handling shield is designed to attenuate radiation from spent fuel, control clusters, and reactor vessel internals to less than 2.0 mrem/hr at the refueling cavity water surface and less than 0.75 mrem/hr in areas adjacent to the spent-fuel pit.

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### 4.2.2.1.2 Auxiliary Shielding

The function of the auxiliary shielding is to protect personnel working near various system components in the chemical and volume control system and the waste disposal system, the sampling system and the high radiation sampling system sentry panels. The shielding provided for the auxiliary building is designed to limit the dose rates to less than 0.75 mrem/hr in normally occupied areas, and at or below 2.0 mrem/hr in intermittently occupied areas during normal activities. Samples may be diverted to a shielded high radiation sampling system tank. Liquid can be pumped from this tank back into the containment.

An additional room has been constructed in the primary auxiliary building (elevation 98-ft) to provide additional shielding protection for site personnel. All gas sample lines to the gas analyzers have been provided with a nitrogen purge capability. This system purges all the sampled gases from the sample lines and returns them to their source.

### 4.2.2.2 Shielding Design

#### 4.2.2.2.1 Fuel Handling Shield

Spent fuel is stored in the spent fuel pit, which is located adjacent to the containment building. Shielding, above grade elevation, for the spent fuel storage pit is provided by concrete walls 6-ft thick and is flooded to a level such that the water height is greater than 13-ft above the spent fuel assemblies.

The fuel handling shield design parameters are listed in Table 4.2-1.

#### 4.2.2.2.2 Auxiliary Shield

The auxiliary shield consists of concrete walls around certain components and piping, which process reactor coolant. In some cases, the concrete block walls are removable to allow personnel access to equipment during maintenance periods.

The shielding material provided throughout the auxiliary building is regular concrete ( $\rho = 2.3 \text{ g/cm}^3$ ). The principal auxiliary shielding provided is tabulated in Table 4.2-2.

### 4.2.3 Radiation Monitoring System

#### 4.2.3.1 Design Bases

The radiation monitoring system is designed to perform two basic functions:

1. Warn of any radiation health hazard, which might develop.
2. Give early warning of a facility malfunction, which might lead to a health hazard or facility damage.

Instruments are located at selected points in and around the facility to detect, compute, and record the radiation levels. In the event the radiation level should rise above a desired setpoint, an alarm is initiated in the control room. The automatic radiation monitoring system operates in conjunction with regular and special radiation surveys and with chemical and radio-chemical analyses performed by the facility staff. Adequate information and warning are thereby provided for the

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continued safe maintenance of the facility and assurance that personnel exposure does not exceed 10 CFR 20 limits.

### 4.2.3.2 Radiation Monitoring Betterment Program

The process radiation monitoring system is a digital system with the following major components: individual radiation monitoring units for each monitored process line; a minicomputer unit located in the technical support center; a CRT display and printer located in the central control room; and annunciators located in the central control room.

The minicomputer unit includes a console with CRT and typer, disk drive and magnetic tape drive. It communicates digitally with the individual radiation monitoring units, and processes, records, and displays data.

Table 4.2-3 shows the process streams monitored by the individual radiation monitor units, along with the normal maximum channel output. Each monitor unit monitors a sample of the process fluid, which is piped through a bypass loop. The sample is cooled if required. To facilitate maintenance and calibration, the bypass loop can be isolated and purged.

The liquid and airborne monitors utilize an off-line sampler(s) and a gamma or beta scintillation detectors to measure radioactivity present in a sample. Each monitor has a micro-processor, which communicates with the minicomputer.

Each monitor will activate an annunciation alarm in the event of failure, high radiation, or high temperature where applicable.

The minicomputer and the CRT/printer unit are powered from a battery-backed inverter. As discussed below, several monitor units receive power from MCC-26A and MCC-26BB, which are powered by the Appendix R / SBO diesel generator in the event of loss of other power sources.

Information on specific monitors is given in the following sections.

#### 4.2.3.2.1 Service Water from Component Cooling Heat Exchangers Monitors

Monitors R39 and R40 monitor the service water from component cooling heat exchangers 21 and 22, respectively. Radioactivity in these streams would indicate a component cooling heat exchanger leak when there is radioactivity in the component cooling loop. These monitors are powered from MCC-26A. They are wired to a control room annunciator, independent of their communications loop through the minicomputer.

#### 4.2.3.2.2 Plant Vent Air Monitor

R44 monitors for gaseous activity. It was historically seismically qualified, and its power supplies was historically class IE. On detection of a high activity level, R44 initiates closure of the gas discharge valve in the waste gas disposal system. Their signals are provided to control room indicators and recorders and to the safety assessment system. Additionally, an indicator for monitor R44 is located at the waste disposal panel.

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### 4.2.3.2.3 Component Cooling Radiation Monitor

This channel, R47, monitors the component cooling loop for radioactivity. An interlock initiates closure of a valve in the component cooling surge tank vent line in the event a high radiation level is detected. Closure of this valve will prevent gaseous activity release. Component cooling activity is recorded and displayed in the control room, and high activity initiates a control room annunciator. The display unit, recorder and annunciator are independent of the minicomputer communications loop. The monitor is isolated from the communications loop by an isolation device. This monitor is powered from MCC-26A.

### 4.2.3.2.4 This subsection not used

### 4.2.3.2.5 Liquid Waste Effluent Radiation Monitor

This monitor, R54, is powered from a Unit 1 motor control center. It alarms in the central control room independent of the communications loop through the minicomputer. This monitor terminates the tank discharges upon detecting high activity.

### 4.2.3.2.6 Unit 1 Stack Radiation Monitor

R60 monitors for gaseous activity in the air in the Unit 1 stack. Particulates are collected on filters and analyzed in the count room.

### 4.2.3.2.7 Sphere Foundation Drain Sump Liquid Effluent

Monitor R-62 monitors the activity of the liquid discharge from the Unit 1 Sphere Foundation Drain Sump drainage. This monitor alarms of the common process radiation monitor panel for high radiation.

## 4.2.3.3 Original Radiation Monitoring System

### 4.2.3.3.1 Control Room Cabinet

Most of the control room system equipment is centralized in three cabinets. High reliability and ease of maintenance are emphasized in the design of this system. Sliding channel drawers are used for rapid replacement of units, assemblies, and entire channels. It is possible to remove the various chassis completely from the cabinet after disconnecting the cables from the rear of these units.

### 4.2.3.3.2 Monitor Channel Output

The maximum channel output of the radiation monitors is given in Table 4.2-3.

### 4.2.3.3.3 Operating Conditions

Where fluid temperature is too high for the monitor, a cooling device with temperature indication is included. The different operating temperature ranges are within the design limits of the sensors.

The relation of the radiation monitoring channels to the systems with which they are associated is given in the sections describing those systems. Routine test and recalibrations will ensure that the channels operate properly.

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The components of the radiation monitoring system are designed according to the following environmental conditions:

1. Temperature - an ambient temperature range of 40°F to 120°F.

*[Note - Equipment located in the control room area may be specified for smaller temperature and humidity ranges because of the controlled environment provided by the heating and ventilating system.]*

2. Humidity - 0 to 100-percent relative humidity.

*[Note - Equipment located in the control room area may be specified for smaller temperature and humidity ranges because of the controlled environment provided by the heating and ventilating system.]*

3. Pressure - components in the auxiliary building and control room are designed for normal atmospheric pressure. Area monitoring system components inside the containment are designed to withstand test pressure.
4. Radiation - process and area radiation monitors are of a nonsaturating design so that they "peg" full-scale if exposed to radiation levels up to 100 times full scale indication. Process monitors are located in areas where the normal and post-accident background radiation levels will not affect their usefulness.

The radiation monitoring system is divided into the following subsystems:

1. The process radiation monitoring system, which monitors various fluid streams for indication of increasing radiation levels.
2. The area monitoring system, which monitors area radiation in various parts of the facility.
3. Environmental radiation monitoring system, which monitors radiation in the area surrounding the facility.

#### 4.2.3.3.4 Original Area Radiation Monitoring System

The Unit 2 area radiation monitoring system consists of two channels, which monitor radiation levels in various areas of Unit 2. These areas are listed as follows:

<u>Channel</u>	<u>Area Monitor</u>
R-1	Control Room
R-5	Spent fuel building

Channels R-1 and R-5 consist of a fixed position gamma sensitive Geiger-Mueller tube detector. The detector output is amplified and the log count-rate is determined by the integral amplifier at the detector. The radiation level is indicated locally at the detector and at the radiation monitoring system (RMS) cabinets. The RMS signals are also logged and trended (recorded) by the plant computer. High radiation alarms are displayed on the main annunciator, the radiation monitoring

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cabinets, and at the detector location. When radiation levels drop below the high level alarm setpoint, the "high" alarms on the monitors are reset automatically. The automatic reset procedure also exists for the "low" alarms.

The control room annunciator provides a single window, which alarms for any channel detecting high radiation. Verification of which channel has alarmed is done at the radiation monitoring system cabinets. A remotely-operated, long half-life radiation check source is provided in each channel. The source strength is sufficient to produce indication of detector response.

A meter is mounted on the front of each computer-indicator module and is calibrated logarithmically from 0.1 mrem/hr to 10 rem/hr.

A remote meter calibrated logarithmically from 0.1 mrem/hr to 10 rem/hr, is mounted at the detector assembly.

Radiation monitoring system cabinet alarms consist of a red indicator light for high radiation and an amber light to annunciate detector or circuit failure. The remote meter and alarm assembly at the detector contains a red indicator light and a buzzer type alarm annunciator actuated on high radiation.

### 4.2.3.4 NUREG-0737 Monitors

The following monitors were installed in conformance with NUREG-0737, "Clarification of TMI Action Plan Requirements":

#### 4.2.3.4.1 Wide Range Gas Monitor (R-27)

The wide range gas monitor is installed in the boric acid evaporator building on the 84-ft elevation along with a sample station. The monitor is intended to provide information about the magnitude of releases of radioactive materials, should they occur.

The monitor is skid-mounted and fixed in place by anchor bolts; the various parts of the sample station are similarly secured to the wall and floor. Connections have been installed for data processors and displays and to supply electrical power and a nitrogen purge capability. The display for this monitor is located on the accident assessment panel in the common Units 1 and 2 central control room.

#### 4.2.3.4.2 Control Room Air Intake

Process radiation monitors R-38-1 and R-38-2 are installed near the intake ducts in the northern and southern sections of the Control Room's fan room. The southern detector is located on the intake air stream for the Unit 1 area of the Control Building excluding the Control Room. The northern detector is near the Unit 2 intake duct where the duct penetrates the north wall of the fan room.

### 4.2.4 Environmental Monitoring Program

Environmental monitoring is discussed in Section 2.8 and requirements are set forth in the ODCM. The environmental monitoring program and results are described in the Annual Radiological Environmental Operating Report.



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### 4.2.5 Radiation Protection and Medical Programs

#### 4.2.5.1 Personnel Monitoring

The official and permanent record of accumulated external radiation exposure received by individuals is obtained principally from a Dosimeter of Legal Record (DLR). Direct reading and electronic dosimeters provide day-by-day indication of external radiation exposure.

Special or additional DLRs are issued as may be required under unusual conditions. These devices are issued as directed by the Radiation Protection department.

The DLRs are processed on a routine basis, typically at 6-month intervals.

Annual reports of personnel monitoring are submitted to the NRC in accordance with 10 CFR 20.2206.

#### 4.2.5.2 Personnel Protective Equipment

All personnel are required to wear appropriate protective clothing as specified by a radiation work permit. The nature of the work to be done is the governing factor in the selection of protective clothing to be worn by individuals. The most common protective apparel available is shoe covers, head covers, gloves, and coveralls. Additional items of specialized apparel such as plastic suits, face shields, and respirators are available. In all cases, radiation protection personnel evaluate the radiological conditions and specify the required items of protective clothing to be worn. Respiratory protective devices are available in any situation in which an airborne radioactive area exists or is expected to exist in excess of applicable limits. In such cases, the airborne concentrations are monitored by radiation protection personnel and the necessary protective devices are specified according to concentration and type of airborne contaminants present.

Respiratory devices available for use include:

1. Full-face respirator (filter or gas canister, negative pressure).
2. Atmosphere supplying respirators (pressure demand, or continuous flow).
3. Airhood.
4. Self-contained breathing apparatus.

Self-contained breathing apparatus will be used in any situation involving oxygen deficient atmospheres.

The appropriate type of respiratory protection equipment required will be determined from 10 CFR, 20.1701-1704.

#### 4.2.5.3 Facilities and Access Provisions

The radiologically controlled area is a portion of an area to which access is limited and additional steps are applied for purposes of occupational dose control and loose radioactive material control. A Radiation Area is an area accessible to personnel in which there exists radiation at such levels that a major portion of the body could receive in any 1 hr a dose in excess of 5.0 mrem at 30 cm from the source. The Radiologically Controlled Areas of IP2 are established, identified, and controlled through procedures.



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Any area in which radioactive material and radiation are present shall be surveyed, classified, and conspicuously posted with the appropriate radiation caution sign as specified in 10 CFR 20.1902.

The general arrangement of the control point facilities is designed to provide access control to the RCA and it also provides a change location for personal clothing.

Friskers and/or Personnel Contamination Monitors are located at all authorized personnel exits from the radiologically controlled area. All personnel will survey themselves before leaving the controlled area.

Personnel decontamination equipment is available in the controlled area decontamination and first aid rooms.

Administrative and physical security measures are employed to prevent unauthorized entry of personnel to any high radiation area. These measures are defined in Technical Specifications 5.7.1 and 5.7.2 of the IP2 PDTS.

### 4.2.5.4 Radiation Instrumentation

Laboratory facilities are provided for the radiation protection and chemistry sections. These facilities include both laboratory and calibration rooms. A health physics control station is equipped to analyze routine air samples and contamination swipe surveys. The control station also serves as a central location for portable radiation survey instruments.

"Friskers" and other type personnel monitors are located at appropriate locations as dictated by the radiation protection program.

A beta-gamma portal monitor is located at all authorized personnel exits from the radiologically controlled area as a final check on personnel leaving the controlled area.

The types of portable radiation survey instruments available for routine monitoring functions are controlled and placed by Health Physics and governed by procedures.

Survey instruments are included in a formal maintenance program to ensure that they are normally calibrated. Calibration and maintenance records are provided for each instrument.

Portable radiation survey instruments are available for use offsite during and following any possible accidental release of radioactivity from the facility. The equipment available and required are controlled by the Emergency Plan and Health Physics procedures.

### 4.2.5.5 Onsite Treatment Facilities, Equipment and Supplies

Onsite treatment facilities consist of a Decontamination Room and an Examination Room located in the Unit 1 Nuclear Services Building adjacent to the Containment Sphere but outside the external concrete biological shield. An alternate location for the treatment of injured and/or contaminated personnel and for the storage of supplies is the Medical Bureau Examination Room located in the Buchanan Service Center.

Onsite equipment and supplies for the treatment of injured and/or contaminated personnel are controlled by Health Physics Procedures and the Emergency Plan and its Implementing Procedures.

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### 4.2.5.6 Treatment Procedures and Techniques

The procedure and techniques used to treat injured and/or contaminated personnel are addressed by Health Physics procedures and the Emergency Plan and its Implementing Procedures.

### 4.2.5.7 Qualifications of Medical Personnel

Arrangements with local hospitals with qualified personnel to provide medical services for injured and/or contaminated personnel are included in the Emergency Plan and its Implementing Procedures.

Onsite Emergency Medical Technicians are certified by New York State. First Aid responders are certified by the American Red Cross, the American Heart Association or other certified First Aid / CPR training association. Health Physics technicians receive personnel decontamination training.

### 4.2.5.8 Transport of Injured Personnel

Arrangements for ambulance service to transport injured and/or contaminated personnel to local hospitals are included in the Emergency Plan and its Implementing Procedures.

### 4.2.5.9 Hospital Facilities

Arrangements with local hospitals with qualified personnel to provide medical services for injured and/or contaminated personnel are included in the Emergency Plan and its Implementing Procedures.

### 4.2.6 Evaluation of Radiation Protection

All liquid waste releases will be assayed for radioactivity to comply with the limits (one-tenth of 10 CFR 20) for unrestricted areas specified.

### 4.2.7 Tests and Inspections

The gas and particulate effluent monitors shall be tested at every two years with calibrated sources and normal response of each monitor shall be tested daily using a remotely-operated test source to verify the instruments response. Liquid effluent monitors shall be tested every two years with calibrated sources and normal response of each monitor shall be tested daily using a remotely-operated test source to verify the instruments response.

### 4.2.8 Handling and Use of Sealed Special Nuclear, Source and By-Product Material

A. Tests for leakage and / or contamination shall be performed as follows:

1. Each sealed source, with a half-life greater than thirty days, shall be tested for leakage and / or contamination at intervals not to exceed six months (see 11.2.8.A.2 for testing of sealed sources that are stored and not being used).
2. Sealed sources that are stored and not being used shall be tested for leakage prior to any use or transfer to another user unless they have been leak tested within six

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months prior to the date of use or transfer. In the absence of a certificate indicating that a test has been made within six months prior to the transfer, sealed sources shall not be put into use until tested.

- B. Sealed sources are exempt from 11.2.8.A when the source contains:
  - 1. Less than or equal to 100 microcuries of beta and / or gamma emitting material, or
  - 2. Less than or equal to 5 microcuries of alpha emitting material.
- C. The leakage test shall be capable of detecting the presence of 0.005 microcuries of radioactive material on the test sample.
- D. If the leakage test reveals the presence of 0.005 microcuries or more of removable contamination, the sealed source shall immediately be withdrawn from use and either decontaminated and repaired, or be disposed of in accordance with USNRC regulations.
- E. If the leakage test reveals the presence of 0.005 microcuries or more of removable contamination, a special report shall be prepared and submitted to the Commission within 30 days.

### BIBLIOGRAPHY FOR SECTION 4.2

Comprehensive Public Water Supply Study for the New York City of New York and County of Westchester, Report CPWS-27, (submitted by Metcalf and Eddy, Hazen and Sawyer, and Malcolm Pirnie Engineers to the New York State Department of Health), August 1967.

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**TABLE 4.2-1**  
**Fuel Handling Shield Design Parameters**

Maximum dose rate adjacent to spent fuel pit	0.75 mrem/hr
Maximum dose rate at water surface	2.0 mrem/hr

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**TABLE 4.2-2**  
**Principal Auxiliary Shielding**

<u>Component</u>	<u>Concrete Shield Thickness</u>
Demineralizers	4-ft - 0-in.
Gas decay tanks	3-ft - 6-in.
Gas compressor	2-ft - 0-in.
Motor control centers and support equipment	1-ft – 0-in.
Design parameters for the auxiliary shielding include:	
Dose rate outside auxiliary building	0.75 mrem/hr
Dose rate in the building outside shield walls	0.75 mrem/hr

Notes:

1. This represents shielding minimum for the panels. The panels themselves contain 7 in. lead shot shielding sandwiched between two steel plates. The base of the panels (up to a height of 2-ft 9-in.) is also shielded by lead shot shielding sandwiched between two steel plates.

