



10 CFR 50.71(e)
10 CFR 50.4(b)(6)

September 24, 2026

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 (DSAR) Update

In accordance with Title 10 Code of Federal Regulations (CFR) 50.71(e) and 50.4(b)(6), Holtec Decommissioning International, LLC (HDI) is submitting the Indian Point Unit 2 (IP2) biennial Defueled Safety Analysis Report (DSAR) update.

Please find enclosed a copy of Revision 3 of the DSAR. Changes to the DSAR are indicated by revision bars. This is a complete revision of the DSAR, and all pages have been converted to Revision 3. The enclosed DSAR Revision 3 reflects the current plant status. A summary of the changes is within revision change table in Enclosure 1.

There are no regulatory Commitment Changes.

Pursuant to 10 CFR 50.59(d)(2), HDI is reporting that for the interval of September 24, 2024 through September 24, 2026, there were no 10 CFR 50.59 evaluations performed for any changes, tests, or experiments made to IP2.

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.

I declare under penalty of perjury that the foregoing is true and correct. Executed on September 24, 2026.

Sincerely,

**William
Noval**

Digitally signed by William Noval
DN: cn=William Noval, c=US,
o=Holtec International,
ou=Regulatory Affairs,
email=w.noval@holtec.com
Date: 2026.09.24 13:57:51 -04'00'

William Noval
Senior Director, Regulatory Affairs, Licensing and Global trade Compliance
Holtec International

Enclosure 1: Indian Point 2 Defueled Safety Analysis Report

HDI-IPEC-26-036



Krishna P. Singh Technology Campus, 1 Holtec Blvd., Camden, NJ 08104

Telephone (856) 797-0900

Fax (856) 797-0909

cc: NRC Senior Project Manager, NRC NMSS
NRC Region I Regional Administrator
NRC Senior Regional Inspector, Indian Point Energy Center
New York State Liaison Officer Designee, NYSERDA
New York State (NYS) Public Service Commission



ENCLOSURE 1

INDIAN POINT UNIT 2

Defueled Safety Analysis Report Revision 3



Indian Point 2

Defueled Safety Analysis Report



KPS Technology Campus, 1 Holtec Blvd., Camden, NJ 08104

IP2 DEFUELED SAFETY ANALYSIS REPORT

Instructions and Key to Unit 2 DSAR Revision 3

Current Changes - Revision 3:

The 2026 revision to the Unit 2 Defueled Safety Analysis Report (DSAR), issued as Revision 3 contains revision bars to annotate the following changes:

Change	Description	Affected Sections
1	Minor editorial changes to reflect that all spent fuel has been transferred from the spent fuel pit to the ISFSI and the facility is being decommissioned.	All
2	Packaging, transfer and storage of Greater-than-Class C (GTCC) waste.	6
3	Segmentation of large, contaminated components (Stm Generators).	4
4	Unit 1 Stack demolition.	4.2.3 & T4.2-2
5	Editorial rewrite of the Sampling System.	3.4

Historical Information:

The UFSAR and previous revisions to the DSAR contained sections 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 Revision 2 removed many sections marked as “Historical Information” because this information is considered background and is not subject to review or periodic updates. The historical information previously contained in the UFSAR and DSAR is retained in previous versions of the UFSAR and DSAR and can be referenced.

DSAR Figures and Plant Drawings:

The figures from the UFSAR that were retained in the DSAR have not been renumbered. Referenced plant drawings are not included in the DSAR. Refer to the most current revision of the official plant drawing.

Structures, Systems, and Components (SSCs)

Some SSCs are no longer necessary due to permanently defueled conditions with all spent fuel located on an ISFSI. Once an SSC is no longer required to be available it is removed from service and any remaining DSAR description is deleted from the DSAR.

INDIAN POINT ENERGY CENTER (IPEC)
UNIT 2
DEFUELED SAFETY ANALYSIS REPORT (DSAR)

Table of Contents

Chapter 1	– Introduction and Summary
Chapter 2	– Site and Environment
Chapter 3	– Facility Design and Operation
Chapter 4	– Waste Disposal and Radiation Protection System
Chapter 5	– Conduct of Facility Activities
Chapter 6	– Safety Analysis

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 the operation of the reactor or placement of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2). On May 28, 2021, the NRC 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, the 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 decommissioning that the UFSAR served during the 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).

As of February 1, 2023, all IP2 spent fuel nuclear assemblies were transferred out of the IP2 spent fuel pit and placed in dry storage within the Independent Spent Fuel Storage Installation (ISFSI). As a result, information related to the wet storage of spent fuel is no longer applicable and has been deleted from the report.

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.

IP2 DEFUELED SAFETY ANALYSIS REPORT

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.

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.

1.2.1.2 Geology and Hydrology

Geologically, the site consists of 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 ft², 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 groundwater supply.

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

IP2 DEFUELED SAFETY ANALYSIS REPORT

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 of spent fuel at IPEC.. 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 station status is continually changing during decommissioning. Current information on systems, radiological conditions and demolition progress is maintained by the HDI decommissioning organization. The principal station structures are the ISFSI Central Alarm Station (ICAS) and the ISFSI. Legacy buildings such as the Containment Building, Primary Auxiliary Building, the Fuel Storage Building, the Turbine Building, and Control Building are located outside the protected area.

The facility incorporates an electrical system, a radioactive waste disposal system, and other onsite facilities required for the safe decommissioning of the site.

The general arrangement of the facility is shown on historical Figures 1.2-1 and 1.2-4, and facility drawing 504688 (Formerly Figure 2.2-2).

1.2.2.1 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.

1.2.2.2 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.

IP2 DEFUELED SAFETY ANALYSIS REPORT

1.2.2.3 Containment

The reactor containment is a steel-lined reinforced concrete cylinder with a hemispherical dome and a flat base. In a permanently shutdown and defueled condition, the containment building is designed to confine the spread of radioactive contamination during natural phenomenon events.

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.

IP2 DEFUELED SAFETY ANALYSIS REPORT

1.3 General Design Criteria (GDC)

The General Design Criteria define or describe safety objectives and approaches incorporated in the original 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.

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. 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.

The following GDCs remain applicable after all spent fuel was transferred from the spent fuel pit to the ISFSI:

1.3.1 Overall Plant Requirements (GDC 1-GDC 5)

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. (GDC 3)

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 (GDC 5).

1.3.2 Nuclear and Radiation Controls (GDC 11 thru GDC 18)

The non-nuclear process instrumentation measures temperatures, pressures, flows, and levels in auxiliary systems. The quantity and types of process instrumentation provided ensures safe handling of radioactive wastes.(GDC 12)

The plant vent and the waste disposal system liquid effluent are monitored for radioactivity concentration during all normal conditions (GDC 17).