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**TABLE 4.2-3**  
**Radiation Monitoring Channel Data**

<u>Effluent Monitors</u>		
<u>Channel</u>	<u>Stream Monitored</u>	<u>Normal Maximum Channel Output</u>
R-27	Wide Range Gas	1.0 x 10 <sup>5</sup> uCi/cc
R-39	Service Water from Component Cooling Heat Exchangers	1.0 x 10 <sup>7</sup> CPM
R-40		1.0 x 10 <sup>7</sup> CPM
R-44	Plant Vent Air Gaseous	1.0 x 10 <sup>7</sup> CPM
R-54	Liquid Waste Effluent	1.0 x 10 <sup>7</sup> CPM
R-60	Unit 1 Stack Air Gaseous	1.0 x 10 <sup>7</sup> CPM
R-62	Unit 1 Sphere Foundation Drain Sump	1.0 x 10 <sup>7</sup> CPM
<u>Process Monitors</u>		
<u>Channel</u>	<u>Stream Monitored</u>	<u>Normal Maximum Channel Output</u>
R-47	Component Cooling Water	1.0 x 10 <sup>7</sup> CPM
R-38-1	Control Room Air Intake	1.0 x 10 <sup>3</sup> mR/hr
R-38-2		1.0 x 10 <sup>3</sup> mR/hr
<u>Area Monitors</u>		
<u>Channel</u>	<u>Stream Monitored</u>	<u>Normal Maximum Channel Output</u>
R-1	Control Room	1.0 x 10 <sup>4</sup> mR/hr
R-5	Spent Fuel Building	1.0 x 10 <sup>4</sup> mR/hr

Note: Radiation monitors listed as Effluent Radiation Monitors in Table 4.2-3 and not specifically listed in Technical Requirements Manual Table 3.3.G-1, ODCM Table D 3.3.1-1, ODCM Table D 3.3.2-1, or Unit 1 Technical Specifications Section 5.2.5 will continue to maintain surveillance requirements imposed by ODCM Table D 3.3.1-1 or ODCM Table D 3.3.2-1 for daily, monthly, quarterly and biennial frequencies.

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### Appendix 4B – HISTORICAL INFORMATION

#### DETERMINATION OF RIVER WATER DILUTION FACTORS BETWEEN THE INDIAN POINT SITE AND THE NEAREST PUBLIC DRINKING WATER INTAKES

##### LIST OF TABLES

###### Table and Title

- 4B-1 Concentrations of Primary Coolant Isotopes to the Hudson River at Indian Point and Chelsea

##### LIST OF FIGURES

###### Figure and Title

- 4B-1 Iodine-131 Concentration vs Days After Burst Release From Indian Point for 1 Curie Release
- 4B-2 Iodine-131 Concentration at Chelsea vs Days After Burst Release From Indian Point for 1 Curie Release
- 4B-3 Maximum Concentration vs Distance Upstream for 1 Curie Release
- 4B-4 Maximum Concentration at Chelsea vs Half-Life for 1 Curie Release
- 4B-5 Time to Reach Peak Concentration at Chelsea vs Half-Life for 1 Curie Release

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### Appendix 4B – HISTORICAL INFORMATION

#### DETERMINATION OF RIVER WATER DILUTION FACTORS BETWEEN THE INDIAN POINT SITE AND THE NEAREST PUBLIC DRINKING WATER INTAKES

The analytical techniques used to analyze the dispersion of continuous and burst releases of liquids are discussed in detail in "Transport of Contaminants in the Hudson River above Indian Point Station," which is referenced in Section 2.5.

There are two potential sources of drinking water in the Hudson River, namely, New York City's Chelsea Pumping Station and the Castle Point Veteran's Hospital. The city of New York's Chelsea Pumping Station is located about 1 mile north of Chelsea, New York, on the east bank of the Hudson River. The pumping station is 22 miles upriver from Indian Point measured along the centerline of the river. The Castle Point Veteran's Hospital is a relatively small intake located approximately 21 miles upriver from the proposed site.

Analyses have been conducted to determine the difference in concentration at Chelsea and Castle Point Veteran's Hospital. The difference in concentration is small; hence, the discussion of the potential intake, namely, Chelsea, is sufficient. (See Reference 3 of Section 2.5 for continuous and burst releases.)

The River drought conditions analyzed have been characterized in terms of salinity because the operation of the Chelsea Station is dependent on the level of salt at the station. Consider the following five drought conditions, i.e., salinities at Chelsea:

Salt Concentration in ppm		Runoff (cfs)	Dispersion Coefficient (Square miles/day)
At Chelsea	At Indian Point		
200	2300	5000	5.24
300	2800	4600	5.28
500	4000	4400	5.43
1000	5500	4000	6.00
2000	7000	3500	7.16

The first two drought conditions correspond to concentrations of salinity at Chelsea, at which the New York City Department of Water Resources would begin to be concerned about using Chelsea for New York City's water supply.

The third condition, a salinity of 500 ppm, corresponds to the "midthousand" level, which might constitute the maximum level at which Chelsea operation would be stopped. This also corresponds to the Public Health Service drinking water standard for total dissolved solids.

The fourth condition, a salinity of 1000 ppm, represents the maximum level at which Chelsea operation would be stopped.

The fifth condition, a salinity of 2000 ppm, corresponds to the highest levels of salinity known to have occurred at Chelsea and represents the most conservative river conditions used in this analysis. This concentration of salinity at Chelsea was reached in late November 1964 at the end of 6 months of Hudson River low flows. Support that the 1964 drought was the worst on record



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after regulation of the Hudson River is given in a recent report concerning the potential of the Hudson River supplementing New York City's water supply system.\*

*[Note - "Comprehensive Public Water Supply Study for the New York City of New York and County of Westchester" - Report CPWS-27 submitted by Metcalf and Eddy, Hazen and Sawyer, and Malcolm Pirnie Engineers to the New York State Department of Health, August 1967.]*

The upstream movement of salt is the result of a rather delicate balance, which is struck between the salinity-induced density currents, which tend to drive the salt itself up the estuary, and fresh water flow, which tends to hold back the salt movement. The river's dispersion characteristics are strongly influenced by this phenomenon, so that salinity profiles become the chief means of estimating the longitudinal dispersion coefficient in the river.

Calculation of dispersion coefficients requires a knowledge of the salinity changes between two fixed points and the river's flow. The essential point, however, is that the behavior of a conservative substance is identical to the salt behavior, which is well-defined; hence, the salinity at Chelsea is an excellent indicator of the upstream movement of any pollutant introduced to the river below the station. This is explained as follows:

1. If salt is not present at Chelsea, then neither will any other pollutant, discharged many miles below Chelsea, be present at Chelsea.
2. When salt is present at Chelsea, the ratio between the salt concentrations at Indian Point and Chelsea is a measure of the "mechanical dilution," i.e., dilution due to the river's flow and dispersion characteristics for non-decaying pollutants.

Hence, for the five drought conditions cited above, the mechanical dilution factors between Indian Point Station and Chelsea may be obtained directly from the ratio of salinity at these two points and are as follows:

Runoff (cfs)	Mechanical Dilution
5000	11.5
4600	9.4
4400	8.0
4000	5.5
3500	3.5

To obtain the concentrations of decaying radionuclides at Chelsea, simple ratios of the salt concentrations at Indian Point and Chelsea are not used. Rather, a material balance on each isotope is struck over any segment of the river by considering the transport mechanisms of net flow and longitudinal dispersion, and the radioactive decay mechanism. The longitudinal dispersion coefficient is obtained from salt profiles. The approach is described in the reference cited above in Section 2.5.

To show how the significant parameters, namely, the salinity and the half-life affect the river's ability to reduce concentration of introduced pollutants, a study was made assuming a normalized continuous release rate for each isotope of 1 Ci/day and a normalized burst release for each isotope of 1 Ci. Since the concentrations at Chelsea are directly proportional to the source term, the normalized curves can be used to determine quickly the concentration at Chelsea due to a known burst or continuous release from Indian Point, or to determine dilution factors.

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### Continuous Release

A hypothetical case where primary coolant with 1-percent failed fuel being released directly to the discharge canal was considered so that the behavior of all isotopes of possible concern in the river could be presented. The activity is released at a constant rate, the value of which is set so that the MPC of the mix will not be exceeded in the discharge water. The most severe drought conditions have been utilized; for the continuous release, these consist of a long-term steady upstream runoff of 3500 cfs, which causes the salt concentration at Chelsea to reach 2000 ppm.

Other pertinent river parameters used in the analysis are as follows:

1. Longitudinal dispersion coefficient, "E" = 7.16 mi<sup>2</sup>/day
2. Average cross-sectional area, "A" = 140,000-ft<sup>2</sup>

The results of this analysis are presented in Table 4B-1 and the computational procedure follows:

1. Column 1 - Unit 3 PSAR, Column 2, Part B, Table 16 (E-3.1).
2. Column 2 - 0.693 divided by half-life in days.
3. Column 3 - allowable release rate based on MPC of mix in discharge canal.
4. Column 4 through 7 - computation procedure for continuous release, QL and M report to Con Edison on Chelsea concentrations (May 1966), and included in both Units 2 and 3 submittals. (Analyses appended to Section 2.5.)
5. Column 8 - concentration at Chelsea divided by concentration at Indian Point.

The minimum dilution factors for all isotopes of concern are given in column 8 of Table 4B-1.

For the effect of all three units at Indian Point releasing radioactivity to the river under the conditions described above, the corresponding Chelsea and Indian Point concentrations can be computed by multiplying the concentrations in these tables by 1,960,000/840,000 or 2.34, the ratio of the total condenser flow to the Units 2 or 3 condenser flow. This assumes that the mix distribution from each unit is the same.

### Burst Release

The results of the normalized burst release studies are presented in Figures 4B-1 through 4B-5. They are based on a 1 Ci burst release of each isotope. The following conclusions can be reached from these Figures.

1. Referring to Figure 4B-1, the peak concentrations at Chelsea and Castle Point are for the purpose of this discussion essentially the same.
2. Referring to Figure 4B-2, variations in drought conditions, i.e., changes in low runoff values do not appreciably affect the peak concentrations at Chelsea.
3. Referring to Figure 4B-5, the runoff does not appreciably affect the time for an isotope to reach a peak concentration at Chelsea; the time to the peak is a weak function of half-life for isotopes with half-lives less than 100 days, and the time to the peak is not sensitive to half-life for isotopes with half-lives greater than 100 days.

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4. Referring to Figures 4B-3 and 4B-4, short-lived (less than 1 day) isotopes will not reach Chelsea; peak concentrations of intermediate isotopes (1 day to 100 days) are strongly dependent on the half-life.

The river dilution factor between Indian Point and Chelsea for the burst release is a nonapplicable concept. When the maximum radioactivity effect of each isotope occurs at Chelsea, the corresponding concentration of that isotope at Indian Point will be very low. Furthermore, Chelsea will not see the maximum concentration of each isotope at the same time. For these reasons, for the burst release, the concentration in the Hudson River is considered for Indian Point one-half day after the release and at Chelsea at the time when the concentration of the given isotope is maximum at that point. Zero time cannot be used at Indian Point because the equations used will yield infinity for the concentration at  $x = 0$ ,  $t = 0$ . One-half day later was used because this corresponds to one tidal cycle, the minimum time necessary to provide the river mixing, which these equations presume.

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TABLE 4B-1 (Sheet 1 of 2)  
Concentrations of Primary Coolant Isotopes in the  
Hudson River at Indian Point and Chelsea

Hypothetical Continuous Release, One Percent Failed Fuel  
MPC in Discharge Canal

(1)	(2)	(3)	(4)	(5)	Behavior At		(7)	(8)
Isotope	Decay Rate (day <sup>-1</sup> )	Discharge Rate (μCi/day)	Indian Point		Chelsea		Fraction of MPC	River Dilution Between Indian Point - Chelsea
			Concentration (μCi/ml)	Fraction of MPC	Concentration (μCi/ml)			
Mn-54	2.3x10 <sup>-3</sup>	1.54 x10 <sup>2</sup>	15.25x10 <sup>-12</sup>	1.5x10 <sup>-7</sup>	3.99x10 <sup>-12</sup>	3.99x10 <sup>-8</sup>	3.82	
Mn-56	6.3	3.33x10 <sup>4</sup>	118.5x10 <sup>-12</sup>	1.2x10 <sup>-6</sup>	5.5x10 <sup>-20</sup>	5.5x10 <sup>-16</sup>	2.16x10 <sup>9</sup>	
Co-58	0.97x10 <sup>-2</sup>	4.62x10 <sup>3</sup>	332x10 <sup>-12</sup>	3.3x10 <sup>-6</sup>	6.35x10 <sup>-11</sup>	5.35x10 <sup>-7</sup>	5.22	
Fe-59	1.5x10 <sup>-2</sup>	1.07x10 <sup>2</sup>	6.77x10 <sup>-12</sup>	1.1x10 <sup>-7</sup>	1.05x10 <sup>-12</sup>	1.75x10 <sup>-8</sup>	6.45	
Co-69	3.6x10 <sup>-4</sup>	5.45x10 <sup>2</sup>	61.8x10 <sup>-12</sup>	1.2x10 <sup>-6</sup>	1.73x10 <sup>-11</sup>	3.45x10 <sup>-7</sup>	3.58	
Br-84	3.15x10 <sup>-3</sup>	1.63x10 <sup>4</sup>	1530x10 <sup>-12</sup>	-	-	-	-	
Rb-88	5.6x10 <sup>-3</sup>	1.54x10 <sup>4</sup>	1.28x10 <sup>-7</sup>	-	-	-	-	
Rb-89	6.48x10 <sup>-3</sup>	3.56x10 <sup>4</sup>	2870x10 <sup>-12</sup>	-	-	-	-	
Sr-89	1.37x10 <sup>-2</sup>	1.20x10 <sup>3</sup>	76.4x10 <sup>-12</sup>	2.5x10 <sup>-5</sup>	1.25x10 <sup>-11</sup>	4.28x10 <sup>-6</sup>	6.11	
Sr-90	0.69x10 <sup>-4</sup>	0.81x10 <sup>2</sup>	9.35x10 <sup>-12</sup>	3.1x10 <sup>-5</sup>	2.68x10 <sup>-12</sup>	8.92x10 <sup>-6</sup>	3.49	
Y-90	2.6x10 <sup>-4</sup>	1.66x10 <sup>2</sup>	2.88x10 <sup>-12</sup>	1.4x10 <sup>-7</sup>	2.24x10 <sup>-14</sup>	1.12x10 <sup>-9</sup>	352	
Sr-91	1.73	7.82x10 <sup>2</sup>	5.32x10 <sup>-12</sup>	0.8x10 <sup>-7</sup>	6.1x10 <sup>-17</sup>	8.70x10 <sup>-13</sup>	8.72x10 <sup>4</sup>	
Y-91	1.2x10 <sup>-2</sup>	3.56x10 <sup>2</sup>	23.9x10 <sup>-12</sup>	8x10 <sup>-7</sup>	4.27x10 <sup>-12</sup>	1.34x10 <sup>-7</sup>	5.60	
Mo-99	2.5x10 <sup>-1</sup>	1.96x10 <sup>6</sup>	3.47x10 <sup>-8</sup>	1.7x10 <sup>-4</sup>	2.84x10 <sup>-10</sup>	1.42x10 <sup>-6</sup>	122	

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TABLE 4B-1 (Sheet 2 of 2)  
Concentrations of Primary Coolant Isotopes in the  
Hudson River at Indian Point and Chelsea

Hypothetical Continuous Release, One Percent Failed Fuel  
MPC in Discharge Canal

(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)
Isotope	Decay Rate (day <sup>-1</sup> )	Discharge Rate ( $\mu$ Ci/day)	Behavior At			Chelsea Concentration ( $\mu$ Ci/ml)	River Dilution Between Indian Point - Chelsea
			Indian Point		Fraction of MPC		
			Concentration ( $\mu$ Ci/ml)	Fraction of MPC			
I-131	8.62x10 <sup>-2</sup>	1.04x10 <sup>6</sup>	3.07x10 <sup>-8</sup>	1x10 <sup>-1</sup>	1.35x10 <sup>-9</sup>	4.5x10 <sup>-3</sup>	22.7
Te-132	0.9x10 <sup>-2</sup>	1.10x10 <sup>5</sup>	8.08x10 <sup>-9</sup>	2.7x10 <sup>-4</sup>	2.38x10 <sup>-12</sup>	7.94x10 <sup>-7</sup>	3400
I-132	7.2	3.56x10 <sup>5</sup>	1.18x10 <sup>-9</sup>	1.5x10 <sup>-4</sup>	1.63x10 <sup>-19</sup>	2.03x10 <sup>-14</sup>	7.25x10 <sup>9</sup>
I-133	0.81	8.05x10 <sup>5</sup>	7.97x10 <sup>-9</sup>	8x10 <sup>-3</sup>	2.82x10 <sup>-12</sup>	2.82x10 <sup>-6</sup>	2830
Te-134	23	1.16x10 <sup>4</sup>	21.6x10 <sup>-12</sup>	-	-	-	-
I-134	19	2.12x10 <sup>5</sup>	4.34x10 <sup>-10</sup>	2.2x10 <sup>-5</sup>	7.70x10 <sup>-26</sup>	3.85x10 <sup>-21</sup>	5.64x10 <sup>15</sup>
Cs-134	0.93x10 <sup>-3</sup>	1.36x10 <sup>5</sup>	1.47x10 <sup>-8</sup>	1.6x10 <sup>-3</sup>	4.01x10 <sup>-9</sup>	4.46x10 <sup>-4</sup>	3.67
I-135	2.39	8.05x10 <sup>5</sup>	4.58x10 <sup>-9</sup>	1.1x10 <sup>-3</sup>	5.88x10 <sup>-15</sup>	1.47x10 <sup>-9</sup>	7.8x10 <sup>5</sup>
Cs-136	5.14x10 <sup>-2</sup>	1.32x10 <sup>4</sup>	4.95x10 <sup>-10</sup>	6x10 <sup>-6</sup>	3.49x10 <sup>-11</sup>	3.88x10 <sup>-7</sup>	14.2
Cs-137	6.3x10 <sup>-4</sup>	5.76x10 <sup>5</sup>	6.34x10 <sup>-8</sup>	3.2x10 <sup>-3</sup>	1.91x10 <sup>-8</sup>	9.55x10 <sup>-4</sup>	3.32
Cs-138	32	2.62x10 <sup>4</sup>	41.8x10 <sup>-12</sup>	-	-	-	-
Ba-140	5.4x10 <sup>-2</sup>	3.56x10 <sup>2</sup>	12.1x10 <sup>-12</sup>	4x10 <sup>-7</sup>	9.09x10 <sup>-13</sup>	3.03x10 <sup>-8</sup>	13.3
La-140	0.415	3.70x10 <sup>2</sup>	5.1x10 <sup>-12</sup>	2.5x10 <sup>-7</sup>	1.33x10 <sup>-14</sup>	6.65x10 <sup>-10</sup>	384
Ce-144	2.44x10 <sup>-3</sup>	1.25x10 <sup>3</sup>	122.5x10 <sup>-12</sup>	1.2x10 <sup>-5</sup>	3.05x10 <sup>-11</sup>	3.05x10 <sup>-6</sup>	4.02
Pr-144	5.13x10 <sup>-2</sup>	1.37x10 <sup>6</sup>	5.13x10 <sup>-8</sup>	-	-	-	-
Tritium	1.49x10 <sup>6</sup>	1.49x10 <sup>6</sup>	1.74x10 <sup>-7</sup>	5.8x10 <sup>-5</sup>	4.75x10 <sup>-8</sup>	1.59x10 <sup>-5</sup>	3.66

Total 9.15x10<sup>6</sup>

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### 4B FIGURES

Figure No.	Title
Figure 4B-1	Iodine-131 Concentration vs Days After Burst Release From Indian Point for 1 Curie Release
Figure 4B-2	Iodine-131 Concentration vs Chelsea vs Days After Burst Release From Indian Point for 1 Curie Release
Figure 4B-3	Maximum Concentration vs Distance Upstream for 1 Curie Release
Figure 4B-4	Maximum Concentration at Chelsea vs Half-Life for 1 Curie Release
Figure 4B-5	Time to Reach Peak Concentration at Chelsea vs Half-Life for 1 Curie Release

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## CHAPTER 5 CONDUCT OF FACILITY ACTIVITIES

### 5.1 Organization and Responsibility

Operation and maintenance of the Indian Point Unit 2 facility is the responsibility of the Holtec Decommissioning International, LLC (HDI) organization. The management organization and functional responsibilities as they relate to the operation and maintenance of the Indian Point facility are discussed in DSAR Section 1.6.3 and in the HDI Decommissioning Quality Assurance Program (DQAP).