Monitoring and alarm instrumentation are provided for 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 (GDC 18).

A controlled ventilation system removes gaseous radioactivity from various areas of the plant and discharges it to the atmosphere via the plant vent. Continuous building ventilation exhaust

IP2 DEFUELED SAFETY ANALYSIS REPORT

stream(s) are sampled for particulates, and releases are calculated and reported in accordance with the Offsite Dose Calculation Manual (ODCM).

1.3.3 Fuel and Waste Storage Systems (GDC 66 thru GDC 69)

Shielding is provided for waste handling and storage facilities to permit operation within the requirements of 10 CFR 20.

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

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

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.

All solid waste is placed in suitable containers and stored onsite until shipped offsite for disposal.

1.4 (not used)

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.

IP2 DEFUELED SAFETY ANALYSIS REPORT

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 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

IP2 DEFUELED SAFETY ANALYSIS REPORT

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.

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

IP2 DEFUELED SAFETY ANALYSIS REPORT

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
Revision 25,	September 2014
Revision 26,	September 2016
Revision 27,	September 2018

Revision 0 established the DSAR effective June 2020. DSAR Revision 1 was issued in September 2022. DSAR Revision 2 was issued in September 2024. 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).

IP2 DEFUELED SAFETY ANALYSIS REPORT

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.

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.

IP2 DEFUELED SAFETY ANALYSIS REPORT

1.7 DESIGN CRITERIA FOR STRUCTURES AND COMPONENTS

1.7.1 Definition of Seismic Design Classifications (HISTORICAL)

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.

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 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.

IP2 DEFUELED SAFETY ANALYSIS REPORT

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 historical UFSAR 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.

IP2 DEFUELED SAFETY ANALYSIS REPORT

1.7.2 Classification of Particular Structures and Equipment

Following the permanent shutdown, defueling, and transfer of all spent fuel to dry storage, no SSCs meet the definition of Class I or Class II.

1.7.3 (Not used)

1.7.4. 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.

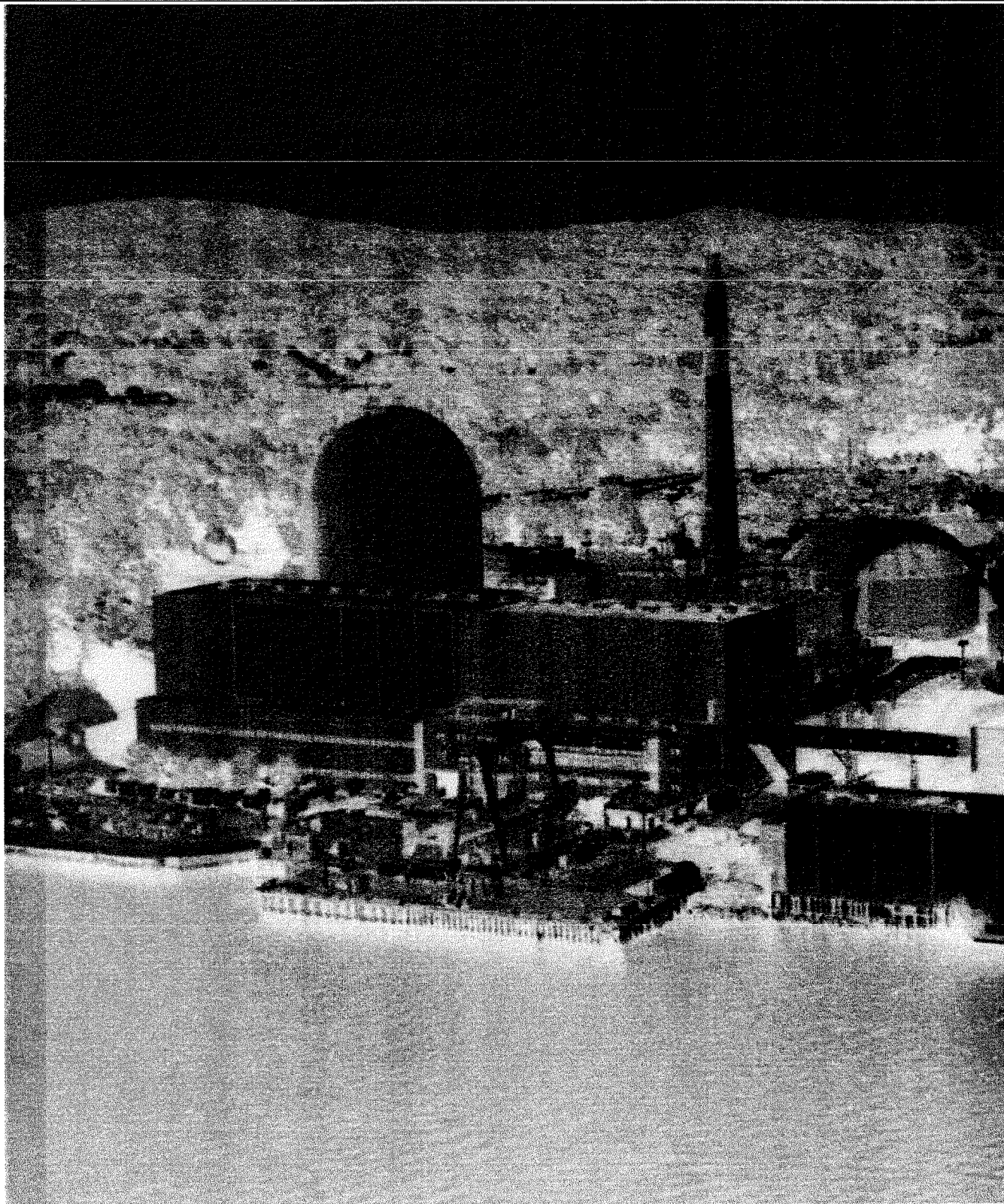
1.7.5 Wind Effects

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.



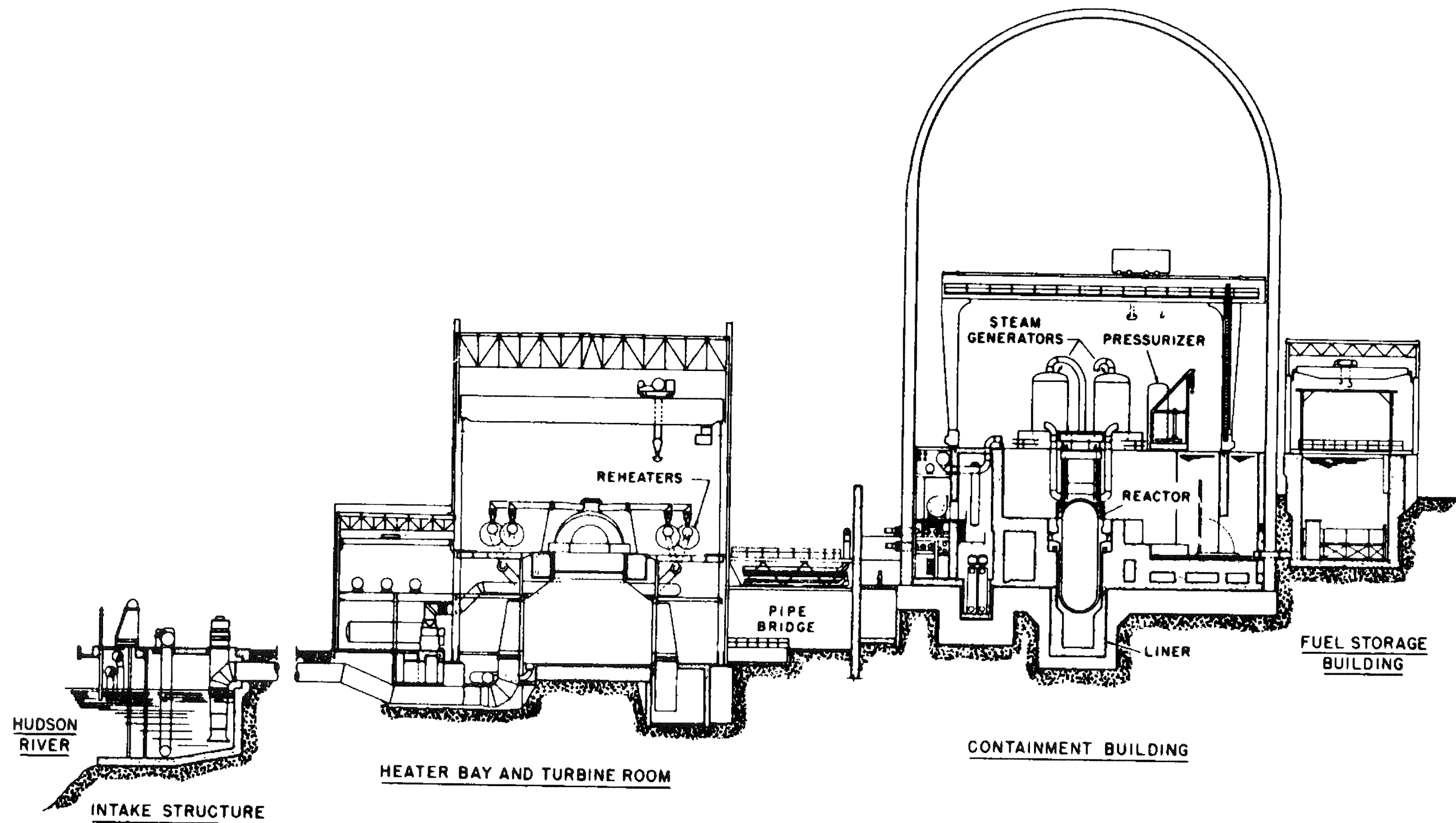
INDIAN POINT UNIT No. 2

UFSAR FIGURE 1.2-1

INDIAN POINT
NUCLEAR GENERATING STATION
UNITS 1 & 2

MIC. No. 1999MC3559

REV. No. 17A



INDIAN POINT UNIT No. 2

UFSAR FIGURE 1.2-4

CROSS SECTION
OF
PLANT

MIC. No. 1999MC3561

REV. No. 17A

CHAPTER 2 SITE AND ENVIRONMENT

2.1 Summary and Conclusions (HISTORICAL)

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. The information previously contained in this section is considered historical and can be viewed in previous revisions of the DSAR and UFSAR

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/ft², 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.

Seismic activity in the Indian Point area is limited to low-level microseismicity. 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 was 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. These data have been used in evaluating the effects of airborne discharges from the facility 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 airborne releases..

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. 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^{4,5} found that environmental radioactivity in the vicinity of the site is no higher than anywhere else in the State of New York.

IP2 DEFUELED SAFETY ANALYSIS REPORT

2.2 Location

2.2.1 General

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 24,000.

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. Portions of the existing waterfront can be 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. 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 "Part 20" areas. Access to the protected areas is controlled via security on a 24-hr basis. Roads within the site are patrolled by security personnel.

2.2.4 Activities On The Site

The principal activities on the site are decommissioning and dry fuel storage.

2.3 Topography

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

IP2 DEFUELED SAFETY ANALYSIS REPORT

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.

2.4 Population and Land Use

The information previously contained in this section is considered historical and can be viewed in previous revisions of the DSAR and UFSAR.

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.

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 can be 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.

The hydrology in the environs of the Indian Point site has been extensively studied. The information previously contained in this section is considered historical and can be viewed in previous revisions of the DSAR and UFSAR.

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.

IP2 DEFUELED SAFETY ANALYSIS REPORT

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.

Historical studies 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.

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.

IP2 DEFUELED SAFETY ANALYSIS REPORT

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 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 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. During the conduct of decommissioning activities, the facility discharge will not exceed its maximum permissible concentration.

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

IP2 DEFUELED SAFETY ANALYSIS REPORT

2.6 Meteorology (HISTORICAL)

2.6.1 General

The information previously contained in this section is considered historical and can be viewed in previous revisions of the DSAR and UFSAR.

2.7 Geology and Seismology (HISTORICAL)

The information previously contained in this section is considered historical and can be viewed in previous revisions of the DSAR and UFSAR.

2.8 Environmental Radioactivity (HISTORICAL)

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.