#### 5.1.1 Facility Staff

The corporate officer with direct responsibility for the facility shall have corporate responsible for the safe storage and handling of nuclear fuel and shall take any measures needed to ensure acceptable performance of the staff in maintaining and providing technical support to the facility to ensure safe management of nuclear fuel.

The Site Vice President is responsible for overall facility operation and has control over those onsite activities necessary for storage and maintenance of nuclear fuel.

The facility organization, duty shift composition, control room occupancy, and other requirements for site personnel are in accordance with the Technical Specifications.

An incipient fire brigade (incipient responder) is maintained on the site at all times. The organization, operation and training of the incipient fire brigade (incipient responders) is discussed in the document under separate cover entitled, "IPEC Fire Protection Program Plan."

#### 5.1.2 Facility Staff Qualifications

Each member of the facility staff meets or exceeds the minimum qualifications of ANSI / ANS-3.1-1978.

Certified Fuel Handlers shall be trained in accordance with the NRC approved training and retraining program for Certified Fuel Handlers.

### 5.2 Training

A retraining and replacement training program for the facility staff shall be maintained.

An NRC approved training and retraining program for Certified Fuel Handlers shall be maintained.

The training program for the incipient fire brigade (incipient responder) is described in the document under separate cover entitled, "IPEC Fire Protection Program Plan."

An emergency plan training program is maintained to cover licensee and non-licensee individuals or groups assigned to the various functional areas of emergency activity.

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Radiation protection training is given to personnel requiring unescorted access to controlled areas of the facility.

Training requirements for the security force are set forth in the "Indian Point, Physical Security, Training and Qualification, Safeguard Contingency Plan, and Independent Spent Fuel Storage Installation Program."

### 5.3 Written Procedures

Written procedures and administrative policies are established, implemented, and maintained in accordance with the DQAP, the IP2 Renewed Facility License, and the Appendices A through C Technical Specifications.

#### 5.3.1 Emergency Plan Implementing Procedures

Emergency plan implementing procedures (EPIPs) provide instructions and outline responsibilities of on-site and off-site personnel for the IPEC facility in the event of an emergency at IP2.

### 5.4 Records

Records concerning facility activities, including historical operations, are maintained in the form of logbooks, charts, and other such internal reports as may be needed to document pertinent facility conditions. The principal logs to be maintained by the shift manager and other site personnel, as applicable. These logs include descriptions of the facility conditions that exist at the time, descriptions of significant operational efforts accomplished during the shift, and such facility events or circumstances as are deemed pertinent to maintain proper continuity of knowledge and understanding of such matters as responsibility in those areas is passed on from shift to shift.

A record of radiation safety conditions, internal and environmental, is maintained in the form of appropriate log entries, and continuous recording chart information in those functional systems and areas provided with radiation survey instruments. In addition, Radiation Work Permit survey information provides the necessary record of radiation exposure conditions prior to job commencement. Actual personnel radiation exposure information is maintained. Records of controlled radiation releases to the environment are maintained by site personnel, and all necessary information describing specific radioactivity concentrations, total volumes released, along with any dilution requirements, are entered on the Radioactive Waste Release Permit prepared for each release.

All abnormal occurrences that occur during the course of facility activities are recorded in the shift manager's logbook and, where appropriate, in the logbooks maintained by site personnel.

Facility modification records (e.g., procedures, drawings, specifications) are maintained on file.

Detailed records of total uranium, U-235, Pu-239, and Pu-241 for all fuel in use or in storage are maintained. Records of fuel transfers are maintained via proper execution of NRC forms. Specific locations for all fuel assemblies in the reactor core or in the fuel storage pools are maintained on appropriate core or fuel storage pool arrangement drawings.



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Record maintenance and retention is in accordance with the requirements of the DQAP. Records are maintained on paper, microfilm/aperture cards, or optical disk storage media. Procedures for maintenance of optical disk records comply with the guidance of NRC Generic Letter 88-18 "Plant Record Storage on Optical Disks."

### 5.5 Review and Audit of Facility Activities

Matters such as design changes to the facility which require a license amendment, changes to facility procedures, or changes to the Technical Specifications, are conducted in accordance with the requirements of 10 CFR 50 and the DQAP. Two committees assist in the review of safety-related items. These committees (i.e., the On-Site Safety Review Committee and the Safety Review Committee) function in accordance with the requirements of the DQAP.

A continuing review of facility activities is performed by the facility staff and at the executive level.

#### 5.5.1 On-Site Safety Review Committee (OSRC)

The On-Site Safety Review Committee functions to advise on all matters related to nuclear safety in accordance with the requirements of the DQAP.

#### 5.5.2 Safety Review Committee (SRC)

The Safety Review Committee functions to provide independent review and audit of designated activities and facility activities in accordance with the requirements of the DQAP.

#### 5.5.3 Qualification of Inspection, Examination, Testing, and Audit Personnel

HDI's commitments and exceptions related to the qualification of inspection, examination, testing, and audit personnel are described in the DQAP.

### 5.6 Plant Security

The program for ensuring the physical security of the Indian Point Unit 2 station has been reviewed by the NRC and found acceptable. The fully implemented security plan provides the protection needed to meet the general performance requirements of 10 CFR 73.55(a) and the objectives of the specific requirements of 10 CFR 73.55, paragraphs (b) through (k), without impairing the ability to operate the facility safely. The approved facility security program, titled "Indian Point, Physical Security, Training and Qualification, Safeguards Contingency Plan, and Independent Spent Fuel Storage Installation Security Program," is addressed in the 10 CFR 50 facility license. The approved security plan documents and the NRC Security Plan Evaluation Report have been withheld from public disclosure pursuant to 10 CFR 2.390(d).

Access to Indian Point Units 1, 2 and 3 areas for all persons is controlled under approved procedures administered by the Station Security Department.

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### 5.7 Emergency Plan

In accordance with 10 CFR 50.54(q), the Indian Point Energy Center (IPEC) Emergency Plan (Plan) outlines the basis for response actions that would be implemented in an emergency. Detailed Plan implementing procedures are maintained separately and used to guide those responsible for implementing emergency actions. This plan documents the methods by which IPEC's Emergency Preparedness Program meets the criteria set forth in 10 CFR 50.47(b) and Appendix E.

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## CHAPTER 6 SAFETY ANALYSIS

### 6.1 Introduction

On May 12, 2020, the licensee certified to the NRC that IP2 had both permanently ceased operations (final shutdown 4/30/2020) and that all fuel had been removed from the reactor vessel and placed in the spent fuel pit (SFP) (Reference 6.6-1). Since IP2 will never again enter any operational mode, reactor related accidents, abnormal operational transients, and special events are no longer a possibility.

This chapter discusses: (a) a postulated fuel handling accident (FHA) associated with fuel movement until the fuel has been transferred to the Independent Spent Fuel Storage Installation (ISFSI), (b) accident release-recycle of waste liquid, (c) accidental release of waste gas, and (d) the postulated drop of a high integrity container (HIC) containing radioactive resins. Bounding conditions, conservatism in equipment design, conformance to high standards of material and construction, the control of loads and strict administrative controls over facility operations all serve to assure the integrity of the fuel while in the SFP and during fuel transfer to the ISFSI.

Accidents involving fuel and the storage system utilized at the ISFSI are discussed in the storage system Final Safety Analysis Report.

New hazards, new initiators or new accidents that may challenge offsite guideline exposures, may be introduced as a result of certain decommissioning activities. These issues will be evaluated when the scope and type of decommissioning activities are finalized.

### 6.2 Fuel-Handling Accidents

The possibility of a fuel-handling incident is very remote because of the many administrative controls and physical limitations imposed on fuel-handling operations. The fuel-handling manipulators and hoists are designed so that fuel cannot be raised above a position that provides adequate shield water depth for the safety of personnel. This safety feature applies to handling facilities in the spent fuel pit area. In the spent fuel pit, the design of storage racks and manipulation facilities is such that:

1. Fuel at rest is positioned by positive restraints in an eversafe, always subcritical, geometrical array. Even if an assembly is not placed in the correct location, sub-criticality is ensured because a minimum boron concentration of 2000 ppm is required at all times in the pool.
2. Fuel can be manipulated only one assembly at a time.
3. Violation of procedures by placing one fuel assembly in juxtaposition with any group of assemblies in racks will not result in criticality.

In addition, administrative controls do not permit the handling of heavy objects above the fuel racks under conditions specified in the Technical Requirements Manual.

Adequate cooling of fuel during underwater handling is provided by convective heat transfer to the surrounding water. The fuel assembly is immersed continuously while in the spent fuel pit.

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The fuel-handling equipment is described in detail in Section 3.5. Special precautions are taken in all fuel-handling operations to minimize the possibility of damage to fuel assemblies. All handling operations on irradiated fuel are conducted under water. The handling tools used in the fuel-handling operations are conservatively designed, and the associated devices are of a fail-safe design.

In the fuel storage area, the fuel assemblies from Unit 2 and Unit 3 are spaced in a pattern that prevents any possibility of a criticality accident. As required by 10 CFR 50.68, "Criticality Accident Requirements," if the spent fuel pit takes credit for soluble boron, then "the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 per cent probability, 95 percent confidence level, if flooded with unborated water. Northeast Technology Corporation report NET-28091-003-01, "Criticality Safety Analysis for the Indian Point Unit 2 Spent Fuel Pool with No Adsorber Panel Credit" and Northeast Technology Corporation report NET-173-02, "Indian Point Unit 2 Spent Fuel Pool (SFP) Boron Dilution Analysis," determined that 10 CFR 50.68(b)(4) will be met during normal SFP operation and all credible accident scenarios if: a) spent fuel pit boron concentration is maintained within the Technical Specification limits and, b) fuel assembly storage location within the spent fuel pit is restricted based on the fuel assembly's initial enrichment, burnup, decay of  $\text{Pu}^{241}$  (i.e., cooling time) and number of Integral Fuel Burnable Absorbers (IFBA) rods. Note that no Boraflex is credited in Northeast Technology Corporation report NET-28091-003-01. As such there is no need to continue the Boraflex monitoring program.

Northeast Technology Corporation report NET-28091-003-01 also evaluated credible abnormal occurrences in accordance with ANSI/ANS-57.2-1983. This evaluation considered the effects of the following: a) a dropped fuel assembly or an assembly placed alongside a rack; b) a misloaded fuel assembly; c) abnormal heat loads; and, d) multiple misloads. Northeast Technology Corporation report NET-28091-003-01 determined that the SFP will maintain a keff of  $\leq 0.95$  under the worst-case accident scenario if the SFP is filled with a soluble boron concentration of  $\geq 1495$  ppm.

Therefore, Northeast Technology Corporation report NET-28091-003-01 confirmed that the requirements in 10 CFR 50.68, "Criticality Accident Requirements," will be met for both normal SFP operation and credible abnormal occurrences if:

- a) Spent Fuel Pit boron concentration is maintained within the limits Technical Specifications, and;
- b) Fuel assembly storage location within the spent fuel pit is restricted in accordance with Technical Specifications based on the fuel assembly's initial enrichment, burnup, decay of Plutonium-241 (i.e. cooling time), and number of Integral Fuel Burnable Absorbers (IFBA) rods.

Northeast Technology Corporation report NET-173-02 evaluated postulated unplanned SFP boron dilution scenarios assuming an initial SFP boron concentration within the Technical Specification limit. The evaluation considered various scenarios by which the SFP boron concentration may be diluted and the time available before the minimum boron concentration necessary to ensure subcriticality for the non-accident condition (i.e. it is not assumed an assembly is misloaded concurrent with the spent fuel pit dilution event). Northeast Technology

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Corporation report NET-173-02 determined that an unplanned or inadvertent event that could dilute the SFP boron concentration from 2000 ppm to 786 ppm is not a credible event because of the low frequency of postulated initiating events and because the event would be readily detected and mitigated by plant personnel through alarms, flooding, and operator rounds through the SFP area.

The motions of the cranes that move the fuel assemblies are limited to a low maximum speed. Caution is exercised during fuel handling to prevent a fuel assembly from striking another fuel assembly or structures in the fuel storage building.

The fuel-handling equipment suspends the fuel assembly in the vertical position during fuel movements.

All these safety features and precautions make the probability of a fuel handling incident very low. Nevertheless, since it is possible that a fuel assembly could be dropped during the handling operations, the radiological consequences of such an incident were evaluated.

Sections 6.2.1 and 6.2.2 specifically address evaluations performed for the following accidents:

1. Fuel-handling accident in the fuel storage building.
2. Fuel-handling cask drop accident.

### 6.2.1 Fuel-Handling Accident in Fuel Storage Building

An FHA may occur in the Fuel Storage Building (FSB) during movement of a fuel assembly. The fuel assembly is moved under water and the accident is assumed to occur when one fuel assembly is damaged. The fission product activity present in the fuel gap of all of the fuel pins in the damaged fuel assembly is released to the spent fuel pool while the FSB exhaust fan is not operating.

The source term and basic assumptions for evaluating the Total Effective Dose Equivalent (TEDE) doses associated with a postulated FHA during refueling were selected to be consistent with Regulatory Guide 1.183 (Reference 6.6-2). The IP2 Control Room (CR) doses were evaluated assuming that the CR ventilation systems were in the normal operation mode for 30 days (i.e., for the entire accident duration).

The radiation fields from external sources including overhead radioactive clouds were calculated. This contribution applies only during the 0 - 2 hours period over which the release was assumed to occur (Reference 6.6-2). The radiation fields from external sources are discussed in Section 6.2.1.7.

#### 6.2.1.1 Method of Analysis

Post-accident FHA radiation fields and exposures in the IP2 CR, Exclusion Area Boundary (EAB), and Low Population Zone (LPZ) were computed using the following:

- a) The methodology and assumptions in Regulatory Guide 1.183 (Reference 6.6-2)

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- b) Appropriate source terms, release pathways, and other assumptions, as described in Table 6.2-1,
- c) Post-accident atmospheric dispersion factors ( $\chi/Q_s$ ), and
- d) The NRC sponsored computer code RADTRAD, Revision 3.03 (Reference 6.6-3) was used to model the design basis FHA and estimate the dose consequences. The CR, EAB, and LPZ doses in terms of TEDE were calculated for the FHA.

Calculations IP-CALC-11-00073 and IP-CALC-11-00074 (References 6.6-4 and 6.25) contain a case for 84 hours of decay that models a ground level release from the limiting FSB surface. The “Normal Mode” case in these calculations does not credit FSB filtration, the high-rad alarm, or dispersion from the FSB ventilation system. This case also does not credit any CR isolation or emergency filtration.

To determine the time following permanent shut down required for the dose at the EAB to be < 1 rem (i.e., the Environmental Protection Agency (EPA) Protective Action Guidelines (PAGs), Reference 6.6-6, Section 2.2) following the FHA, the “Normal Mode” RADTRAD 3.03 model from the IP3 FHA (IP-CALC-11-00074, Reference 6.6-5) is used. The IP3 model used as the EAB dose consequences bounds the IP2 model from IP-CALC-11-00073 (Reference 6.6-4) as shown from the dose consequences presented in References 6.6-4 and 6.6-5. The decay period in this RADTRAD model, originally 84 hours, is increased until the resulting EAB TEDE dose is less than 1 rem.

### 6.2.1.2 Fission Product Inventory

The fission product inventory in the core is based on full power operation (3216 MWt + 2% uncertainty, i.e., 3280.3 MWt). The core inventory of radionuclides of interest at 84 hours decay is shown in Table 6.2-2 (Reference 6.6-7).

### 6.2.1.3 Release Fractions and Composition

The fission product gap release fractions, for each radionuclide group for the DBA FHA are shown below:

I-131	0.12
Kr-85	0.30
Other iodines and noble gases	0.10

The values for Kr-85 and the other iodines and noble gases are from Regulatory Guide 1.25 (Reference 6.6-8). The I-131 value originates in Reference 6.6-9. Note that there are lower values identified in Table 3 of Regulatory Guide 1.183 (Reference 6.6-2) but these cannot be used because the conditions for their use (specified in footnote 11 in Regulatory Guide 1.183) cannot be assured.

Appendix B of Regulatory Guide 1.183 specifies that the iodine released from the fuel rods is 95% cesium iodine, 4.85% elemental and 0.15% organic. It also states that it should be assumed that

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all the cesium iodine is converted instantaneously to elemental iodine. Therefore, the iodine species release should be 99.85% (95% + 4.85%) elemental and 0.15% organic.

Appendix B of Regulatory Guide 1.183 also states that if the depth of water above the damaged fuel is 23 feet or greater, the overall effective decontamination factor (DF) is 200 (per the IP2 Technical Specifications, there are requirements for  $\geq 23$  feet of water above the stored spent fuel). Therefore, from the following equation the DF of 285 for elemental iodine (based on an overall DF of 200) can be calculated:

$$100 / [(99.85 / x) + 0.15] = 200$$

Where, x is the DF for elemental iodine and is calculated to be 285. The fraction of the iodine released from the water pool that is in the elemental form thus becomes:

$$99.85/285 = 0.35 \text{ elemental and } 0.15/1 = 0.15 \text{ organic}$$

The elemental fraction is:

$$0.35 \text{ (elemental released)} / (0.35 + 0.15) \text{ (total released, elemental and organic)} = 0.7$$

and the organic fraction is:

$$0.15 \text{ (organic released)} / (0.35 + 0.15) \text{ (total released, elemental and organic)} = 0.3$$

These values (70% elemental and 30% organic) were used in the RADTRAD computer code.

### 6.2.1.4 Control Room Dose Consequences

For the IP2 CR, the TEDE analysis should consider all sources of radiation that will cause exposure to CR personnel (Reference 6.6-10) specifically:

- Internal radiation in the CR atmosphere by the intake of airborne radioactive material contained in the radioactive plume released from the accident.
- External Cloud exposure.

The  $\chi/Q$ s associated with the transport of released radioactivity to the CR intake are as follows:

#### IP2 CR $\chi/Q$ s from FSB Releases (Reference 6.6-11)

Interval	Release Location	
0 – 2 hrs	FSB Surface	FSB Door
$\chi/Q$ (sec/m <sup>3</sup> )	8.31E-04	5.39E-04

Note: Without the FSB exhaust fan operating, the CR  $\chi/Q$  value from the FSB surface release is higher than that from the FSB door release. Therefore, the CR TEDE dose was only calculated for an FSB Surface (i.e., FSB roof) release.



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The atmospheric dispersion factors for the application of site meteorology for Indian Point Unit 2 FHA analysis in the FSB were calculated using the meteorological data consisted of 4 years-worth of onsite hourly values. The meteorological data obtained during calendar years 2007 through 2010. Data were presented as hourly average and were representative of overall site conditions and were free from local effects such as building and cooling tower wakes, brush and vegetation. The 4 years of hourly data used in the atmospheric dispersion factors assessment were enough to reflect long-term site-specific meteorological trends.

Since releases are assumed to be completed in the first 2 hours [Regulatory Guide 1.183 (Reference 6.6-2)], no additional time periods are presented.

The CR characteristics are as follows:

Free air volume	102,400 ft <sup>3</sup> (Reference 6.6-12)
Normal Operation Air Flows	(Reference 6.6-12)
Filtered makeup	0 cfm
Filtered recirculation	0 cfm
Unfiltered makeup	920 cfm
Unfiltered inleakage	700 cfm
Intake filter efficiencies	Elemental iodine 95%, methyl (organic) iodines 90%, and particulates 99%

The breathing rate was set at 3.5E-04 (m<sup>3</sup>/sec) for the duration of the accident (Reference 6.6-2, Sec. 4.2.6). Regulatory Guide 1.183 requires 100% occupancy of the CR during only the first 24 hours of a postulated accident; for days 2, 3 and 4 the occupancy factor is reduced to 60%, and for periods beyond 4 days 40% occupancy is allowed (Reference 6.6-2, Sec. 4.2.6).

### 6.2.1.5 Offsite Dose Consequences

The  $\chi/Q_s$  associated with the transport of released radioactivity to the offsite outdoor receptors are as follows (Reference 6.6-12):

#### $\chi/Q_s$ Ground Level Released

Interval	0-2 hours	Interval	0-8 hours
$\chi/Q$ (sec/m <sup>3</sup> )	(EAB) 7.50-04	$\chi/Q$ (sec/m <sup>3</sup> )	(LPZ) 3.5E-04

The breathing rate at the various receptors of interest for the duration of the accident is as follows (Reference 6.6-2, Sec. 4.1.3):

Receptor Location	Time Interval (hours)	Breathing Rate (m <sup>3</sup> /second)
EAB	0 - 2	3.5E-4
LPZ	0 - 8	3.5E-4
	8 - 24	1.75E-4
	24 - 720	2.32E-4



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### 6.2.1.6 Activity Released from the SFP

To find the activity released from the water pool, the following calculation is used:

Activity =	$\frac{[\text{Core Activity (Ci)} \times \text{Radial Peaking Factor} \times \text{Gap Fraction}]}{[\text{Decontamination Factor} \times \text{Number of Fuel Assemblies}]}$
------------	--

Example: I-131 activity at 84 hours =  $6.94\text{E}+07 \times 1.7 \times 0.12 / (200 \times 193) = 3.67\text{E}+02$  Ci

For all other halogens, Decontamination Factor is also 200.

Example: I-130 activity at 84 hours =  $3.44\text{E}+04 \times 1.7 \times 0.10 / (200 \times 193) = 1.52\text{E}-01$  Ci

Example: Kr-85 activity at 84 hours =  $1.10\text{E}+06 \times 1.7 \times 0.30 / (1 \times 193) = 2.91\text{E}+03$  Ci

For all others noble gases, Decontamination Factor is one.

Example: Xe-131m activity at 84 hours =  $9.85\text{E}+05 \times 1.7 \times 0.10 / (1 \times 193) = 8.68\text{E}+02$  Ci

The activity released from the spent fuel pool for an accident occurring 84 hours after shutdown is shown in Table 6.2-3. All leakage is immediately released to the environment from the FSB without holdup, plate-out or dilution.

The radial peaking factor is from Reference 6.6-12.

The CR, EAB and LPZ radiation exposures following a design-basis FHA were calculated using the RADTRAD 3.03 computer code and the data and assumptions listed above. Copies of the RADTRAD 3.03 inputs and outputs are provided in References 6.6-4 and 6.6-13.

### 6.2.1.7 Gamma Radiation from External Sources

In addition to the dose calculated above from the activity entering the CR, there are dose contributions to the operators from the gamma radiation emanating by the cloud of activity around the CR. The dose contribution to the CR operators from the cloud external to the CR was determined using the LPZ TEDE doses from Reference 6.6-14 and adjusting those doses by the ratio of the  $\chi/Q$  associated with the CR intake to the LPZ  $\chi/Q$  and applying the shielding factor which is equivalent to 0.5 inch thickness of steel.

The ratio of CR  $\chi/Q$  to LPZ  $\chi/Q$  is:  $8.31\text{E}-04 / 3.5\text{E}-04 = 2.37$

Therefore, the dose contribution to the IP2 CR operators from the cloud external would be:

$2.630\text{E}-02$  (from Reference 6.6-14)  $\times [2.37 / 1.84$  (from Reference 6.6-14)] =  $3.388\text{E}-02$  rem

### 6.2.1.8 Time Frame for Doses to be Less than the EPA PAG

Using the IP3 RADTRAD model from IP-CALC-11-00074 (Reference 6.6-5), a Decay Period of 636 hours was determined to reduce the EAB dose to below the EPA PAG of < 1 Rem. When

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added to the original 84 hours that the accident source term is based on, results in a total time following permanent shut down of 720 hours or 30 days.

### 6.2.1.9 Results

The IP2 CR, EAB and LPZ doses were calculated in the event a design basis FHA occurs in the FSB without the FSB Exhaust Fan. The CR doses were calculated without credit for CR filtration (normal mode). The following table shows the summary of results. The results indicate that the EAB, LPZ and CR doses are within the allowable limits established in 10 CFR 50.67 and Regulatory Guide 1.183.

**IP2 FHA - CR, EAB, and LPZ TEDE Doses (at 84 Hours Decay)**

Location	Dose (Rem)	TEDE Limit (Rem)
CR	3.88	5.0
EAB	4.2	6.3
LPZ	2.0	6.3

Following a total decay time of 720 hours or 30 days, the TEDE dose at the EAB from a potential FHA is 0.47 Rem. This dose is below the 1 rem PAG limit with considerable margin. While the results could be refined to reduce the decay time even further, this is not required. The zirconium fire calculation requires a decay time of 16.5 months (Reference 6.6-15). The TEDE dose at the EAB for a potential FHA was also evaluated following 15 months of decay in accordance with the zirconium fire window. The TEDE dose for this FHA was calculated to be 1.23 mrem.

### 6.2.2 Fuel Cask Drop Accident

As discussed in Sections 3.5.5.4, 3.5.5.5, and 3.5.6.1, the IP2 fuel storage building spent fuel cask handling operations are now conducted using a single-failure-proof 110-Ton Ederer Gantry Crane that conforms to the requirements in NUREG-0554 (Single-Failure-Proof Cranes for Nuclear Power Plants, May 1979). The Ederer Gantry Crane performs spent fuel cask handling activities without the necessity of having to postulate the drop of a spent fuel cask. With the Ederer crane's 110-ton main hoist qualified as single-failure-proof, the crane is used as part of a single-failure-proof handling system for critical lifts as discussed in Revision 1 of Section 9.1.5, Overhead Heavy Load Handling Systems, Sub-section III.4.C of NUREG-0800, and a cask drop accident is not a credible event and need not be postulated. Even though the IP2 fuel storage building 40-ton bridge crane is no longer used for spent fuel cask handling, the following fuel cask drop accident provisions and results are being retained since the analysis bounds other drop accidents that may be postulated in the fuel storage building and cask loading pit even though a cask drop accident is no longer credible.

Performing an evaluation using the analysis assumptions for the fuel-handling accident shows that even with damage to a full core of recently discharged fuel assemblies by a fuel cask dropped into the spent fuel pool, the calculated fuel-handling accident doses would not be exceeded if 90 days had elapsed after shutdown. Since the fuel cask is handled by the single failure proof 110-ton gantry crane approved for use by the NRC in Technical Specification Amendment #244, this accident is not probable. Additional protections making this accident highly improbable are

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outlined below. In addition, Technical Requirements Manual Sections B 3.9.C and 3.9.E preclude movement of a spent fuel cask over any spent fuel storage racks.

During normal operation, if a spent fuel cask were placed in or removed from its position in the spent fuel pit, limit switches on the rails and logic built into the single-failure-proof control system would not allow for the cask to be moved any farther north or east of the spot reserved for the cask in the pit.

It is extremely improbable that the cask would be inadvertently or otherwise dropped during the process of transfer. This is due to the following provisions:

1. Conservative design margins used for the cask-related handling equipment (crane, rigging, hooks, etc.).
2. Periodic nondestructive equipment tests and inspection procedures.
3. Use of qualified crane operators and riggers.
4. Use of approved operating and administrative procedures.

These provisions will be rigorously met so that the inadvertent drop of the cask into the pool is highly improbable. However, should such a highly unlikely accident occur, the basic assumptions for analysis are as follows:

1. The drop would be from the highest position of the cask, which is 5-ft above the water surface and 43-ft above the bottom of the pool.
2. The cask is fully loaded and weighs 40 tons.

The results of the analysis indicate that the cask would hit the bottom of the pit with a velocity of approximately 40-ft/sec, assuming a conservative drag coefficient of 0.5. In comparison, the cask would have reached a velocity of 52-ft/sec if dropped through 43-ft in air.

Using the Ballistic Research Laboratories formula for the penetration of missiles in steel, the depth of penetration of the cask into the 1-in. wear plate covering the 1/4-in. pit liner plate would be 0.35-in., assuming the cask struck the wear plate while in a perfectly vertical position. In the event that the cask falls through the water at an angle, terminal velocity of the cask would be somewhat less because of the increased drag. However, the cask would strike the wear plate with an initial line contact and would penetrate the wear plate and the pit liner plate, causing some cracking of the concrete below. This reinforced concrete is a minimum of 3-ft thick and rests on solid rock.

Water would initially flow through the punctured liner plate and fill the cracks in the concrete. Since the pit is founded on solid rock and since the bottom of the pit is approximately 24 feet below the surrounding grade, very little water can be lost from the pit. The capacity of the makeup demineralized water supply to the pit is 150 gpm. In addition, the spent fuel pit cooling system piping has a 4-in. flange connection for temporary cooling and/or makeup water.

Because the bottom of the spent fuel pit is 24-ft below grade and no equipment areas are in the vicinity, there can be no flooding of other areas with subsequent damage to equipment.

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**Table 6.2-1**  
**Fuel Handling Accident - Design Input Data\***

Parameter	Data
Plant Power	Section 6.2.1.2
Core Inventories	Table 6.2-2
Activity Released from the SFP	Table 6.2-2
Fission Product Gap Fraction	Section 6.2.1.3
Decay Period (hours)	84
Amount of Fuel Damage	1 assembly
Radial Peaking Factor	1.70
Duration of Releases	2 hours
Water Depth	23 feet
Iodine Decontamination Factor	Section 6.2.1.3
Chemical Form Release	Section 6.2.1.3
CR $\chi/Q$	Section 6.2.1.3
CR Free Air Volume	102,400 ft <sup>3</sup>
CR Parameters	Section 6.2.1.4
CR Breathing Rate	Section 6.2.1.4
CR Occupancy Factors	Section 6.2.1.4
Offsite $\chi/Q$	Section 6.2.1.5
Offsite Breathing Rate	Section 6.2.1.5
Operation of the CR HVAC	Normal Operation
FSB Ventilation System	None

\* Data from Reference 6.6-7

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**Table 6.2-2**  
**Core Inventories of Nuclides for use in Radiological Design-Basis Applications**  
**(at 84 Hours Decay)\***

Nuclide Halogens	Activity (Ci)		Nuclide Noble Gases	Activity (Ci)
I-130	3.44E+04		Kr-85m	0.00E+00
I-131	6.94E+07		Kr-85	1.10E+06
I-132	6.39E+07		Kr-87	0.00E+00
I-133	1.17E+07		Kr-88	0.00E+00
I-134	0.00E+00			
I-135	2.62E+04		Xe-131m	9.85E+05
			Xe-133m	2.91E+06
			Xe-133	1.36E+08
			Xe-135m	4.20E+03
			Xe-135	7.83E+05
			Xe-138	0.00E+00

\* Data from Reference 6.6-7

## IP2 DEFUELED SAFETY ANALYSIS REPORT

**Table 6.2-3**  
**Activity Released from the SFP\***

Nuclide Halogens	Activity (Ci)		Nuclide Noble Gases	Activity (Ci)
I-130	1.52E-01		Kr-85m	0.00E+00
I-131	3.67E+02		Kr-85	2.91E+03
I-132	2.81E+02		Kr-87	0.00E+00
I-133	5.15E+01		Kr-88	0.00E+00
I-134	0.00E+00			
I-135	1.15E-01		Xe-131m	8.68E+02
			Xe-133m	2.56E+03
			Xe-133	1.20E+05
			Xe-135m	3.70E+00
			Xe-135	6.90E+02
			Xe-138	0.00E+00

\* See Section 6.2.1.6 for calculation of activity released from the SFP

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### 6.3 Accidental Release - Waste Gas

The Waste Gas System has been retired. An accidental release of Waste Gas is no longer postulated.

### 6.4 Accidental Release-Recycle of Waste Liquid

Any potential liquid waste release collects in building sumps or is retained in building vaults. It is not released to the environment. As such, the hazard from these releases is derived only from any volatilized components. The volatilized components are what comprise the waste gas accident. Thus, the release of liquid waste is already evaluated in Section 6.3. Furthermore, a liquid waste release accident is not one of the accident analyses required for EP exemption as specified by ISG-02 (Reference 6.6-23). Therefore, a separate liquid-specific release accident evaluation is not required.

### 6.5 High Integrity Container Drop Event

A calculation (Reference 6.6-25) was conducted to establish the dose at the Exclusion Area Boundary (EAB) from a hypothetical High Integrity Container (HIC) drop. The subsequent release of radioactivity is less than the dose that would require declaration of a General Emergency (1 rem). Further, the dose at the EAB from a hypothetical HIC drop is less than the dose previously calculated from a Fuel Handling Accident (FHA) at 15 months after shutdown.

The HICs are located in the IP2 Alleyway between the PAB and FSB, the IP3 Annex building, and the IP1 Fuel Handling Floor. Since  $\chi/Q_s$  for IP1 are not available, a bounding atmospheric dispersion value of less than 1 sec/m<sup>3</sup> (i.e. no wind dispersion) is used for the HIC drop accident. However, aerosolized fractions will be credited. Since these casks are in dry storage, the only way for a release to occur is for the contents to become aerosolized. The bounding scenario for this accident is a HIC is dropped and the entire contents becomes engulfed in a fire. A release fraction of 0.78% (Reference 6.6-24) is utilized, this is the percentage of the contents anticipated to become aerosolized due to the fire. The total effective dose equivalent (TEDE) at the site boundary for the postulated HIC-drop with a maximum hypothetical activity of 250 Ci Cs-137 is 0.82 millirem, which is well below the level required for a General Emergency (1,000 mrem) and is below the EAB dose from a postulated release from an FHA.

## IP2 DEFUELED SAFETY ANALYSIS REPORT

### 6.6 References

1. Permanently Defueled Technical Specification License Amendment 294
2. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000
3. NUREG/CR-6604, RADTRAD, "A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," USNRC, April 1998
4. Calculation No. IP-CALC-11-00073, "AST Analysis of IP2 Fuel Handling Accident in the Fuel Storage Building without FSB Exhaust Fan Operation," Rev. 0, September 28, 2011
5. Calculation No. IP-CALC-11-00074, "AST Analysis of IP3 Fuel Handling Accident in the Fuel Storage Building without FSB Exhaust Fan Operation," Rev. 0, September 28, 2011
6. EPA-400/R-17/001, "PAG Manual: Protective Action Guides and Planning Guidance for Radiological Incidents," January 2017
7. CN-REA-03-4, "Core Radiation Sources to Support the Indian Point 2 Power Upate Project," Rev. 0, April 3, 2003
8. Regulatory Guide 1.25, "Assumptions used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," March 1972. [Historical reference]
9. NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," February 1988
10. CN-REA-03-24, "Control Room Direct Dose Based on NUREG-1465 Source Term," Revision 2, October 15, 2003
11. Calculation No. IP-CALC-11-00060, Rev. 0, "Analysis of IP2 Control Room and Technical Support Center Atmospheric Dispersion Factors due to Releases from the IP2 FSB and RWST," Revision 0
12. Entergy Letter, PU2-E-03-20, "Entergy Nuclear Northeast – Indian Point 2 – Power Uprating Program – Inputs Approved by the Technical Review Committee," April 15, 2003
13. Calculation No. IP-CALC-19-00003, "Post-Permanent Shutdown Analyses of Fuel Handling, Waste Handling, and High Integrity Container Drop Accidents for Indian Points Units 2 and 3," Rev. 1, April 28, 2022
14. CN-CRA-03-64, "Indian Point 2 – Fuel Handling Accident for Stretch Power Upate Program," Rev. 0, October 15, 2003
15. Calculation No. IP-CALC-18-00064, "IPEC Zirconium Fire Calculation," Revision 1
16. Offsite Dose Calculation Manual for Indian Point Units 1, 2 and 3, Rev. 4, August 2012



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17. Regulatory Guide 1.194, Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants, June 2003
18. Federal Guidance Report No. 12, External Exposure to Radionuclides in Air, Water, and Soil September 1993
19. Calculation No. IP-RPT-11-00025, "Analysis of Control Room and Technical Support Center X/Q Values for Releases at Indian Point Generating Station Unit Number 3 Fuel Handling Building Using the ARCON 96 Computer Code"
20. IP3 Updated Final Safety Analysis Report, Rev. 7
21. IP3-CALC-RAD-00008, "IP3 – Control Room Habitability Following a Fuel Handling Accident & a Gas Decay Tank Rupture," Rev. 0
22. Plant Drawing 9321-1002, Westinghouse Electric Corporation, Indian Point Unit 2 Drawing, "Plot Plan – UFSAR Figure No. 1.2 3," Rev. 9
23. NSIR/DPR-ISG-02, Emergency Planning Exemption Requests for Decommissioning Nuclear Power Plants, May 11, 2105 (ADAMS Accession Number ML14106A057)
24. DOE-HDBK-3010-94, "Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities, Volume 1 – Analysis of Experimental Data," December 1994 (ML13078A031)
25. Report No. IPEC-RPT-22-015 R0, "HIC Drop Accident," prepared June 2022

# **IP2 DEFUELED SAFETY ANALYSIS REPORT**

## **Appendix A**

### **LICENSE RENEWAL**

# **IP2 DEFUELED SAFETY ANALYSIS REPORT**

## **A.1 INTRODUCTION**

This appendix provides the information submitted in an Updated Final Safety Analysis Report Supplement as required by 10 CFR 54.21(d) for the Indian Point Energy Center (IPEC) License Renewal Application (LRA). The LRA contains the technical information required by 10 CFR 54.21(a) and (c). Appendix B of the IPEC LRA provides descriptions of the programs and activities that manage the effects of aging for the period of extended operation. Appendix B was used to prepare the original program and activity descriptions for the IP2 Updated Final Safety Analysis Report (UFSAR) Supplement and the applicable descriptions continue to be used for the IP2 Defueled Safety Analysis Report (DSAR) Supplement information in this appendix.