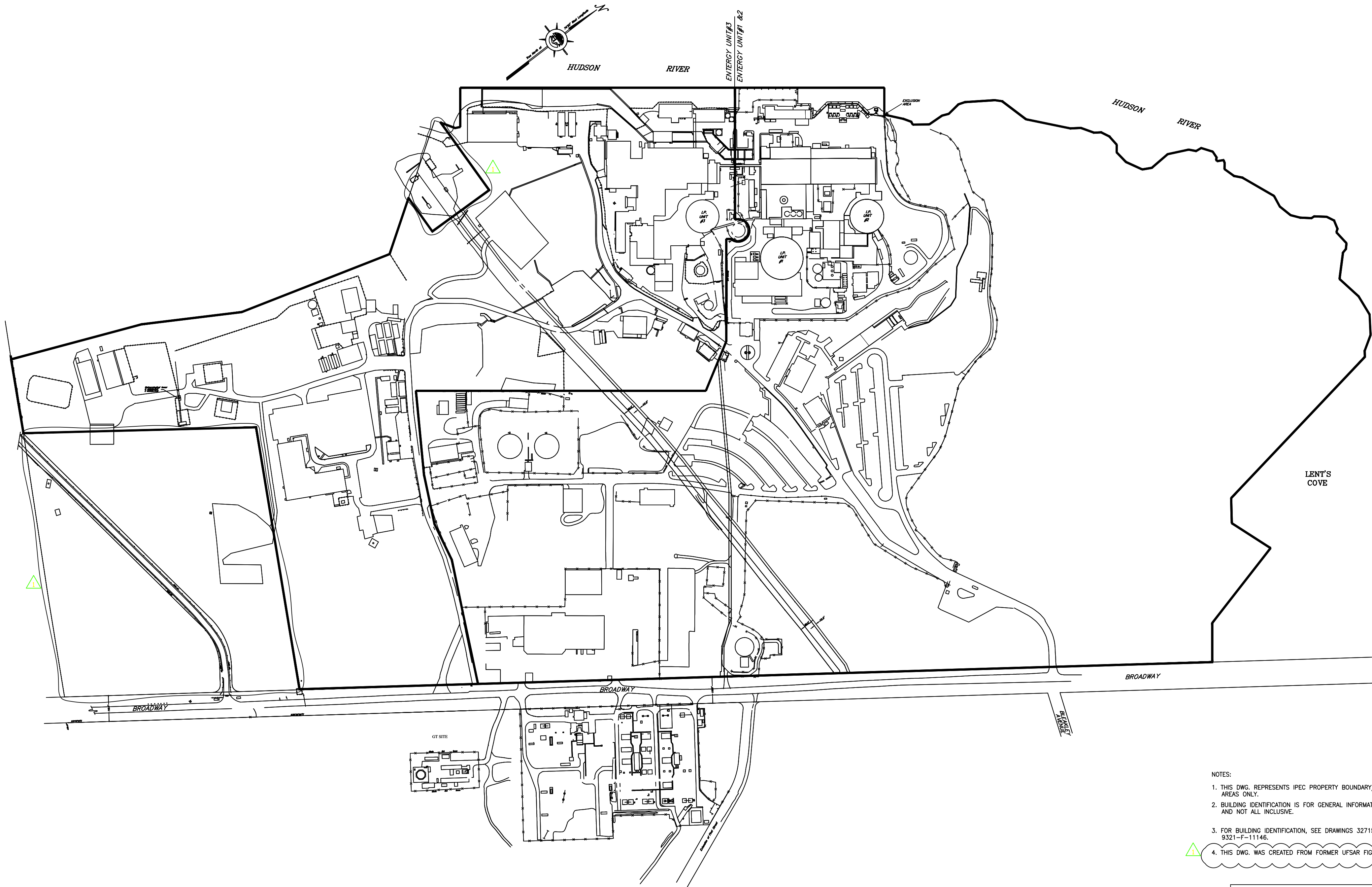
Determinations of radioactivity in the environment are made regularly. Sample locations are defined in the Offsite Dose Calculation Manual (ODCM).

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.

The information previously contained in this section is considered historical and can be viewed in previous revisions of the DSAR and UFSAR.

1	INCORPORATED EC-85797	2/19/20	AI	APPROVAL SIGNATURES ON FILE
REV	DESCRIPTION	DATE	BY	CHKD



- NOTES:
1. THIS DWG. REPRESENTS IPEC PROPERTY BOUNDARY/EXCLUSION AREAS ONLY.
 2. BUILDING IDENTIFICATION IS FOR GENERAL INFORMATION ONLY AND NOT ALL INCLUSIVE.
 3. FOR BUILDING IDENTIFICATION, SEE DRAWINGS 327152 & 9321-F-11146.
 4. THIS DWG. WAS CREATED FROM FORMER UFSAR FIG. 2.2-2

NOT TO SCALE

INDIAN POINT ENERGY CENTER – UNIT 2&3

Old UFSAR Fig 2.2-2

INDIAN POINT PROPERTY
BOUNDARY IDENTIFICATION
CIVIL/STRUCTURAL

DWG NO. 504688
 SCALE NOT TO SCALE

REV 1
 SHEET

IP2 DEFUELED SAFETY ANALYSIS REPORT

CHAPTER 3 FACILITY DESIGN AND OPERATION

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. As of February 2023, all nuclear fuel assemblies were placed in dry storage within the Independent Spent Fuel Storage Installation (ISFSI).

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, fuel reprocessing, and dry storage.

The information previously contained in this section is considered historical and can be viewed in previous revisions of the DSAR and UFSAR.

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:

- CVCS Hold Up Tanks (HUT)
- CVCS Hold Up Tank Transfer Pumps
- Hold Up Tank Recirculation Pump

In addition, the following equipment is part of the Waste Disposal System.

IP2 DEFUELED SAFETY ANALYSIS REPORT

- Waste Hold Up Tank (WHUT)
- Spent Resin Storage Tank
- Waste Hold Up Transfer Pump

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 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.

IP2 DEFUELED SAFETY ANALYSIS REPORT

3.3 Dry Fuel Storage

The IPEC site contains an Independent Spent Fuel Storage Installation (ISFSI) located in the owner-controlled area in its own standalone protected area northeast of IP2. The ISFSI facility consists of two concrete storage pads sized to store 142 spent fuel casks and Greater-than-Class C (GTCC) waste casks. IP2 spent fuel is stored on the pad in a natural convection air-cooled HI-STORM 100 System designed to contain up to 32 fuel assemblies per storage cask. The HI-STORM System consists of a stainless-steel multi-purpose canister (MPC) with a welded base plate and lid, which is placed inside a coated carbon steel and concrete overpack. GTCC waste is stored in HI-SAFE Bolted GTCC Canisters designed for radioactive non-fuel waste, including reactor-related waste and hardware.

The elevation of the ISFSI is approximately 90 feet above MSL, well above the 15-foot MSL most severe hypothetical flooding condition. Drainage and slope of the pads preclude the possibility of any significant ponding.

The Spent Fuel Dry Cask Storage operations were conducted under a general license in accordance with Subpart K of 10 CFR Part 72. The general license issued by 10 CFR 72.210, "General license issued," authorizes a 10 CFR Part 50 nuclear power plant licensee to store spent fuel at an onsite ISFSI. Subpart K of 10 CFR Part 72 also includes 10 CFR 72.212, "Conditions of general license issued under §72.210," which requires the use of a dry cask storage system that is pre-approved by the Nuclear Regulatory Commission, as evidenced by its listing 10 CFR 72.214.

3.4 Sampling System

3.4.1 Design Basis

This system provides samples for laboratory analysis to evaluate auxiliary systems chemistry during normal conditions.

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.

Sampling of some process coolants, such as from 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.

3.4.2 System Design and Operation

The Sampling System provides representative samples for laboratory analysis. Analysis 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.

Sampling System equipment is located inside the auxiliary building in the sampling room.

The sample sink, which is contained in the laboratory bench as a part of the sampling hood,

IP2 DEFUELED SAFETY ANALYSIS REPORT

contains a drain line to the Waste Disposal System.

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.

Sample Sink

The sample sink is located in a hooded enclosure that is equipped with an exhaust ventilator. The work area around the sink and the enclosure is large enough for sample collection and for storage of radiation monitoring equipment. The sink perimeter has a raised edge to contain any spilled liquid. The enclosure is penetrated by sample lines from the facility and a demineralized water line, all of which discharge into the sink. The sink and work areas are stainless steel.

Piping and Fittings

All liquid and gas sample lines are austenitic stainless-steel tubing (USAS B31.1) and are designed for high pressure service. Compression fittings and 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.

Valves

Manual 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 samples flow rate.

All valves in the system are constructed of austenitic stainless steel or equivalent corrosion resistant material.

Leakage Provisions

Leakage of radioactive material from the most likely places outside the Containment is collected by placing the entire sampling station under a hood provided with an offgas vent to the building exhaust. Liquid leakage from the valves in the hood is drained to the chemical drain tank.

Incident Control

The system operates under administrative manual control.

IP2 DEFUELED SAFETY ANALYSIS REPORT

3.5. Control of Heavy Loads

The information previously contained in this section is considered historical and can be viewed in previous revisions of the DSAR and UFSAR.

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 is comprised of 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 cool the 22 Standby Diesel Generator.

Water is drawn from the river and passes under a debris wall, through two racks in parallel and finally two traveling screens.

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

IP2 DEFUELED SAFETY ANALYSIS REPORT

reactor in the event of a fire, is no longer 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. 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 water to areas where hose connections are located for general usage.

City water for the Indian Point Unit 2 comes from the city water main on Broadway via the Unit 1 mains and storage tanks.

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. A connection has been provided in the station air system to allow a backup supply of air from portable compressed air equipment.

3.6.4.2 Station Air System

The station air system 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.

A package boiler is 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 the boiler links the existing steam header to Unit 2 and also to Unit 3, so that output from the package boiler may be made available for the heating requirements of Unit 1, Unit 2, or Unit 3.

IP2 DEFUELED SAFETY ANALYSIS REPORT

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. Other heating services extend to the fan room, the fuel storage building, the primary auxiliary building, and the primary water storage tank.

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- γ 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 (Fe_3O_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.

IP2 DEFUELED SAFETY ANALYSIS REPORT

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 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.

IP2 DEFUELED SAFETY ANALYSIS REPORT

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 10 CFR 20 limits. These limits are specified and controlled in accordance with the Offsite Dose Calculation Manual (ODCM).

3.8.2 System Design and Operation

The primary auxiliary building ventilation system 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>Exhaust¹</u>			
Fans, standard conditions	2	55,500 cfm	1
Fan pressure	-	10.3 in. H ₂ 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 ₂ 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.

IP2 DEFUELED SAFETY ANALYSIS REPORT

3.9 (Not Used)

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.

3.10.2 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.

3.11 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 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.

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 Offsite Dose Calculation Manual (ODCM) provides the methodology to calculate radiation does 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.

3.13 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 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.

IP2 DEFUELED SAFETY ANALYSIS REPORT

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.

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.

"Page" or "Party" line communication can be initiated in the CCR from either communication consoles or from handset stations.

The facility radio and telephone communications systems are automatically supplied from a back-up power source, upon failure of the normal power source. The page/party system is powered from the DC system (through an inverter) with backup power from a bus.

An in-house radio system provides communications between personnel at the facility. Field units are low-wattage, hand-held units.

IP2 DEFUELED SAFETY ANALYSIS REPORT

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. 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 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.

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.

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.

3.15.1 ELECTRICAL SYSTEM DESIGN

3.15.1.1 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. Detailed information on Electrical System components and layout is considered historical and can be viewed in previous revisions of the DSAR and UFSAR.

IP2 DEFUELED SAFETY ANALYSIS REPORT

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 an unmonitored release does not occur.

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

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.

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.

3.16.1.1.2 Performance Standards

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 de-classified. Its structural members had 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.

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 information previously contained in this section is considered historical and can be viewed in previous revisions of the DSAR and UFSAR.

IP2 DEFUELED SAFETY ANALYSIS REPORT

3.16.1.3 Codes and Standards

The information previously contained in this section is considered historical and can be viewed in previous revisions of the DSAR and UFSAR.

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 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).

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.

IP2 DEFUELED SAFETY ANALYSIS REPORT

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.

The refueling canal connects the reactor cavity with a sealed 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.

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 information previously contained in this section is considered historical and can be viewed in previous revisions of the DSAR and UFSAR.

3.16.2.3 Material Specifications

The information previously contained in this section is considered historical and can be viewed in previous revisions of the DSAR and UFSAR.

3.16.2.4 Cathodic Protection

The information previously contained in this section is considered historical and can be viewed in previous revisions of the DSAR and UFSAR.

CHAPTER 4 WASTE DISPOSAL AND RADIATION PROTECTION SYSTEM

4.1 Waste Disposal System

4.1.1 (Not Used)

4.1.2 System Design and Operation

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 only 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.

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 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 can be aligned to the Discharge Canal. 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.

When necessary, processed liquid waste in the Unit 1 Waste Distillate Storage Tanks can be routed to the Unit 2 Chemical Volume Control System (CVCS) Hold-Up Tanks.

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 and spent filters are removed from the site using a high integrity container, which is dewatered and prepared for transportation in accordance with the Process Control Program.

Miscellaneous solid wastes such as paper, rags and glassware, are processed in accordance with the Process Control Program. When possible, solid waste is reduced per the Process Control Program.