With inclusion of the Supplement in the DSAR, future changes to the descriptions of the programs and activities will be made in accordance with 10 CFR 50.59.

## IP2 DEFUELED SAFETY ANALYSIS REPORT

### A.2 AGING MANAGEMENT

The following information is integrated into the DSAR to document aging management programs and activities credited in the license renewal review.

#### A.2.0 Supplement for Renewed Operating License

The Indian Point Energy Center license renewal application (Reference A.2-1) and information in subsequent related correspondence provided sufficient basis for the NRC to make the findings required by 10 CFR 54.29 (Final Safety Evaluation Report) (Reference A.2-2). As required by 10 CFR 54.21(d), this supplement contains a summary description of the programs and activities for managing the effects of aging (Section A.2.1). After the Facility License was renewed, IP2 was permanently shut down and defueled. In addition, the Facility License was modified to denote that the renewed license would remain in effect until the Commission notifies the licensee in writing that the license is terminated. In accordance with the schedule defined in the Post-Shutdown Decommissioning Activities Report, the IP2 spent fuel will be transferred from the IP2 spent fuel pit to an Independent Spent Fuel Storage Installation.

#### A.2.1 Aging Management Programs and Activities

The integrated plant assessment for license renewal identified aging management programs necessary to provide reasonable assurance that components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis (CLB). This section describes the aging management programs and activities. All aging management programs were implemented prior to entering the period of extended operation.

IPEC and HDI quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B. The Decommissioning Quality Assurance Program (DQAP) applies to safety-related structures and components. Corrective actions and administrative (document) control for both safety-related and nonsafety-related structures and components are accomplished per the IPEC corrective action program and document control program and are applicable to all aging management programs and activities. The confirmation process is part of the corrective action program and includes reviews to assure that proposed actions are adequate, tracking and reporting of open corrective actions, and review of corrective action effectiveness. Any follow-up inspection required by the confirmation process is documented in accordance with the corrective action program. The corrective action, confirmation process, and administrative controls of the DQAP are applicable to all aging management programs and activities.

The Operating Experience Program (OEP) and the Corrective Action Program (CAP) help to assure continued effectiveness of aging management programs through evaluations of operating experience. The OEP implements the requirements of NRC NUREG-0737, "Clarification of TMI Action Plan Requirements," Section I.C.5 and evaluates site and industry operating experience for impact on IPEC. The CAP implements the requirements of 10 CFR 50, Appendix B, Criterion XVI and is used to evaluate and effect appropriate actions in response to operating experience relevant to IPEC that indicates a condition adverse to quality or a non-conformance.

#### A.2.1.2 through A.2.1.34 Not Used

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### A.2.1.35 Structures Monitoring Program

The Structures Monitoring Program is a program that performs inspections in accordance with 10 CFR 50.65 (Maintenance Rule) as addressed in Regulatory Guide 1.160 and NUMARC 93-01. Periodic inspections are used to monitor the condition of structures and structural commodities to ensure there is no loss of intended function.

The Structures Monitoring Program was enhanced to include the following.

- Guidance was added to the Structures Monitoring Program to inspect inaccessible concrete areas that are exposed by excavation for any reason. The site will also inspect inaccessible concrete areas in environments where observed conditions in accessible areas exposed to the same environment indicate that significant concrete degradation is occurring.
- Guidance to perform evaluation of groundwater samples was added to the Structures Monitoring Program. To assess the aggressiveness of groundwater to concrete, IPEC will obtain samples from at least five wells that are representative of the ground water surrounding below-grade site structures at least once every five years and perform an engineering evaluation of the results from those samples for sulfates, pH and chlorides. Additionally, to assess potential indications of spent fuel pool leakage, IPEC will sample for tritium in groundwater wells in close proximity to the IP2 spent fuel pool at least once every three months.
- Enhanced the Structures Monitoring Program to perform inspection of the degraded areas of the water control structure once every three years rather than the normal frequency of once every five years during the aging management period.

Enhancements were implemented prior to the period of extended operation.

### A 2.1.36 to A.2.2 Not Used

### A.2.3 References

- A.2-1 Letter from F. Dacimo, Indian Point Energy Center, to Document Control Desk, NRC, *License Renewal Application*, dated April 23, 2007.
- A.2-2 NRC Safety Evaluation Report (SER), *Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3*, dated October 2009 Supplement 1, dated August 2011, and Supplement 2 dated November 2014.

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## **Appendix B**

### **TECHNICAL REQUIREMENTS**

#### **MANUAL**

#### **(TRM)**

**IP2 DEFUELED SAFETY ANALYSIS REPORT**  
**LIST OF EFFECTIVE SECTIONS — TECHNICAL REQUIREMENTS**  
**MANUAL**

TRM SECTION	Rev	Page(s)	EFFECTIVE DATE
Table of Contents	2	2	09/15/2022
1.1	0	2	06/01/2020
1.2	0	1	06/01/2020
1.3	0	1	06/01/2020
1.4	0	1	06/01/2020
3.0	0	4	06/01/2020
B 3.0	0	9	06/01/2020
3.3.A	0	3	06/01/2020
B 3.3.A	0	1	06/01/2020
3.3.D	0	2	06/01/2020
B 3.3.D	0	1	06/01/2020
3.3.G	0	3	06/01/2020
B 3.3.G	0	1	06/01/2020
3.3.I	0	1	06/01/2020
B 3.3.I	0	1	06/01/2020
3.3.J	0	1	06/01/2020
B 3.3.J	0	1	06/01/2020
3.7.B	0	Deleted	04/25/2022
B 3.7.B	0	Deleted	04/25/2022
3.7.E	0	1	06/01/2020
B 3.7.E	0	2	08/30/2022
3.8.B	0	2	03/02/2022
B 3.8.B	0	3	03/02/2022
3.8.C	0	2	06/01/2020
B3.8.C	0	3	06/01/2020
3.9.C	0	1	06/01/2020
B 3.9.C	0	1	06/01/2020
3.9.D	0	1	06/01/2020
B 3.9.D	0	1	06/01/2020
3.9.E	0	1	06/01/2020
B 3.9.E	0	2	07/06/2022
3.10.A	1	Deleted	12/30/2020
B 3.10.A	1	Deleted	12/30/2020
3.10.B	1	Deleted	12/30/2020
B 3.10.B	1	Deleted	12/30/2020
3.10.C	2	Deleted	01/30/2022
B 3.10.C	2	Deleted	01/30/2022
3.11.A	0	2	06/01/2020
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5.0	0	8	08/30/2022

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## IP2 DEFUELED SAFETY ANALYSIS REPORT

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## 1.0 USE AND APPLICATION

### 1.1 Definitions

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#### - NOTES -

1. Definitions are defined in Section 1.1 of the Technical Specifications and are applicable throughout the Technical Requirements Manual (TRM) and Bases. Only definitions specific to the TRM will be defined in this section.
  2. The defined terms of this section and the Technical Specifications (TS) appear in capitalized type and are applicable throughout the TRM and the TRM Bases.
  3. When a term is defined in both the TS and the TRM, TRM definition takes precedence within the TRM and the TRM Bases.
- 

<u>Term</u>	<u>Definition</u>
ACTIONS	ACTIONS shall be that part of a Requirement that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
OPERABLE — OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified function(s) are also capable of performing the related support function(s).

## 1.0 USE AND APPLICATION

### 1.2 Logical Connectors

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Logical Connectors are discussed in Section 1.2 of the Technical Specifications and are applicable throughout the Technical Requirements Manual and Bases.

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## 1.0 USE AND APPLICATION

### 1.3 Completion Times

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Completion Times are discussed in Section 1.3 of the Technical Specifications and are applicable throughout the Technical Requirements Manual and Bases.

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## 1.0 USE AND APPLICATION

### 1.4 Frequency

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Frequency is discussed in Section 1.4 of the Technical Specifications and is applicable throughout the Technical Requirements Manual and Bases.

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## 1.0 USE AND APPLICATION

1.1 Definitions

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**- NOTES -**

1. Definitions are defined in Section 1.1 of the Technical Specifications and are applicable throughout the Technical Requirements Manual (TRM) and Bases. Only definitions specific to the TRM will be defined in this section.
  2. The defined terms of this section and the Technical Specifications (TS) appear in capitalized type and are applicable throughout the TRM and the TRM Bases.
  3. When a term is defined in both the TS and the TRM, TRM definition takes precedence within the TRM and the TRM Bases.
- 

<u>Term</u>	<u>Definition</u>
ACTIONS	ACTIONS shall be that part of a Requirement that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel FUNCTIONALITY. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an in-place qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps.
CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

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**CHANNEL OPERATIONAL  
TEST (COT)**

A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify FUNCTIONALITY of all devices in the channel required for channel FUNCTIONALITY. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints required for channel FUNCTIONALITY such that the setpoints are within the necessary range and accuracy. The COT may be performed by means of any series of sequential, overlapping, or total channel steps.

**FUNCTIONAL -  
FUNCTIONALITY**

From NRC Inspection Manual Chapter 0326, "[FUNCTIONALITY] is an attribute of an SSC(s) that is not controlled by TS. An SSC not controlled by TS is [FUNCTIONAL] or has [FUNCTIONALITY] when it is capable of performing its function(s) as set forth in the Current Licensing Basis (CLB). These CLB function(s) may include the capability to perform a necessary and related support function for an SSC(s) controlled by TS."

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## 1.0 USE AND APPLICATION

### 1.2 Logical Connectors

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Logical Connectors are discussed in Section 1.2 of the Technical Specifications and are applicable throughout the Technical Requirements Manual and Bases.

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## 1.0 USE AND APPLICATION

### 1.3 Completion Times

---

Completion Times are discussed in Section 1.3 of the Technical Specifications and are applicable throughout the Technical Requirements Manual and Bases.

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## 1.0 USE AND APPLICATION

### 1.4 Frequency

---

Frequency is discussed in Section 1.4 of the Technical Specifications and is applicable throughout the Technical Requirements Manual and Bases.

---

## B 3.0 TECHNICAL REQUIREMENTS FOR OPERATION (TRO) APPLICABILITY

### BASES

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TROs	TRO 3.0.A through TRO 3.0.E establish the general requirements applicable to all TROs in Section 3.3 and Sections 3.7 to 3.11 and apply at all times, unless otherwise stated.
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TRO 3.0.A	TRO 3.0.A establishes the Applicability statement within each individual Requirement as the requirement for when the TRO is required to be met (i.e., when the facility is in the specified conditions of the Applicability statement of each Requirement).
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TRO 3.0.B	<p>TRO 3.0.B establishes that upon discovery of a failure to meet a TRO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of a TRO are not met. This Requirement establishes that:</p>
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- a. Completion of the Required Actions within the specified Completion Times constitute compliance with a Requirement; and
- b. Completion of the Required Actions is not required when a TRO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the TRO must be met. This time limit is the Completion Time to restore a NON-FUNCTIONAL system or component to FUNCTIONAL status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, the facility must be placed in a condition in which the Requirement is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit the facility activity to continue that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable justification for continued activity.

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## BASES

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TRO 3.0.B (continued)

Completing the Required Actions is not required when a TRO is met or is no longer applicable, unless otherwise stated in the individual Requirement.

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual TRO's ACTIONS specify the Required Actions where this is the case.

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of TRSs, preventive maintenance, corrective maintenance, or investigation of facility problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for convenience. Additionally, if intentional entry into ACTIONS would result in redundant equipment being NON-FUNCTIONAL, alternatives should be used instead. Doing so limits the time both subsystems/trains of a function are NON-FUNCTIONAL and limits the time conditions exist which may result in TRO 3.0.C being entered. Individual Requirements may specify a time limit for performing a TRS when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

When a change in specified condition is required to comply with Required Actions, the facility may enter a specified condition in which another Requirement becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Requirement becomes applicable and the ACTIONS Condition(s) are entered.

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TRO 3.0.C

TRO 3.0.C establishes the actions that must be implemented when a TRO is not met and:

- a. An associated Required Action and Completion Time is not met and no other Condition applies; or

## BASES

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TRO 3.0.C (continued)

- b. The condition of the facility is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the facility. Sometimes, possible combinations of Conditions are such that entering TRO 3.0.C is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that TRO 3.0.C be entered immediately.

This TRO delineates the time limits for placing the facility in a safe condition when the facility cannot be maintained within the limits as defined by the TRO and its action. It is not intended to be used as a convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being NON-FUNCTIONAL.

Upon entering TRO 3.0.C, 1 hour is allowed to initiate action to implement appropriate compensatory actions, to verify the unit is not in an unanalyzed condition, and to verify that a required safety function is not compromised. Within 12 hours, the Plant Manager or Decomm Manager's approval of the compensatory actions and the plan for exiting TRO 3.0.C must be obtained. The use and interpretation of specific times to complete the actions of TRO 3.0.C are consistent with the discussion of Section 1.3, Completion Times.

The actions required in accordance with TRO 3.0.C may be terminated and TRO 3.0.C exited if any of the following occurs:

- a. The TRO is now met;
- b. A Condition exists for which the Required Actions have now been performed; or
- c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time TRO 3.0.C is exited.

Exceptions to TRO 3.0.C are addressed in the individual Requirements.

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BASES

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## TRO 3.0.D

TRO 3.0.D establishes limitations on changes in specified conditions in the Applicability when a TRO is not met. It allows placing the facility in a specified condition stated in that Applicability (e.g., the Applicability desired to be entered) when facility conditions are such that the requirements of the TRO would not be met, in accordance with TRO 3.0.D.a or TRO 3.0.D.b.

TRO 3.0.D.a allows entry into a specified condition in the Applicability with the TRO not met when the associated ACTIONS to be entered following entry into the specified condition in the Applicability will permit continued action within the other specified condition for an unlimited period of time. Compliance with Required Actions that permit facility activities to continue for an unlimited period of time in a specified condition provides an acceptable level of safety. This is without regard to the status of the facility before or after the specified condition change. Therefore, in such cases, entry into a specified condition in the Applicability may be made in accordance with the provisions of the Required Actions.

TRO 3.0.D.b allows entry into a specified condition in the Applicability with the TRO not met based on a Note in the Requirement which states TRO 3.0.D.b is applicable. These specific allowances permit entry into specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for facility activities to continue for an unlimited period of time. This allowance may apply to all the ACTIONS or to a specific Required Action of a Requirement. For this reason, TRO 3.0.D.b is typically applied to Requirements which describe values and parameters.

The provisions of this Requirement should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to FUNCTIONAL status before entering an associated specified condition in the Applicability.

The provisions of TRO 3.0.D shall not prevent changes on specified conditions in the Applicability that are required to comply with ACTIONS.

Upon entry into a specified condition in the Applicability with the TRO not met, TRO 3.0.A and TRO 3.0.B require entry into the applicable Conditions and Required Actions until the Condition is resolved, until the TRO is met, or until the facility is not within the Applicability of the Technical Requirement.

Surveillances do not have to be performed on the associated NON-FUNCTIONAL equipment (or on variables outside the specified limits), as permitted by TRS 3.0.A. Therefore, utilizing TRO 3.0.D is not a violation of TRS 3.0.A or TRS 3.0.D for any surveillances that have not

BASES

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## TRO 3.0.D (continued)

been performed on NON-FUNCTIONAL equipment. However, TRSs must be met to ensure FUNCTIONALITY prior to declaring the associated equipment FUNCTIONAL (or variable within limits) and restoring compliance with the affected TRO.

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## TRO 3.0.E

TRO 3.0.E establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared NON-FUNCTIONAL to comply with ACTIONS. The sole purpose of this Requirement is to provide an exception to TRO 3.0.B (e.g., to not comply with the applicable Required Action(s)) to allow the performance of required testing to demonstrate:

- a. The FUNCTIONALITY of the equipment being returned to service;  
or
- b. The FUNCTIONALITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate FUNCTIONALITY. This Requirement does not provide time to perform any other preventive or corrective maintenance.

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## B 3.0 TECHNICAL REQUIREMENTS SURVEILLANCE (TRS) APPLICABILITY

BASES

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TRSs	TRS 3.0.A through TRS 3.0.D establish the general requirements applicable to all Surveillance Requirements in Section 3.3 and Sections 3.7 to 3.11 and apply at all times, unless otherwise stated.
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TRS 3.0.A	TRS 3.0.A establishes the requirement that TRSs must be met during the specified conditions in the Applicability for which the requirements of the TROs apply, unless otherwise specified in the individual TRSs. This TRS is to ensure that TRSs are performed to verify the FUNCTIONALITY of systems and components, and that variables are within specified limits. Failure to meet a TRS within the specified Frequency, in accordance with TRS 3.0.B, constitutes a failure to meet a TRO.
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Systems and components are assumed to be FUNCTIONAL when the associated TRSs have been met. Nothing in this TRS, however, is to be construed as implying that systems or components are FUNCTIONAL when:

- a. The systems or components are known to be NON-FUNCTIONAL, although still meeting the TRS(s); or
- b. The requirements of the TRS(s) are known not to be met between required TRS performances.