Solid radioactive wastes in the form of dry activated waste (DAW) or solidified (or dewatered) resins may be stored in the Solid Radwaste Storage Building prior to offsite shipment. The facility is a Part 37 protected, non-safety, non-seismic concrete structure with approximate dimensions of 100' X 100' located at the south end of the site. Pursuant to Generic Letter 81-38, the facility was designed and constructed in accordance with Appendix 11.4-A of NUREG-0800 to minimize the impact of storage to the public and the environment. Adequate shielding is provided to limit the offsite doses to less than 5 mrem/year. In addition, the facility incorporates the ALARA methodology of Regulatory Guides 8.8 and 8.10. The quantity of radioactive material allowed and the shielding configurations ensure compliance with the dose rate criteria for both the site boundary and unrestricted areas onsite.

The Original Steam Generator Storage Facility (OSGSF) is used for solid radwaste storage and for segmenting large contaminated components such as the Original Steam Generators (OSGs). The OSGSF is a non-safety, non-seismic 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, north of the the Buchanan Service Center access road.

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. Filtered ventilation provides negative air pressure inside building and ventilation discharge is monitored in accordance with ODCM requirements..

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 were applied to the floor slab and lower portion of the walls to ease decontamination, if required.

Greater-Than-Class C (GTCC) waste is stored at the ISFSI awaiting shipment to a permanent offsite disposal facility.

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.

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

All resin has been removed from the spent resin storage tank (SRST) and the SRST has been sluiced.

The SRST was designed to retain resin discharged from the facility demineralizers. Resins can be stored in the tank for decay of short-lived isotopes. 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.

4.1.2.2.11 Pumps

Pumps used throughout the system for draining tanks and transferring liquids.

4.1.2.2.12 Piping

Piping carrying liquid wastes is stainless 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

Operating experiences demonstrate that the Waste Disposal System for the facility provides for suitable control of radioactive materials to the environment. A description of Indian Point 2 releases is submitted periodically to the NRC in the Annual Radioactive Effluent Release Report, in accordance with the requirements of US NRC Regulatory Guide 1.21. The Indian Point 2 releases, as evidenced by the aforementioned reports, are well within the limits of 10 CFR 20.

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 is the original plant preoperational assessment of river water dilution factors between the Indian Point site and the nearest public drinking water intake and it is considered historical. Appendix 4B can be viewed in previous revisions of the DSAR and UFSAR.

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 or DOT-7A Containers for removal to a burial facility. When possible, miscellaneous solid waste is reduced in accordance with the Process Control Program. All solid radioactive wastes are packaged for removal to a licensed low-level waste burial facility in accordance with applicable regulations and burial site criteria.

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, ¹ Class C
Sump tank	No code
Spent resin storage tanks	ASME III, ¹ Class C
Gas decay tanks	ASME III, ¹ Class C
Waste holdup tank	No code
Distillate storage tank	No code
Waste filter	No code
Piping and valves	USAS-B31.1, ² 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.

TABLE 4.1-2 (Sheet 1 of 2)
Component Summary Data

Tanks	Quantity	Type	Volume	Design Pressure	Design Temperature °F	Material
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 ³	Atm	150	ss
Spent resin Storage	1	V	300-ft ³	100 psig	150	ss
Distillate storage	2	H	25000 gal	17 psig	250	cs

Pumps	Quantity	Type	Flow gpm	Head ft	Design Pressure psig	Design Temperature F°	Material₁
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 ₂	20	100	100	180	ss

TABLE 4.1-2 (Sheet 2 of 2)
Component Summary Data

Pumps	Quantity	Type	Flow gpm	Head Ft	Design Pressure psig	Design Temperature °F	Material₁
Sump tank	2	H, C ₂	20	100	150	180	ss
Waste transfer	1	H, C ₂	30	215	105	70	ss
Distillate recirculation	2	H, C ₂	200	100	43 ₃	120 ₄	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

Key:

H = Horizontal
V = Vertical

C = Centrifugal
CC = Centrifugal canned

CS = 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

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.

4.2.1.1 Monitoring Radioactivity Releases

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 are exhausted from the plant vent, which is monitored. Any contaminated liquid effluent discharged to the condenser circulating water canal is monitored.

4.2.1.2 Monitoring Fuel and Waste Storage

Monitoring and alarm instrumentation are provided to detect excessive radiation levels. Radiation monitors or sampling is 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.

A controlled ventilation system removes gaseous radioactivity from the atmosphere of waste treating areas of the auxiliary building and discharges it to the atmosphere via the plant vent.

4.2.1.3 Fuel and Waste Storage Radiation Shielding

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.

4.2.1.4 Protection Against Radioactivity Release From Spent Fuel and Waste Storage

All waste disposal systems are contained and equipment designed so that accidental releases directly to the atmosphere are monitored and will not exceed applicable limits. 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 work control review process. Shielding arrangements may be altered consistent with the radiation protection plan and the ALARA program station procedures.

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 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.

4.2.2.2 Shielding Design

Auxiliary Shield

The auxiliary shield consists of concrete walls around certain components and piping. 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 4.2-1.

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. 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 continued safe maintenance of the facility and assurance that personnel exposure does not exceed 10 CFR 20 limits.

4.2.3.2 System Design

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; a CRT display and annunciators.

The minicomputer unit includes a console with CRT, disk drive and magnetic tape drive. It communicates digitally with the individual radiation monitoring units, and processes, records, and displays data.

Table 4.2-2 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.

Information on specific monitors is given in the following sections.

4.2.3.2.1 Plant Vent Air Monitor

R44 monitors for gaseous activity. It was historically seismically qualified, and its power supplies was historically class IE. Particulates are collected on filters and analyzed in the count room.

4.2.3.2.2 Liquid Waste Effluent Radiation Monitor

This monitor, R54, is powered from a Unit 1 motor control center. It alarms independent of the communications loop through the minicomputer. This monitor terminates the tank discharges upon detecting high activity.

4.2.3.2.3 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 the common process radiation monitor panel for high radiation.

4.2.3.2.4 U1 Stack Vent

During releases, the U1 Stack Vent is monitored. Particulates are collected on filters and analyzed in the count room.

4.2.3.2.5 Unit 2 Steam Generator Mausoleum Vent

The Unit 2 Steam Generator Mausoleum discharge vent is continuously monitored when solid waste is being cut and processed. Particulates are collected on filters and analyzed in the count room.

4.2.4 Environmental Monitoring Program

Environmental monitoring is discussed in Section 2 and requirements are set forth in the ODCM. The environmental monitoring program and results are described in the Annual Radiological Environmental Operating Report.

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. Powered air purifying respirator (PAPR) hood or mask
3. Airhood.

Self-contained breathing apparatus can be obtained for situations 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.

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 facilities is designed to provide access control to the RCA.

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.

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 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 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.

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.

Onsite equipment and supplies for the treatment of injured and/or contaminated personnel are controlled by site procedures

4.2.5.6 Treatment Procedures and Techniques

The procedure and techniques used to treat injured and/or contaminated personnel are addressed by site 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 provided.

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 site 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 site 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 exhaust paths shall be tested in accordance with the ODCM 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

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 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 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.

TABLE 4.2-1
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.

TABLE 4.2-2
Radiation Monitoring Channel Data

<u>Effluent Monitors</u>		
<u>Channel</u>	<u>Stream Monitored</u>	<u>Normal Maximum Channel Output</u>
R-44	Plant Vent Air Gaseous	1.0 x 10 ⁷ CPM
R-54	Liquid Waste Effluent	1.0 x 10 ⁷ CPM
R-62	Unit 1 Sphere Foundation Drain Sump	1.0 x 10 ⁷ CPM

IP2 DEFUELED SAFETY ANALYSIS REPORT

CHAPTER 5 CONDUCT OF FACILITY ACTIVITIES & ADMINISTRATIVE CONTROLS

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 detailed in facility procedures.

5.1.1 Facility Staff

The Site Vice President 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 manager responsible for overall operational activities shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during an absence.

The manager responsible for overall operational activities or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

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

IPEC staff has a level of education, experience, and skill commensurate with their level of responsibility. Their qualifications provide reasonable assurance that decisions and actions during the decommissioning of IPEC will not constitute a hazard to the health and safety of the public.

The recommendations of ANSI N3.1-1978 for equivalent positions within the ISFSI organization are used to establish minimum qualifications for applicable management, supervisory, and professional-technical and other personnel in the plant organization as directed by the DQAP.

Personnel qualifications shall meet ANSI/ANS 3.1-1978 with the following exceptions:

- a. The radiation protection manager shall meet or exceed the qualifications of Regulatory Guide 1.8, Revision 2, 1987.
- b. The following qualifications may be considered equivalent to a bachelor's degree:
 - 4 years of post-secondary schooling in science or engineering,

IP2 DEFUELED SAFETY ANALYSIS REPORT

- 4 years of applied experience at a nuclear facility in the area for which qualification is sought,
- 4 years of operational or technical experience/training in nuclear power, or any combination of the above totaling 4 years.

Years of experience used to meet the education requirements, as allowed by this exception, may not also be used to meet the experience requirements.

5.2 Training

A training and retraining program for the facility staff 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 for the various functional areas of emergency activity.

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 (IPEC) Independent Spent Fuel Storage Installation-Only, Security, Training and Qualification, and Safeguard Contingency Plan."

5.3 Written Procedures

Written procedures and administrative policies are established, implemented, and maintained in accordance with the DQAP and the IP2 Renewed Facility License. The procedures applicable to the safe storage of nuclear fuel recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978 are provided as described in the DQAP or as referenced in the Updated DSAR.

5.3.1 Emergency Plan Implementing Procedures

Emergency plan implementing procedures provide instructions and outline responsibilities of IPEC personnel in the event of an emergency at IP2.

5.4 Records

Quality Assurance records are identified, generated, collected, compiled, stored, maintained, and retained as described in the DQAP. Records are maintained in the sites Electronic Data Management System (EDMS) or 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

IP2 DEFUELED SAFETY ANALYSIS REPORT

with the requirements of 10 CFR 50 and the DQAP. The Safety Review Committee) functions 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 Safety Review Committee (SRC)

The Safety Review Committee functions to advise on all matters related to nuclear safety in accordance with the requirements of the DQAP.

5.5.2 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 requirements of 10 CFR 72.212(b)(9) without impairing the ability to operate the facility safely. The approved facility security program, titled "Indian Point (IPEC) Independent Spent Fuel Storage Installation-Only, Security, Training and Qualification, and Safeguard Contingency Plan," 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.

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 as exempted for a decommissioned facility.

5.8 (Not Used)

5.9 QUALITY ASSURANCE PROGRAM

5.9.1 General

The Quality Assurance Program (QAP) for Indian Point Unit 2 is described in the Decommissioning Quality Assurance Program (DQAP). The DQAP and associated implementing documents provide for control of activities that affect the quality of safety-related nuclear facility structures, systems, and components. The DQAP 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).

IP2 DEFUELED SAFETY ANALYSIS REPORT

5.9.2 Scope

The DQAP implements 10 CFR 50 Appendix B, 10 CFR 71 Subpart H, and 10 CFR 72 Subpart G. 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. 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.

5.9.3 Organization and Responsibilities

The organizational structure responsible for implementation of the Quality Assurance Program is described in the 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.

5.10 Offsite Dose Calculation Manual (ODCM)

a. The ODCM shall contain 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, and

b. 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.

c. Licensee initiated changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:
 - a. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s) and
 - b. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations,
2. Shall become effective after approval of the manager responsible for overall operational activities, and

IP2 DEFUELED SAFETY ANALYSIS REPORT

3. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.11 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

- a. Limitations on the functional capability of radioactive liquid and airborne monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,
- b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001 – 20.2402,
- c. Monitoring, sampling, and analysis of radioactive liquid and airborne effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM,
- d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit/facility to unrestricted areas, conforming to 10 CFR 50, Appendix I,
- e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year is in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days,
- f. Limitations on the functional capability and use of the liquid and airborne effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I,
- g. Limitations on the dose rate resulting from radioactive material released in airborne effluents from the site to areas at or beyond the site boundary shall be in accordance with the following:

For iodine-131, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ,

IP2 DEFUELED SAFETY ANALYSIS REPORT

- h. Limitations on the annual and quarterly doses to a member of the public from tritium, and all radionuclides in particulate form with half lives > 8 days in airborne effluents released from each unit/facility to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I, and
- i. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

5.12 Storage Tank Radioactivity Monitoring Program

This program provides controls for 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."

The program shall include:

- a. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents, and
- b. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

5.13 Reporting Requirements

-----NOTE-----

A single submittal may be made for a multiple unit/facility station. The submittal should combine sections common to all units/facilities at the station.

Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the unit/facility during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

IP2 DEFUELED SAFETY ANALYSIS REPORT

A full listing of the information to be contained in the Annual Radiological Environmental Operating Report is provided in the ODCM.

Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit/facility in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit/facility. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

IP2 DEFUELED SAFETY ANALYSIS REPORT

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.

As of February 2023, all IP2 spent nuclear fuel assemblies were placed in dry storage within the ISFSI. Associated License Amendment #299 contains a design requirement that spent fuel shall not be stored in the spent fuel pit. As a result, a fuel handling accident is no longer a possibility.