TRSs do not have to be performed when the facility is in a specified condition for which the requirements of the associated TRO are not applicable, unless otherwise specified.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given TRS. In this case, the unplanned event may be credited as fulfilling the performance of the TRS. This allowance includes those TRSs whose performance is normally precluded in a given specified condition.

TRSs, including TRSs invoked by Required Actions, do not have to be performed on NON-FUNCTIONAL equipment because the ACTIONS define the remedial measures that apply. TRSs have to be met and performed in accordance with TRS 3.0.B, prior to returning equipment to FUNCTIONAL status.

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BASES

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## TRS 3.0.A (continued)

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment FUNCTIONAL. This includes ensuring applicable TRSs are not failed and their most recent performance is in accordance with TRS 3.0.B. Post maintenance testing may not be possible in the current specified conditions in the Applicability due to the necessary facility parameters not having been established. In these situations, the equipment may be considered FUNCTIONAL provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow facility activities to proceed to a specified condition where other necessary post maintenance testing can be completed.

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## TRS 3.0.B

TRS 3.0.B permits a 25% extension of the interval specified in the Frequency. This extension facilitates TRS scheduling and considers facility conditions that may not be suitable for conducting the TRS (e.g., other ongoing TRS or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the TRS at its specified Frequency. This is based on the recognition that the most probable result of any particular TRS being performed is the verification of conformance with the TRSs. The requirements of regulations take precedence over the TRM. The TRM cannot in and of itself extend a test interval specified in the regulations.

The provisions of TRS 3.0.B are not intended to be used repeatedly merely as a convenience to extend TRS intervals beyond those specified.

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## TRS 3.0.C

TRS 3.0.C establishes the flexibility to defer declaring affected equipment NON-FUNCTIONAL or an affected variable outside the specified limits when a TRS has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time it is discovered that the TRS has not been performed in accordance with TRS 3.0.B, and not at the time that the specified frequency was not met.

This delay period provides adequate time to complete TRSs that have been missed. This delay period permits the completion of a TRS before complying with Required Actions or other remedial measures that might preclude completion of the TRS.

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## IP2 DEFUELED SAFETY ANALYSIS REPORT

TRS Applicability  
TRS B 3.0

### BASES

#### TRS 3.0.C (continued)

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The basis for this delay period includes consideration of facility conditions, adequate planning, availability of personnel, the time required to perform the TRS, the safety significance of the delay in completing the required TRS, and the recognition that the most probable result of any particular TRS being performed is the verification of conformance with the requirements. When a TRS with a Frequency based not on time intervals, but upon specified conditions, is discovered not to have been performed when specified, TRS 3.0.C allows the full delay period of up to the specified frequency to perform the TRS. However, since there is not a time interval specified, the missed TRS should be performed at the first reasonable opportunity. TRS 3.0.C provides a time limit for and allowances for, the performance of, TRSs that become applicable as a consequence of specified condition changes imposed by Required Actions.

Failure to comply with specified Frequencies for TRSs is expected to be an infrequent occurrence. Use of the delay period established by TRS 3.0.C is a flexibility which is not intended to be used as a convenience to extend TRS intervals. While up to 24 hours or the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the missed TRS will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on facility risk (from delaying the TRS as well as any facility configuration changes required) and impact on any analysis assumptions, in addition to facility conditions, planning, availability of personnel, and the time required to perform the TRS. All missed TRSs will be placed in the licensee's Corrective Action Program.

If a TRS is not completed within the allowed delay period, then the equipment is considered NON-FUNCTIONAL or the variable then is considered outside the specified limits and the Completion Times of the Required Actions for the applicable TRO Conditions begin immediately upon expiration of the delay period. If a TRS is failed within the delay period, then the equipment is NON-FUNCTIONAL, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable TRO Conditions begin immediately upon the failure of the TRS.

Completion of the TRS within the delay period allowed by this TRS, or within the Completion Time of the ACTIONS, restores compliance with TRS 3.0.A.

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#### TRS 3.0.D

TRS 3.0.D establishes the requirement that all applicable TRSs must be met before entry into a specified condition in the Applicability.

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## BASES

## TRS 3.0.D (continued)

This Requirement ensures that system and component FUNCTIONALITY requirements and variable limits are met before entry into specified conditions in the Applicability for which these system and components ensure safe continuation of facility activities. The provisions of this Requirement should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to FUNCTIONAL status before entering an associated specified condition in the Applicability.

A provision is included to allow entry into a specified condition in the Applicability when a TRO is not met due to Surveillance not being met in accordance with TRO 3.0.D.

However, in certain circumstances, failing to meet a TRS will not result in TRS 3.0.D restricting a specified condition change. When a system, subsystem, division, component, device, or variable is NON-FUNCTIONAL or outside its specified limits, the associated TRS(s) are not required to be performed, per TRS 3.0.A, which states that TRSs do not have to be performed on NON-FUNCTIONAL equipment. When equipment is NON-FUNCTIONAL, TRS 3.0.D does not apply to the associated TRS(s) since the requirement for the TRS(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in a TRS 3.0.D restriction to changing specified conditions of the Applicability. However, since the TRO is not met in this instance, TRO 3.0.D will govern any restrictions that may (or may not) apply to specified condition changes. TRS 3.0.D does not restrict changing specified conditions of the Applicability when a Surveillance has not been performed within the specified Frequency, provided the requirement to declare the TRO not met has been delayed in accordance with TRS 3.0.C.

The provisions of TRS 3.0.D shall not prevent entry into specified conditions in the Applicability that are required to comply with ACTIONS.

The precise requirements for performance of TRSs are specified such that exceptions to TRS 3.0.D are not necessary. The specific time frames and conditions necessary for meeting the TRSs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a TRS procedure require entry into the specified condition in the Applicability of the associated TRO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the TRO Applicability would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the TRS may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of TRSs' annotation is found in Section 1.4, Frequency.

### 3.0 TECHNICAL REQUIREMENTS FOR OPERATION (TRO) APPLICABILITY

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TRO 3.0.A	TROs shall be met during the MODES or other specified conditions in the Applicability, except as provided in TRO 3.0.B.
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TRO 3.0.B	<p>Upon discovery of a failure to meet a TRO, the Required Actions of the associated Conditions shall be met, except as provided in TRO 3.0.E.</p> <p>If the TRO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.</p>
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TRO 3.0.C	<p>When a TRO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, action shall be initiated within 1 hour to:</p> <ul style="list-style-type: none"> <li>a. Implement appropriate compensatory actions as needed;</li> <li>b. Verify that the facility is not in an unanalyzed condition;</li> <li>c. Verify that a required safety function is not compromised by the NON-FUNCTIONAL equipment; and</li> <li>d. Within 12 hours, obtain the Plant Manager or Decomm Manager approval of the compensatory actions and the plan for exiting TRO 3.0.C.</li> </ul> <p>Exceptions to this TRO are stated in the individual TROs.</p> <p>Where corrective measures are completed that permit facility activities to continue in accordance with the TRO or ACTIONS, completion of the actions required by TRO 3.0.C is not required.</p>
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TRO 3.0.D	<p>When a TRO is not met, entry into a specified condition in the Applicability shall only be made:</p> <ul style="list-style-type: none"> <li>a. When the associated ACTIONS to be entered permit facility activities to continue in the MODE or other specified condition in the Applicability for an unlimited period of time; or</li> </ul>
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### 3.0 TRO APPLICABILITY

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#### TRO 3.0.D (continued)

- b. When an allowance is stated in the individual value, parameter, or other requirement.

This requirement shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS.

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#### TRO 3.0.E

Equipment removed from service or declared NON-FUNCTIONAL to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its FUNCTIONALITY or the FUNCTIONALITY of other equipment. This is an exception to TRO 3.0.B for the system returned to service under administrative control to perform the testing required to demonstrate FUNCTIONALITY.

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### 3.0 TECHNICAL REQUIREMENTS SURVEILLANCE (TRS) APPLICABILITY

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TRS 3.0.A TRSs shall be met during the specified conditions in the Applicability for individual TROs, unless otherwise stated in the TRS. Failure to meet a TRS, whether such failure is experienced during the performance of the TRS or between performances of the TRS, shall be failure to meet the TRO. Failure to perform a TRS within the specified Frequency shall be failure to meet the TRO except as provided in TRS 3.0.C. TRSs do not have to be performed on NON-FUNCTIONAL equipment or variables outside specified limits.

---

TRS 3.0.B The specified Frequency for each TRS is met if the TRS is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as “once,” the above interval extension does not apply.

---

TRS 3.0.C If it is discovered that a TRS was not performed within its specified Frequency, then compliance with the requirement to declare the TRO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the TRS.

If the TRS is not performed within the delay period, the TRO must immediately be declared not met, and the applicable Condition(s) must be entered.

When the TRS is performed within the delay period and the TRS is not met, the TRO must immediately be declared not met, and the applicable Condition(s) must be entered.

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3.0 TRS APPLICABILITY

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TRS 3.0.D      Entry into a specified condition in the Applicability of a TRO shall only be made when the TRO's Surveillances have been met within their specified Frequency, except as provided by TRS 3.0.C. When a TRO is not met due to Surveillances not having been met, entry into a specified condition in the Applicability shall only be made in accordance with TRO 3.0.D.

This provision shall not prevent entry into specified conditions in the Applicability that are required to comply with ACTIONS.

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## IP2 DEFUELED SAFETY ANALYSIS REPORT

Meteorological Monitoring Instrumentation  
B TRM 3.3.A

### 3.3 INSTRUMENTATION

#### B 3.3.A Meteorological Monitoring Instrumentation

##### BASES

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FUNCTIONALITY of the meteorological monitoring system instrumentation ensures that sufficient meteorological data at the site is available for estimating potential radiation doses to the public as a result of routine or accidental releases of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, Rev. 0.

This specification ensures the FUNCTIONALITY of the meteorological monitoring instrumentation and the collection of meteorological data at the plant site. This data is used for estimating potential radiation doses to the public resulting from routine or accidental releases of radioactive materials to the atmosphere. A meteorological data collection program, as described in this specification, is necessary to meet the requirements of 10 CFR 50.36.a (a) (2), Appendix E to 10 CFR 50 and 10 CFR 51.

Meteorological data shall be summarized and reported as required for inclusion in the Radioactive Effluent Release Report pursuant to Technical Specification 5.6.3.

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## IP2 DEFUELED SAFETY ANALYSIS REPORT

Meteorological Monitoring Instrumentation  
TRM 3.3.A

### 3.3 INSTRUMENTATION

#### 3.3.A Meteorological Monitoring Instrumentation

TRO 3.3.A The meteorological monitoring instrumentation in Table 3.3.A-1 shall be FUNCTIONAL with indication of the tabulated parameters available in the control room.

APPLICABILITY: At all times.

#### ACTIONS

**- NOTE -**

Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more of the required meteorological monitoring channels NON-FUNCTIONAL.	A.1 Restore to FUNCTIONAL status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Prepare a corrective action report outlining the cause of the malfunction(s) and the plans for restoring the channel(s) to FUNCTIONAL status.	10 days

## IP2 DEFUELED SAFETY ANALYSIS REPORT

Meteorological Monitoring Instrumentation

TRM 3.3.A

### SURVEILLANCE REQUIREMENTS

**- NOTE -**

TRS 3.3.A.1 and TRS 3.3.A.2 apply to each Meteorological Monitoring Instrumentation Function in Table 3.3.A-1.

SURVEILLANCE		FREQUENCY
TRS 3.3.A.1	Perform CHANNEL CHECK.	24 hours
TRS 3.3.A.2	Perform CHANNEL CALIBRATION.	184 days

## IP2 DEFUELED SAFETY ANALYSIS REPORT

Meteorological Monitoring Instrumentation  
TRM 3.3.A

Table 3.3.A-1 (page 1 of 1)  
Meteorological Monitoring Instrumentation

INSTRUMENT	MINIMUM REQUIRED FUNCTIONAL	INSTRUMENT ACCURACY
1. Wind Speed		
a. Nominal Elevation 10m <sup>(a)</sup>	1	± 0.5 mph <sup>(b)</sup>
b. Nominal Elevation 60m	1	± 0.5 mph <sup>(b)</sup>
c. Nominal Elevation 122m	1	± 0.5 mph <sup>(b)</sup>
2. Wind Direction		
a. Nominal Elevation 10m	1	± 5°
b. Nominal Elevation 60m	1	± 5°
c. Nominal Elevation 122m	1	± 5°
3. Air Temperature Differential (Delta T)		
a. Nominal Elevation 60 – 10m	1	± 0.1°C
b. Nominal Elevation 122 – 10m	1	± 0.1°C

(a) 10 m as measured by the primary or backup meteorological tower.

(b) Starting speed of anemometer shall be < 1 mph.

## IP2 DEFUELED SAFETY ANALYSIS REPORT

Spent Fuel Storage Area Radiation Monitoring

TRM B 3.3.D

### 3.3 INSTRUMENTATION

#### B 3.3.D Spent Fuel Storage Area Radiation Monitoring

##### BASES

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Radiation levels in the spent fuel storage area are monitored by radiation monitor R-5.

---

## IP2 DEFUELED SAFETY ANALYSIS REPORT

Spent Fuel Storage Area Radiation Monitoring  
TRM 3.3.D

### 3.3 INSTRUMENTATION

#### 3.3.D Spent Fuel Storage Area Radiation Monitoring

TRO 3.3.D Radiation levels in the spent fuel storage area shall be monitored continuously.

APPLICABILITY: When irradiated fuel movement is taking place in the spent fuel storage area.

#### ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel storage area radiation monitor NON-FUNCTIONAL.	A.1 ----- <b>- NOTE -</b> Suspension of fuel movement shall not preclude completion of movement to a safe position. -----  Stop irradiated fuel movement in the spent fuel storage area.  <u>OR</u>	Immediately
	A.2  Establish alternate radiation monitoring capability in the spent fuel storage area.	Immediately

#### SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
TRS 3.3.D.1 Perform CHANNEL CHECK.	24 hours
TRS 3.3.D.2 Perform COT.	31 days

## IP2 DEFUELED SAFETY ANALYSIS REPORT

Spent Fuel Storage Area Radiation Monitoring  
TRM 3.3.D

### SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
TRS 3.3.D.3	Perform CHANNEL CALIBRATION.	24 months

### 3.3 INSTRUMENTATION

#### B 3.3.G Post Accident Monitoring (PAM) Instrumentation

##### BASES

---

No bases information is provided.

---

## IP2 DEFUELED SAFETY ANALYSIS REPORT

PAM Instrumentation  
TRM 3.3.G

### 3.3 INSTRUMENTATION

#### 3.3.G Post Accident Monitoring (PAM) Instrumentation

TRO 3.3.G PAM Instrumentation for the Function in Table 3.3.G-1 shall be FUNCTIONAL.

APPLICABILITY: At all times.

#### ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Function with required channels NON-FUNCTIONAL.	A.1 Restore required channel to FUNCTIONAL status.	7 days
B. Not Used.	B.1 Not Used.	Not Used
C. Required Action and associated Completion Time not met.	C.1 Initiate alternative method of monitoring the appropriate parameter(s).	Immediately
	<u>AND</u> C.2 Prepare a Corrective Action Program report.	14 days



## SURVEILLANCE REQUIREMENTS

**- NOTE -**

Refer to Table 3.3.G-1 to determine which TRSs apply for each Post Accident Monitoring Instrumentation Function.

SURVEILLANCE		FREQUENCY
TRS 3.3.G.1	Perform CHANNEL CHECK.	12 hours
TRS 3.3.G.2	Not Used.	Not Used
TRS 3.3.G.3	Perform CHANNEL CALIBRATION.	24 months

## IP2 DEFUELED SAFETY ANALYSIS REPORT

PAM Instrumentation  
TRM 3.3.G

Table 3.3.G-1 (page 1 of 1)  
Post Accident Monitoring

FUNCTION		REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS
1	Plant Vent Noble Gas Effluent Monitor (R-27)	1 <sup>(a)</sup>	TRS 3.3.G.1 TRS 3.3.G.3

(a) Encompass the entire channel from sensor to display where either an indicator, recorder, or alarm is acceptable.

## IP2 DEFUELED SAFETY ANALYSIS REPORT

Process Radiation Monitoring  
TRM B 3.3.I

### 3.3 INSTRUMENTATION

#### B 3.3.I Process Radiation Monitoring

##### BASES

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Process radiation monitoring includes the following monitors for the associated system:

R-39: Service Water from Component Cooling Heat Exchangers

R-40: Service Water from Component Cooling Heat Exchangers

R-47: Component Cooling Radiation

---

## IP2 DEFUELED SAFETY ANALYSIS REPORT

Process Radiation Monitoring  
TRM 3.3.I

### 3.3 INSTRUMENTATION

#### 3.3.I Process Radiation Monitoring

TRO 3.3.I Process radiation monitors shall be FUNCTIONAL.

APPLICABILITY: When the associated system is in operation.

#### ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more process radiation monitors NON-FUNCTIONAL	A.1 Initiate action to establish alternate method for monitoring affected process.	Immediately
	<u>AND</u> A.2 Initiate action to restore to FUNCTIONAL status.	Immediately

#### SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TRS 3.3.I.1	Perform CHANNEL CHECK.	24 hours
TRS 3.3.I.2	Perform COT.	31 days
TRS 3.3.I.3	Perform CHANNEL CALIBRATION.	24 months

## IP2 DEFUELED SAFETY ANALYSIS REPORT

Area Radiation Monitoring  
TRM B 3.3.J

### 3.3 INSTRUMENTATION

#### B 3.3.J Area Radiation Monitoring

##### BASES

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Area Radiation monitoring includes the following monitors for the associated system:

R-1: Control Room

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REFERENCES	1.	DSAR Table 4.2-3, "Radiation Monitoring Channel Data"
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## IP2 DEFUELED SAFETY ANALYSIS REPORT

Area Radiation Monitoring  
TRM 3.3.J

### 3.3 INSTRUMENTATION

#### 3.3.J Area Radiation Monitoring

TRO 3.3.J Area radiation monitors shall be FUNCTIONAL.

APPLICABILITY: At all times.

#### ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Radiation monitor NON-FUNCTIONAL.	A.1 Establish compensatory measures to ensure radiation protection for personnel in the area of the NON-FUNCTIONAL monitor.	Immediately
	<u>AND</u>	
	A.2 Initiate action to restore to FUNCTIONAL status.	Immediately

#### SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TRS 3.3.J.1	Perform CHANNEL CHECK of area radiation monitoring.	24 hours
TRS 3.3.J.2	Perform COT of area radiation monitoring.	31 days
TRS 3.3.J.3	Perform CHANNEL CALIBRATION of area radiation monitoring.	24 months

## IP2 DEFUELED SAFETY ANALYSIS REPORT

City Water Supply  
TRM B 3.7.E

### 3.7 FACILITY SYSTEMS

#### B 3.7.E City Water Supply

#### BASES

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**BACKGROUND** Two City water supply headers provide a water supply for both Unit 2 and Unit 3. One city water header supplies the City Water Storage Tank (CWST) through a fill valve, meter and a second fill valve. A second City water supply header provides water to the Unit 3 fire water storage tank. The CWST provides a protected water inventory because the continued supply of city water from offsite can not be guaranteed.

---

#### APPLICABLE SAFETY ANALYSES

The CWST is 42' high and is continually providing a water supply for normal facility uses while maintaining a reserve of water for postulated events. The CWST serves as an alternate source of water for Unit 2 fire fighting, and the Unit 3 CST in the event of its loss (e.g., due to a Tornado missile). The following events are considered simultaneously and their requirements constitute the need for a bounding reserve that is conservatively addressed by the minimum volume requirement of 655,000 gallons:

1. Cooling of the Unit 2 SBO / Appendix R diesel – The Appendix R / SBO diesel is a water cooled engine. The engine cooling water requirement is 205 gpm. The CWST provides the engine cooling water for a minimum of 4 hours supply of cooling water to the engine (this requires 50,000 gallons of reserve). Engine cooling can be transferred to Service Water supply, as required (see TRM for SBO / App. R diesel).
2. Fire Fighting Water Supply – The facility is committed to having a dedicated water inventory of 300,000 gallons for fire fighting.
3. Coincident users assumed to be provided water at 500 gpm for two hours. This requires 60,000 gallons of protected inventory.

---

**TRO** The CSWT must contain a reserve of >655,000 gallons to assure that Unit 2 and 3 postulated events are certain to have an adequate protected inventory of water.

---

**APPLICABILITY** At all times.  
**BASES**

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**ACTIONS** A.1

With the CWST <655,000 gallons the CWST is considered NON-FUNCTIONAL and there are 12 hours to restore the water level. This is a reasonable period of time given that restoration involves the isolation of non essential water usage to allow the city water fill line to restore the required level.

## IP2 DEFUELED SAFETY ANALYSIS REPORT

City Water Supply  
TRM B 3.7.E

### B.1 and B.2

With the CWST not restored to the required water level in 12 hours, the potential exists for the inability to provide the necessary water for Unit 2 & 3 licensing basis events. The declaration of the SBO / Appendix R diesel NON-FUNCTIONAL initiates the actions in TRM 3.8.B. The declaration of the CWST tank NON-FUNCTIONAL initiates the actions of Unit 3 TS 3.7.7.

---

### SURVEILLANCE REQUIREMENTS

#### TRS 3.7.E.1

This SR verifies that the CWST level requirements are met. The time frame is considered reasonable to assure that an adequate protected water inventory is maintained, and is on the same periodicity as the Unit 3 TS 3.7.7 surveillance of water pressure.

#### TRS 3.7.E.2

The CWST supplies water to the Unit 2 Appendix R / SBO diesel heat exchangers, and the fire header. The supply to the fire header is a preexisting requirement of the fire protection system and valves are tested as required by that program. The supply valves to the Appendix R / SBO diesel are required to be verified FUNCTIONAL to ensure they function when required.

#### TRS 3.7.E.3

Testing of the function of the altitude valves is verification that the valves all work as intended to assure continued water makeup. This can be by observation of the refill function.

#### TRS 3.7.E.4

The instrumentation necessary for operations to verify the CWST level is within the required value must be calibrated to assure sufficient accuracy.

---



## IP2 DEFUELED SAFETY ANALYSIS REPORT

City Water Supply  
TRM 3.7.E

### 3.7 FACILITY SYSTEMS

#### 3.7.E City Water Supply

TRO 3.7.E City water shall be FUNCTIONAL.

APPLICABILITY: At all times.

#### ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. City Water Storage Tank NON-FUNCTIONAL.	A.1 Restore City Water Storage Tank to FUNCTIONAL status.	12 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Declare the SBO / Appendix R Diesel NON-FUNCTIONAL.	Immediately

#### SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
TRS 3.7.E.1 Verify City Water Storage Tank maintains a water level of >655,000 gallons	12 hours
TRS 3.7.E.2 Verify City Water Storage Tank valves to required loads are FUNCTIONAL	90 days
TRS 3.7.E.3 Verify altitude valve and city water makeup valves are FUNCTIONAL	90 days
TRS 3.7.E.4 Perform channel Calibration of City Water Storage Tank level monitoring instruments	24 months

## IP2 DEFUELED SAFETY ANALYSIS REPORT

SBO / Appendix R Diesel Generator and Electrical Distribution System

TRM B 3.8.B

### 3.8 ELECTRICAL POWER

#### B 3.8.B SBO / Appendix R Diesel Generator and Electrical Distribution System

##### BASES

---

**BACKGROUND** After the 10 CFR 50.82(a)(1) and (2) certifications are docketed by the NRC, 10 CFR 50, Appendix R and 10 CFR 50.63 no longer apply, because the facility is permanently shut down and defueled. The SBO / Appendix R diesel generator will remain as a reliable and independent standby power source.

---

**TRO** The SBO / Appendix R diesel generator must be FUNCTIONAL to provide an independent source of power to Unit 2 during a loss of off-site power and to serve as a back-up for the Unit 3 SBO / Appendix R diesel generator. A FUNCTIONAL SBO / Appendix R diesel generator consists of the diesel generator, support equipment such as starting batteries, fuel oil, cooling water, as well as the electrical distribution system.

---

**APPLICABILITY** The SBO / Appendix R diesel generator must be functional at all times.

---

**ACTIONS** A. With the Appendix R diesel and/or the associated equipment required line-up to the 13.8 kV Bus NON-FUNCTIONAL, these systems must be restored to FUNCTIONAL status within 30 days. TRS 3.8.B.8 demonstrate the ability to perform the required lineups in the required time frames so that functionality is assured.

---

##### SURVEILLANCE REQUIREMENTS

###### TRS 3.8.B.1

The Appendix R diesel uses 172 gallons of fuel per hour when loaded at peak capacity. The Appendix R Diesel is assumed to be run for 72 hours. Therefore, there must be  $\geq 12,500$  gallons of usable fuel in the tank dedicated to the diesel. This fuel is normally supplied from the storage tank in Unit 1 Turbine Building. Other fuel oil may be credited where adequate time to refuel exists.

---

## IP2 DEFUELED SAFETY ANALYSIS REPORT

SBO / Appendix R Diesel Generator and Electrical Distribution System

TRM B 3.8.B

### BASES

#### SURVEILLANCE REQUIREMENTS (continued)

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##### TRS 3.8.B.2

Inspect the SBO / Appendix R diesel generator support systems, including check of the diesel fuel oil level, the closed cooling water system temperature, battery and battery charger. This surveillance is consistent with industry practice. When the battery is checked it should be looked at for unacceptable signs such as cracking, bulging, corrosion, leakage, or an electrolyte level not above the plates.

##### TRS 3.8.B.3

This Surveillance verifies the batteries are maintained at 12 V. The capability of the batteries to perform a function is established when they are used to start the SBO / Appendix R diesel every quarter.

##### TRS 3.8.B.4

This surveillance establishes that the battery charger is operating at the required parameters to support the battery.

##### TRS 3.8.B.5

Start and run the SBO / Appendix R diesel generator for a period of time sufficient to reach stable operating temperatures. DEMONSTRATE the proper operation of the output breaker. Starting and bringing the SBO / Appendix R diesel to operating conditions on a quarterly frequency is consistent with the Alternate AC Power Criteria identified in Appendix B section of NUMARC 87-00, "Guidelines and Technical Bases For NUMARC Initiatives."

##### TRS 3.8.B.6

The surveillance to sample and analyze fuel oil from dedicated bulk storage according to applicable standards meets the Alternate AC Power Criteria identified in Appendix B section B8(c) of NUMARC 87-00, "Guidelines and Technical Bases For NUMARC Initiatives." The frequency of once per year (Annual) is deemed sufficient.

##### TRS 3.8.B.7

Start the SBO / Appendix R diesel generator, line it up and load it between 2335 to 2435 kW, and run for at least 2 hours. Starting and loading the Appendix R diesel to rated capacity on a 24 month frequency is consistent with the Alternate AC Power Criteria identified in Appendix B section B10 of NUMARC 87-00, "Guidelines and Technical Bases For NUMARC Initiatives."

##### TRS 3.8.B.8

DEMONSTRATE the ability to line up and provide power within 60 minutes from the IP2 SBO / Appendix R diesel to 6.9 kV Bus 5 and Bus 6 and to IP3 Appendix R SBO diesel bus loads. Validate lineup through either the ASS breaker to the 12 FD3 and 12 RW3 switchgear or the SBOH breaker to the SBO 13.8kV – 6.9kV transformer to breaker SBOL and GT25 for Bus 5 and to GT26 for Bus 6. These demonstrations may be made through a combination of tests and simulated actions. The time to identify the necessity for the SBO / Appendix R

## IP2 DEFUELED SAFETY ANALYSIS REPORT

SBO / Appendix R Diesel Generator and Electrical Distribution System

TRM B 3.8.B

### BASES

#### SURVEILLANCE REQUIREMENTS (continued)

---

diesel must be estimated and then the time to initiate actions to line up and provide power must be demonstrated. The demonstration must include the ability to transfer from the cooling water of the CWST to the service water cooling. This transfer must be demonstrated to be made within two hours to assure adequate city water (see Reference 3). DEMONSTRATE the ability to line up and provide power to the Unit 3 Appendix R diesel loads. This demonstrates the ability to provide backup power. This demonstration may be made through a combination of tests and simulated actions.

TRS 3.8.B.9      Start the SBO / Appendix R diesel and operate in unit to test the circuitry of the governor and its ability to control the SBO / Appendix R diesel. This recognizes the separate circuitry in this mode.

---

### REFERENCES

1.      EC 5000033794, "IP2 Station Blackout and Appendix R Diesel Generator Set"
  2.      Risk Assessment for Extending the Proposed IP2 Appendix R Diesel Generator (ARDG) AOT
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-

## IP2 DEFUELED SAFETY ANALYSIS REPORT

SBO / Appendix R Diesel Generator and Electrical Distribution System

TRM 3.8.B

### 3.8 ELECTRICAL POWER

3.8.B SBO / Appendix R Diesel Generator and Electrical Distribution System

TRO 3.8.B The SBO / Appendix R Diesel Generator and Electrical Distribution System shall be FUNCTIONAL

APPLICABILITY: At all times.

#### ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SBO / Appendix R diesel generator is NON-FUNCTIONAL.	A.1 Restore the SBO / Appendix R diesel generator to FUNCTIONAL status.	30 days

#### SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
TRS 3.8.B.1	Verify the fuel oil storage tank contains $\geq 12,500$ gallons of usable fuel oil reserved for the diesel.	7 days
TRS 3.8.B.2	Visually inspect the SBO / Appendix R diesel generator support systems, including check of the diesel fuel oil level, the closed cooling water system temperature, battery and battery charger.	7 days
TRS 3.8.B.3	Verify individual battery voltage $\geq 12V$	31 days
TRS 3.8.B.4	Verify the battery charger output voltage $\geq 24V$ and output current $\leq 2Amps$	31 days

## IP2 DEFUELED SAFETY ANALYSIS REPORT

SBO / Appendix R Diesel Generator and Electrical Distribution System  
TRM 3.8.B

### SURVEILLANCE REQUIREMENTS (continued)

	<b>SURVEILLANCE</b>	<b>FREQUENCY</b>
TRS 3.8.B.5	Start and run the SBO / Appendix R diesel generator for a period of time sufficient to reach stable operating temperatures. Demonstrate proper operation of the output breaker.	92 days
TRS 3.8.B.6	Sample and analyze fuel oil for the SBO / Appendix R Diesel to ensure applicable standards are met.	Annual
TRS 3.8.B.7	Start the SBO / Appendix R diesel generator, load it between 2335 to 2435 kW, and run for at least 2 hours.	24 months
TRS 3.8.B.8	DEMONSTRATE the ability to line up and provide power from the SBO / Appendix R diesel to the Unit 2 Bus loads, and to Unit 3 Appendix R / SBO diesel bus loads within 60 minutes. Validate the ability to transfer SBO /Appendix R EDG cooling from City Water to Service Water within 2 hours.	24 months
TRS 3.8.B.9	DEMONSTRATE the governor circuitry operates properly in unit.	24 months

## IP2 DEFUELED SAFETY ANALYSIS REPORT

TSC Diesel Generator and TSC UPS  
TRM B 3.8.C

### 3.8 ELECTRICAL POWER

#### B 3.8.C Technical Support Center (TSC) Diesel Generator and TSC Plant Computer Uninterruptible Power Supply (UPS)

##### BASES

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**BACKGROUND** NUREG-0696, "Functional Criteria for Emergency Response Facilities" (Reference 1) describes the facilities and systems to be used to improve responses to emergency situations. The facilities include the Control Room, onsite Technical Support Center (TSC), onsite Operational Support Center (OSC), and nearsite Emergency Operations Facility (EOF). Data systems are the safety parameter display system (SPDS) and nuclear data link (NDL). Together, these facilities and systems make up the total Emergency Response Facilities (ERFs).

NUREG-0696 provides the following guidance: "Sufficient alternate or backup power sources shall be provided to maintain continuity of TSC functions and to immediately resume data acquisition, storage, and display of TSC data if loss of the primary TSC power sources occurs." The TSC Diesel Generator and TSC Plant Computer Battery UPS serve as these backup power sources.

The requirement to have a TSC comes from NUREG-0654 (Reference 2) and Article IV.E.8 of 10 CFR 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities" (Reference 3). NUREG-0654 requires that each licensee establishes a Technical Support Center and an onsite Operations Support Center (assembly area) in accordance with NUREG-0696, Revision 1. Article IV.E of 10 CFR 50, Appendix E requires that adequate provisions are made and described for emergency facilities and equipment. Item 8 of article IV.E requires, "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."

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**TRO** The TSC Diesel Generator and TSC Plant Computer Battery must be FUNCTIONAL to provide backup power to the TSC facility if loss of the primary TSC power source occurs.

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**APPLICABILITY** The TSC Diesel Generator and TSC Plant Computer UPS are required to be FUNCTIONAL at all times.

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## IP2 DEFUELED SAFETY ANALYSIS REPORT

TSC Diesel Generator and TSC UPS  
TRM B 3.8.C

### BASES

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#### ACTIONS     A.1

A NON-FUNCTIONAL TSC Diesel Generator or TSC Plant Computer UPS does not constitute a major loss of emergency assessment capability, and does not require notifying the NRC Operations Center via the Emergency Notification System in accordance with 10CFR50.72 (b)(1)(v). As noted in NUREG-0696, the TSC is one of the facilities that make up the total emergency response facilities (ERFs). Code of Federal Regulations 10CFR50.72 (b)(3)(xiii) requires an eight hour report for any event that results in a major loss of emergency assessment capability, or communications capability (e.g., significant portion of control room indication, Emergency Notification System, or offsite notification system). There is no corresponding Part 50.73 requirement. Therefore, no Licensee Event Report is required.

#### B.1

Operations tracks equipment FUNCTIONALITY and action statements. When Condition B is entered, a CR must be written and a CA issued to evaluate return to FUNCTIONALITY. The 7 day completion time was chosen because it is assumed that for the first 6 days, efforts were concentrated on returning the equipment to FUNCTIONAL. Seven additional days is sufficient time to issue the CR and CA. At the end of the additional seven days, the availability goal for the TSC data system has been exceeded by two times. The CA should address the impact that the NON-FUNCTIONAL condition has had upon the availability goal.

---

#### SURVEILLANCE REQUIREMENTS

##### TRS 3.8.C.1

A check of battery bank float voltage, cell electrolyte level, and battery physical condition is performed to ensure overall battery condition is adequate to support the TSC Diesel Generator. This program is implemented in accordance with PT-M67. Battery Bank voltage  $\geq$  24 VDC Electrolyte level Between Min & Max Physical Condition Satisfactory.

##### TRS 3.8.C.2

The TSC Diesel Generator is started and run unloaded for a minimum time to inspect diesel systems to ensure engine will start for availability. This requirement is implemented in accordance with PT-M67.



## IP2 DEFUELED SAFETY ANALYSIS REPORT

TSC Diesel Generator and TSC UPS  
TRM B 3.8.C

### BASES

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#### SURVEILLANCE REQUIREMENTS (continued)

##### TRS 3.8.C.3

The TSC Diesel Generator is started and run under load to verify generator operation and output breaker operation. This requirement is implemented in accordance with 2-PT-A046.

##### TRS 3.8.C.4

≥2500 Gallons of Fuel Oil in Ignition Tank #11 is reserved for the TSC Diesel Operation. This TRS is satisfied by the Operator rover rounds which verifies there is > 4250 gallons of fuel oil in ignition tank #11, 2500 gallons of which is reserved for operation of the TSC diesel for 48 hours. The 48 hrs. of operating time that 2500 gallons of fuel affords is sufficient time to bring in an alternate supply of fuel oil by tanker truck as required ref. SOP 29.19.

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#### REFERENCES

1. NUREG-0696, "Functional Criteria for Emergency Response Facilities," Published February 1981.
  2. NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants."
  3. Code of Federal Regulations 10 CFR 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities."
  4. NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73."
-

## IP2 DEFUELED SAFETY ANALYSIS REPORT

TSC Diesel Generator and TSC UPS  
TRM 3.8.C

### 3.8 ELECTRICAL POWER

#### 3.8.C Technical Support Center (TSC) Diesel Generator and TSC Plant Computer Uninterruptible Power Supply (UPS)

TRO 3.8.C The TSC Diesel Generator and TSC Plant Computer UPS shall be FUNCTIONAL.

APPLICABILITY: At all times.

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#### - NOTE -

TRO 3.0.C is not applicable.

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#### ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. TSC Diesel Generator or TSC Plant Computer UPS NON-FUNCTIONAL.	A.1 Restore TSC Diesel Generator and TSC Plant Computer UPS to FUNCTIONAL.	6 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Issue a CR and, in response to a CA, prepare a corrective action report outlining the cause of the NON-FUNCTIONAL required equipment, the extent of condition, and the plans and schedule for restoring the NON-FUNCTIONAL equipment to FUNCTIONAL status.	7 days

## IP2 DEFUELED SAFETY ANALYSIS REPORT

TSC Diesel Generator and TSC UPS  
TRM 3.8.C

### SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TRS 3.8.C.1	Inspect TSC Diesel Generator Battery. Battery Bank Voltage $\geq$ 24 VDC Electrolyte level between Min & Max Physical Condition Satisfactory.	31 days
TRS 3.8.C.2	Verify TSC Diesel Generator starts.	92 days
TRS 3.8.C.3	Verify TSC Diesel Generator starts and accepts load.	12 months
TRS 3.8.C.4	$\geq$ 2500 Gallons of Fuel Oil in Ignition Tank #11 Reserved for the TSC Diesel Operation.	7 days

## IP2 DEFUELED SAFETY ANALYSIS REPORT

Fuel Storage and Operations With Irradiated Fuel in the Spent Fuel Pit  
TRM B 3.9.C

### 3.9 SPENT FUEL PIT OPERATIONS

#### B 3.9.C Fuel Storage and Operations With Irradiated Fuel in the Spent Fuel Pit

##### BASES

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The spent fuel cask shall only be moved over the spent fuel pit using the Ederer 110 ton single failure proof gantry crane approved by the NRC under License Amendment #224. Any load in excess of the nominal weight of a spent fuel storage rack and associated handling tool shall not be moved on or above El. 95' in the Fuel Storage Building unless handled by the single failure proof 110 ton gantry crane. Loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool shall not be moved over spent fuel in the spent fuel pit. The weight of installed crane systems shall not be considered part of these loads.