This chapter discusses: (a) a postulated drop of a high integrity container (HIC) containing radioactive resins, (b) a postulated drop of a storage container loaded with Type B/C solid radioactive waste, and (c) a postulated drop of cask containing Greater-Than-Class C (GTCC) waste inside Containment.

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 within the 10 CFR 50.59 screening process.

6.2 Fuel-Handling Accidents (No longer postulated)

6.2.1.9 Fuel-Handling Accidents Results (HISTORICAL)

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

IP2 DEFUELED SAFETY ANALYSIS REPORT

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 (No Longer Postulated)

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 6.7-1). Therefore, a separate liquid-specific release accident evaluation is not required.

6.5 High Integrity Container Drop Event

A calculation (Reference 6.7-2) 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 are well within the limits of 10 CFR 50.67, RG 1.183, 10 CFR 100, and EPA-400/R-17/001. 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.

When used, the HICs can be located in the IP2 Alleyway between the PAB and FSB, the IP3 Annex building, and the IP1 Fuel Handling Floor. Since χ/Q_s for IP1 are not available, a bounding atmospheric dispersion value of less than 1 sec/m³ (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 6.7-3) 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 within regulatory limits and is below the EAB dose from a postulated release from an FHA.

6.6 Class B/C Solid Radwaste Container Drop Event

Class B/C waste from reactor core and vessel segmentation will be packaged in Containment and transported to the Solid Radwaste Storage Building (SRSB) for interim storage. After loading of the B/C box in Containment is complete, the Polar Crane is used to lift and place the box on a rail system at the Containment hatch. The box is then transferred to a heavy haul trailer and transported to the SRSB. Inside the building, the box is lifted from the trailer and placed in storage.

The hypothetical drop of a loaded Class B/C container (B/C box) during this activity is evaluated

IP2 DEFUELED SAFETY ANALYSIS REPORT

in Reference 6.7-2 and 6.7-3. Two drop accidents are considered:

- a) Drop of the B/C box outside the Containment hatch and exposure of contents.
- b) Drop of the B/C box inside the Solid Radwaste Storage Building and exposure of contents

Dose at the site boundary (EAB), Control Room, and low population zone (LPZ) from a hypothetical accident resulting in loss of shielding for a loaded B/C box is calculated and compared to the following limits:

- Dose limits and associated durations per 10 CFR 50.67 & RG. 1.183
- Dose rates well within 10 CFR 100 limits (covered by RG 1.183 check)
- 1 R over 4 days per EPA-400/R-17/001

6.6.1 Evaluation

EAB: Calculated dose rates at the EAB are:

- a) Accidental drop just outside Containment: 7.8 mr/hr
- b) Accidental drop inside FLEX building: 30.9 mr for 32 hour duration, or 0.97 mr/hr

Bounding value for the above accidents is 7.8 mr/hr. Note that this value is based on a distance to the EAB of 230 meters, which is the approximate distance from the Containment hatch opening to the Hudson River in a direction unobstructed by the Containment Building.

2-Hr dose = $7.8 \text{ mr/hr} \times 2 \text{ hrs} = 15.6 \text{ mr}$, which is negligible compared to the limit of 6.3 R per R.G. 1.183.

4-Day dose = $7.8 \text{ mr/hr} \times 4 \text{ hrs} = 31.2 \text{ mr}$, which is well below the limit of 1 R over 4 days per EPA-400/R-17/001. Note that an occupancy factor of 1 hr per day is applied to the EAB at the Hudson River, which is conservative per guidance of RG 1.109.

Control Room: Calculated dose rate at the Control Room:

Conservatively use dose rate at 140 meters with no shielding from structures = 30.6 mr / hr.

2-Hr dose = $30.6 \text{ mr/hr} \times 2 \text{ hrs} = 61 \text{ mr}$, which is well below the limit of 5.0 R per R.G. 1.183.

LPZ: Dose is judged to be negligible compared to limit of 6.3 R per R.G. 1.183.

6.6.2 Conclusions

Consequences of a hypothetical drop of a B/C box from segmentation activities are well within limits specified in 10 CFR 50.67, R.G. 1.183, 10 CFR 100, and EPA-400/R-17/001.

IP2 DEFUELED SAFETY ANALYSIS REPORT

6.7 GTCC Cask Drop Event

GTCC waste from reactor internals segmentation will be packaged in Containment and transported to the ISFSI for interim storage. Inside Containment, GTCC waste is loaded in a canister inside a HI-TRAC transfer cask. The transfer cask is then placed on top of a HI-SAFE storage cask and the canister with GTCC is lowered into the storage cask. The lid-less HI-SAFE is then placed on the Containment hatch rail system and travelled out of Containment for transport to the ISFSI.

The hypothetical drop of a loaded HI-TRAC or HI-SAFE cask and exposure of its contents inside Containment was evaluated. Dose at the exclusion area boundary (EAB) due to streaming through the open equipment hatch is calculated and compared to the following limits:

- Dose limits and associated durations per 10 CFR 50.67 & RG. 1.183
- Dose rates well within 10 CFR 100 limits and less than 1 R over 4 days per EPA-400/R-17/001

A hypothetical cask drop during transit operations outside the Containment is not a bounding event because calculations demonstrate no loss of shielding would occur.

6.7.1 Evaluation

EAB: Highest 2-hr dose considering occupancy factors is 11 mrem at the nearest residence, which is negligible compared to the limit of 6.3 R per R.G. 1.183. This value is also less than the Class B/C waste container drop discussed in Section 6.6 above.

The highest 4-day dose considering occupancy factors is 528 mrem at the nearest residence, which is below the limit of 1 R over 4 days per EPA-400/R-17/001.

Control Room: The Control Room is shielded by the containment concrete wall (i.e. no direct line-of-sight through the equipment hatch) and therefore would not be subject to significant dose from this hypothetical accident.

LPZ: Dose is judged to be acceptable compared to the limit of 6.3 R per R.G. 1.183.

6.7.2 Conclusions

Consequences of a hypothetical drop of a loaded HI-TRAC or HI-SAFE cask and exposure of its contents inside Containment are well within the limits specified in 10 CFR 50.67, R.G. 1.183, 10 CFR 100, and EPA-400/R-17/001.

6.8 References

- 6.8-1. NSIR/DPR-ISG-02, Emergency Planning Exemption Requests for Decommissioning Nuclear Power Plants, May 11, 2105 (ADAMS Accession Number ML14106A057)
- 6.8-2 Report No. IPEC-RPT-22-015 R0, "HIC Drop Accident," prepared June 2022

IP2 DEFUELED SAFETY ANALYSIS REPORT

6.8-3 WMG 22-369-001, BFA Tank Boundary Dose Assessment, Rev. 2