---

##### SURVEILLANCE REQUIREMENTS

##### TRS 3.9.C.1

Verifying spent fuel pit level every 30 days is acceptable based on observation of the Control Room annunciators. The spent fuel pit Control Room alarm is set greater than 24 feet above the top of the fuel assemblies. Therefore, the absence of the alarm provides assurance the spent fuel pit level is adequate, and the additional foot of level provides time to increase level to meet the 23 feet requirement.

---

## IP2 DEFUELED SAFETY ANALYSIS REPORT

Fuel Storage and Operations With Irradiated Fuel in the Spent Fuel Pit  
TRM 3.9.C

### 3.9 SPENT FUEL PIT OPERATIONS

#### 3.9.C Fuel Storage and Operations With Irradiated Fuel in the Spent Fuel Pit

TRO 3.9.C Spent Fuel Pit water level shall be  $\geq 23$  ft over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Anytime the spent fuel pit contains irradiated fuel.

#### ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel water level less than required.	A.1 Suspend all movement of fuel assemblies in the spent fuel storage pit.	Immediately
	<u>AND</u>	
	A.2 Suspend crane operations with loads over the spent fuel in the spent fuel pit.	Immediately
	<u>AND</u>	
	A.3 Restore water level to within limit.	4 hours

#### SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
TRS 3.9.C.1 Verify spent fuel pit level $\geq 23$ ft above the top of irradiated fuel.	30 days

### 3.9 SPENT FUEL PIT OPERATIONS

#### B 3.9.D Operations In The Spent Fuel Storage Area

##### BASES

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##### SURVEILLANCE REQUIREMENTS

##### TRS 3.9.D.1

The load assumed by the refueling crane for this test must be equal to or greater than the maximum load to be assumed by the crane during the refueling operation.

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## IP2 DEFUELED SAFETY ANALYSIS REPORT

Refueling Operations  
TRM 3.9.D

### 3.9 SPENT FUEL PIT OPERATIONS

#### 3.9.D Operations In The Spent Fuel Storage Area

TRO 3.9.D The spent fuel bridge refueling crane shall be FUNCTIONAL.

APPLICABILITY: When fuel or heavy loads are being moved in the spent fuel storage area.

#### ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The spent fuel bridge refueling crane is NON-FUNCTIONAL.	<p>-----</p> <p><b>- NOTE -</b></p> <p>Suspension of operations shall not preclude completion of movement of components to a safe conservative position.</p> <p>-----</p>	
	A.1 Suspend all operations.	Immediately

#### SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TRS 3.9.D.1	Perform dead-load test with a load equal to or greater than the maximum load to be assumed, on the spent fuel pit bridge refueling crane.	Once prior to movement of fuel or heavy loads.
TRS 3.9.D.2	<p>-----</p> <p><b>- NOTE -</b></p> <p>To be performed after TRS 3.9.D.1 is complete.</p> <p>-----</p> <p>Visually inspect the spent fuel pit bridge refueling crane.</p>	Once prior to movement of fuel or heavy loads.

## IP2 DEFUELED SAFETY ANALYSIS REPORT

Ederer Gantry Crane  
TRM B 3.9.E

### 3.9 SPENT FUEL PIT OPERATIONS

#### B 3.9.E Ederer Gantry Crane

##### BASES

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##### BACKGROUND

License Amendment #244 allowed the use of a new single-failure-proof crane for moving spent fuel casks up to 110 tons in weight (when fully loaded with fuel) into and out of the spent fuel pit. These are to allow transfer of spent fuel to the independent spent fuel storage installation (ISFSI). The Holtec HI-STORM® 100 cask system has been selected for use at the ISFSI. The HI-STORM® cask system utilizes the HI-TRAC®100 or HI-TRAC Version MS transfer cask for transporting a multi-purpose canister (MPC) from the spent fuel pit, and for inter-cask MPC transfers required for on-site storage.

The amendment allows the use of the 110-ton design rated gantry crane to move spent fuel casks up to 110 tons into and out of the spent fuel pit by lifting a fully loaded Holtec HI-TRAC® 100 or HI-TRAC Version MS spent fuel transfer cask and its associated components. The existing 40-ton non-single-failure-proof overhead crane, located in the IP2 fuel storage building (FSB), does not have the capacity to handle the HI-TRAC® 100 or HI-TRAC Version MS transfer cask and its associated components, but will remain in place after the installation of the new crane. However, this crane is restricted from handling casks over spent fuel in the spent fuel pit and will only be utilized for other loading activities in the FSB.

The gantry crane main hoist has a capacity of 110 tons maximum critical load (MCL) to handle the HI-TRAC100® or HI-TRAC Version MS transfer cask, while an auxiliary hoist rated at 45 tons MCL will be used to handle ancillary components associated with the HI-STORM® 100 cask system. The crane will not be inadvertently used for unintended purposes (e.g. lifting fuel elements from the spent fuel racks.) The new gantry crane was specifically designed to handle the Holtec HI-TRAC® 100D and MPC-32, and both the 110-ton and the 45-ton hoists were designed to mate only with the HI-TRAC® trunnions and MPC lift cleats, respectively. The HI-TRAC Version MS and MPC-32M CBS may also be handled via the gantry crane.

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##### TRO

The Ederer Gantry Crane is FUNCTIONAL when the licensing bases are met. The Ederer Crane will only be moved on safe load paths. Gantry crane operating procedures utilized for cask and cask component lifts will be prepared to include: the steps and proper sequence to be followed in handling the load; defining the safe load path; and other precautions. A specific cask loading and handling procedure will provide additional details for controlled movement during cask handling operations. Crane operators will receive



## IP2 DEFUELED SAFETY ANALYSIS REPORT

Ederer Gantry Crane  
TRM B 3.9.E

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### BASES

#### TRO (continued)

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training that includes provisions of Chapter 2-3 of American National Standards Institute (ANSI) standard B30.2-1976. In addition, completion of a crane specific on-the-job training qualification card is required.

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### ACTIONS

Before any heavy load is lifted the Ederer Crane should be FUNCTIONAL. If at any time the crane becomes NON-FUNCTIONAL, the lifting should cease immediately after the load is in a safe condition.

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### SURVEILLANCE TRS 3.9.E.1

Performance of maintenance, testing and inspection activities in accordance with Chapter 2-2 of ANSI B30.2-1976 assures that the Ederer crane maintains the required capability and level of safety. NUREG-0612 Section 6.1.1(6) requires that test and inspections be performed prior to use where it is not practical to meet the frequencies of ANSI B30.2 for periodic inspection or test or where frequency of crane use is less than the specified inspection and test frequency (e.g., the polar crane inside a PWR containment may only be used every 12 to 18 months during refueling operations, and is generally not accessible during power operations. ANSI B30.2, however, calls for certain inspections to be performed daily or monthly. For such cranes having limited usage, the inspections, test, and maintenance should only be performed prior to their use.)

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## IP2 DEFUELED SAFETY ANALYSIS REPORT

Ederer Gantry Crane  
TRM 3.9.E

### 3.9 SPENT FUEL PIT OPERATIONS

#### 3.9.E Ederer Gantry Crane

TRO 3.9.E The 110 ton Ederer Gantry Crane shall be FUNCTIONAL.

APPLICABILITY: When moving spent fuel casks up to 110 tons into and out of the spent fuel pit in the fuel storage building.

#### ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. 110 ton Ederer gantry crane is NON-FUNCTIONAL	A.1 NO dry cask storage cask handling can proceed.	Immediately

#### SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
TRS 3.9.E.1 Perform maintenance, testing and inspection activities in accordance with Chapter 2-2 of ANSI B30.2-1976.	As required by the ANSI Standard modified by NUREG 612 Section 5.1.1(6).

## BASES

3.11 B.5.b MITIGATING STRATEGIES

B 3.11.A B.5.b Equipment

## BASES

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**BACKGROUND** As a result of the terrorist events of September 11, 2001, the NRC issued EA-02-026, "Order for Interim Safeguards and Security Compensatory Measures" (the ICM Order) dated February 25, 2002. The ICM Order, which is designated as Safeguards Information (SGI), modified then-operating licenses for commercial power reactor facilities to require compliance with specified interim safeguards and security compensatory measures. Section B.5.b of the ICM Order requires licensees to adopt mitigation strategies using readily available resources to maintain or restore core cooling, containment, and SFP cooling capabilities to cope with the loss of large areas of the facility due to large fires and explosions from any cause, including beyond-design-basis aircraft impacts.

Events at the Fukushima – Daiichi Nuclear Power Station following the March 11, 2011, earthquake and tsunami highlight the potential importance of B.5.b mitigating strategies in responding to beyond design basis events.

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**TRO** The existing guidance on the implementation of the strategies, which was adopted by all licensees to meet the regulatory requirements for mitigating strategies, does not describe in detail the practices necessary for maintenance and testing of the B.5.b equipment. In accordance with NEI 06-12 guidelines for FLEX equipment, B.5.b equipment associated with external mitigation strategies shall meet standard industry practices for procuring and maintaining commercial equipment. For a multiple unit site, B.5.b assumes only one unit is affected by the event. The equipment to meet Column 2 requirements are required to be FUNCTIONAL at all times to assure redundancy of function. An additional spare diesel-driven pump is maintained in order to ensure continuity should one pump become unavailable.

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**APPLICABILITY** A B.5.b event could occur regardless of specified condition and there will be a need for the equipment as defined in procedures.

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## BASES

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ACTIONS	The Conditions, Required Actions, and Completion Times are in accordance with the guidelines discussed in Section 11.5 of NEI 12-06 for FLEX equipment. Potential compensatory measures which may be considered include use of suitable FLEX equipment or alternate equipment rented from offsite.
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A.1, A.2 and A.3

The equipment to meet Column 2 FUNCTIONAL requirements can be out of service for up to 90 days provided the redundant equipment is FUNCTIONAL and immediately staged at the Primary B.5.b location. The requirement to move the redundant pump to the Primary B.5.b location is to ensure that it is located outside the potential impact zone and will be available during the event. An additional action is added to report to OSRC in two weeks if corrective action is not completed within 15 days. This provides added management of oversight of the restoration process.

B.1

If the equipment to meet Column 1 FUNCTIONAL requirements becomes non-functioning then initiate actions to restore one of the redundant pieces of equipment within 24 hours. Compensatory actions must be taken if the equipment to meet Column 1 FUNCTIONAL requirements is not expected to be restored or is not restored within 24 hours. Action is required to be initiated immediately since adequate time exists to determine the scope of the compensatory action and completion should be practical within a limited time.

C.1

If the equipment to meet Column 1 FUNCTIONAL requirements becomes non-functioning and potential for a site specific external event is identified, action should be initiated immediately to supplement the equipment with alternate suitable equipment.

D.1

If the equipment to meet Column 1 FUNCTIONAL requirements cannot be restored in 24 hours after the redundant components become non-functioning, then initiate actions immediately to implement compensatory actions. The completion of this activity should reflect the need to quickly restore the function.

BASES

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SURVEILLANCE REQUIREMENTS

Periodic testing and frequency should be determined based on equipment type and expected use. Testing should be done to verify design requirements and/or basis. The basis should be documented and deviations from vendor recommendations and applicable standards should be justified. This activity is not controlled by the TRM.

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## IP2 DEFUELED SAFETY ANALYSIS REPORT

B.5.b Equipment  
TRM 3.11.A

3.11 B.5.b MITIGATING STRATEGIES

3.11.A B.5.b Equipment

TRO 3.11.A The B.5.b equipment specified in TRM Table 3.11.A-1 shall be FUNCTIONAL.

APPLICABILITY: At all times.

### ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. B.5.b component specified in TRM Table 3.11.A-1 does not meet the Column 2 FUNCTIONAL requirements.	A.1 Ensure B.5.b component specified in TRM Table 3.11.A-1 Column 1 staged at Primary B.5.b location	Immediately
	AND	
	A.2 Restore the B.5.b component to Column 2 FUNCTIONAL status	90 days
	AND	
	A.3 If not restored within 15 days, present a report to OSRC giving why out of service and plan to repair	14 days
B. B.5.b component specified in TRM Table 3.11.A-1 does not meet the Column 1 FUNCTIONAL requirements.	B.1 Restore site B.5.b capability to Column 1 FUNCTIONAL status	24 hours
C. B.5.b component specified in TRM Table 3.11.A-1 does not meet the Column 1 FUNCTIONAL requirements during a forecast site specific external event.	C.1 Initiate actions to supplement the B.5.b component with alternate suitable equipment	Immediately

## IP2 DEFUELED SAFETY ANALYSIS REPORT

B.5.b Equipment  
TRM 3.11.A

### ACTIONS (Continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition B not met	D.1 Initiate actions to Implement compensatory measures	Immediately

### SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Not Controlled per TRM	

TRM Table 3.11.A-1

### B.5.b EQUIPMENT THAT DIRECTLY PERFORMS A B.5.b MITIGATION STRATEGY FOR THE KEY SAFETY FUNCTIONS

COMPONENT	NUMBER REQUIRED TO SUPPORT B.5.b STRATEGIES (Column 1)	NUMBER TO MEET B.5.b SPARE REQUIREMENTS (Column 2)
Diesel-Driven Pump w/ Battery and Trailer	1	2

## 5.0 ADMINISTRATIVE CONTROLS

### 5.1 Responsibilities

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5.1.A The Site Vice President (SVP) shall be responsible for overall facility operation in accordance with the Technical Requirements Manual.

5.1.B The Shift Manager shall be responsible for ensuring facility operations are in accordance with the Technical Requirements Manual.

Example: Technical Requirements for Operation (TRO) are met or Required Actions are met within associated Completion Time(s).

5.1.C Department Managers shall be responsible for ensuring work activities are performed in accordance with the Technical Requirements Manual.

Example: Technical Requirement Surveillance (TRS) are met, Technical Requirements for Operations (TRO) are met.

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5.0 ADMINISTRATIVE CONTROLS

5.2 Not Used

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## 5.0 ADMINISTRATIVE CONTROLS

### 5.3 Procedures

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5.3.A Written procedures shall be established, implemented, and maintained covering the Technical Requirements Manual activities.

5.3.B Each procedure of Requirement 5.3.A, and changes thereto, shall be reviewed and approved in accordance with an approved process that meets the requirements of the HDI Decommissioning Quality Assurance Program (DQAP)) prior to implementation.

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## IP2 DEFUELED SAFETY ANALYSIS REPORT

Reporting Requirements  
TRM 5.4

### 5.0 ADMINISTRATIVE CONTROLS

#### 5.4 Reporting Requirements

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##### 5.4.A Hurricane Alert

- a. If the National Weather Service issues a Hurricane Warning for a hurricane with wind in excess of 87 knots (approximately 100 mph) within 500 nautical miles of the facility, a prompt report shall be made to the NRC Incident Response Center within 1 hour of receipt of that Hurricane Warning. This notification is in lieu of the reporting requirements of 10 CFR 50.73.
  - b. Not Used.
  - c. Upon receipt of Hurricane Warnings for the mid-Atlantic coast of the United States, reports issued by the National Weather Service and the National Hurricane Center shall be monitored at least every hour.
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## 5.0 ADMINISTRATIVE CONTROLS

### 5.5 Programs

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#### 5.5.A Offsite Dose Calculation Manual (ODCM)

##### PURPOSE:

Technical Specification 5.5.1, "Offsite Dose Calculation Manual" 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 radiological environmental monitoring program.

The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating, and Radioactive Effluent Release Reports required by Technical Specification 5.6.2 and 5.6.3.

##### PROCEDURE SECTION:

The ODCM is implemented by the ODCM Part 2: Calculational Methodologies

##### REFERENCE:

Technical Specification 5.5.1, "Offsite Dose Calculation Manual (ODCM)"

#### 5.5.B Not Used

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## 5.5 Programs

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### 5.5.C Radioactive Effluent Control Program

#### PURPOSE:

Technical Specification 5.5.3, "Radioactive Control Program" provides a program to conform with 10 CFR 50.36a for control of radioactive effluents and for maintaining doses to members of the public from radioactive effluents as low as reasonably achievable.

#### PROCEDURE SECTION:

The Radioactive Effluent Controls Program is implemented by the ODCM Part 1: "Radiological Effluent Controls."

#### REFERENCE:

Technical Specification 5.5.3, "Radioactive Effluent Controls Program"

### 5.5.D through 5.5.I Not Used

### 5.5.J Explosive Gas and Storage Tank Radioactive Monitoring Program

#### PURPOSE:

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure." The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures."

#### PROCEDURE SECTION:

The Explosive Gas and Storage Tank Radioactivity Monitoring Program is implemented through the following procedures:

ODCM, D 3.1.4 "Liquid Holdup Tanks"

ODCM, D 3.2.6 "Gas Storage Tanks"

5.5 Programs

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5.5.J Explosive Gas and Storage Tank Radioactive Monitoring Program (continued)

REFERENCE:

Technical Specification 5.5.10, "Explosive Gas and Storage Tank Radioactivity Monitoring Program"

5.5.K Not Used

5.5.L Technical Specification (TS) Bases Control Program

PURPOSE:

Technical Specification 5.5.12, "Technical Specification (TS) Bases Control Program" provides a program to processing changes to the Bases of the Technical Specifications.

PROCEDURE SECTION:

The Technical Specification (TS) Bases Control Program is implemented by EN-LI-113, "Licensing Basis Document Change Process".

REFERENCE:

Technical Specification 5.5.12, "Technical Specification (TS) Bases Control Program"

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## 5.0 ADMINISTRATIVE CONTROLS

### 5.6 Record Retention

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In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

5.6.A The following records shall be retained for at least 5 years:

- a. Records of changes made to the procedures required by Technical Requirements Manual.
- b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to Technical Requirements Manual.
- c. Records of surveillance activities, inspections, and calibrations required by the Technical Requirements Manual.

5.6.B The following records shall be retained for the duration of the 10 CFR 50 Facility License:

- a. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments required by the Technical Requirements Manual and pursuant to 10 CFR 50.59.
  - b. Records of reviews and audits required by the Technical Requirements Manual.
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