CHAPTER 1
INTRODUCTION AND SUMMARY

1.1 Introduction

On February 8, 2017, Entergy Nuclear Operations, Inc. (Entergy) notified the U.S. Nuclear Regulatory Commission (NRC) that it would permanently cease power operations at Indian Point Nuclear Generating Station Unit No. 2 (IP2) no later than April 30, 2020. On May 12, 2020, Entergy submitted certifications for permanent cessation of operations and permanent removal of fuel from the reactor vessel to the NRC in accordance with 10 CFR 50.82(a)(1)(i) and (ii). Following the NRC docketing those certifications, the 10 CFR Part 50 license no longer permits operation of the reactor or placement of fuel in the reactor vessel in accordance with 10 CFR 50.82(a)(2).

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

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

By NRC order dated August 27, 2001 (Reference 1.1-1), Con Edison's ownership/operation of Indian Point 1 and 2 was transferred to Entergy Nuclear Indian Point 2 (ENIP2), LLC, as the owner of Indian Point 1 and 2 plants, and Entergy Nuclear Operations (ENO), Inc. as the operator of Indian Point 2 and maintainer of Indian Point 1. Consequently, references to Con Edison (or derivatives thereof) in this document remain only when used in historical context.

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

REFERENCES FOR SECTION 1.1

1. NRC letter to Consolidated Edison, Indian Point Nuclear Generating Unit Nos. 1 and 2 – Order Approving Transfer of Licenses from the Consolidated Edison Company of New York, Inc., to Entergy Nuclear Indian Point 2, LLC, and Entergy Nuclear Operations, Inc. and Approving Conforming Amendments (TAC Nos. MB0743 and MB0744), August 27, 2001.

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 Entergy Nuclear and Entergy Nuclear Operations, Inc.) is adjacent to and south of Unit 1. The site is about 24 miles north of the New York City boundary line. The nearest city is Peekskill, 2.5 miles northeast of Indian Point. An aerial photograph, [Historical] Figure 2.2-1, shows the site and about 58 miles^2 of the surrounding area.

1.2.1.1 Meteorology

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

1.2.1.2 Geology and Hydrology

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

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

1.2.1.3 Seismology

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

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

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

1.2.1.4 Environmental Radiation Monitoring

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

1.2.1.5 Conclusions

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

1.2.2 Facility Description

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

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

1.2.2.1 Spent Fuel Storage

Auxiliary systems are provided to perform the following functions:

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

1.2.2.2 Electrical System

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

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

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

1.2.2.3 Control Room

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

1.2.2.4 Diesel Generators

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

1.2.2.5 Waste Disposal System

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

1.2.2.6 Fuel Handling System

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

The system also includes the following features:

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

1.2.2.7 Structures

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

1.2.2.8 Containment

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

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

1.2 FIGURES

1.3 General Design Criteria (GDC)

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

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

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

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

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

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

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

Reference sections:

Section Title	Section
Site & Environment;	
Meteorology	2.6
Geology and Seismology	2.7

Introduction & Summary;	
Design Criteria for Structures and Equipment	1.7

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

Reference sections:

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

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

Reference sections:

Section Title	Section
Conduct of Operations;	
Records	5.4
Introduction & Summary;	
Quality Assurance Program	1.6

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

The facility is equipped with a control room.

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

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

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

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

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

Reference sections:

Section Title	Section
Auxiliary Systems;	
Auxiliary Coolant System	3.3
Waste Disposal & Radiation Protections System;	
Radiation Protection	4.2

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

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

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

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

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

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

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

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

Reference sections:

Section Title	Section
Auxiliary Systems;	
Sampling System	3.4
Waste Disposal & Radiation Protection System;	
Waste Disposal System	4.1
Radiation Protection Systems	4.2

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.

Radioactive gases are pumped by compressors through a manifold to one of the gas decay tanks where they are held a suitable period of time for decay. Cover gases in the nitrogen blanketing system are reused to minimize gaseous wastes. During normal activities, gases are discharged intermittently at a controlled rate from these tanks through the monitored facility vent. The system is provided with discharge controls so that environmental conditions do not restrict the release of radioactive effluents to the atmosphere. Liquid wastes are processed to remove most of the radioactive materials. The spent resins from the demineralizers and the filter cartridges are packaged and stored onsite until shipment offsite for disposal.

Reference section:

Section Title	Section
Waste Disposal & Radiation Protection System; Waste Disposal System	4.1

REFERENCES FOR SECTION 1.3

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

1.4 Design Parameters

1.4.1 Fuel Cladding

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

1.4.2 Fuel Assembly Design

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

1.5 Supplements and Revisions to Original FSAR

1.5.1 Supplements

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

1.5.2 Revisions

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

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

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

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

1.6 QUALITY ASSURANCE PROGRAM

1.6.1 General

The IPEC Quality Assurance Program (QAP) for Indian Point Unit 2 is described in the IPEC Quality Assurance Program Manual (QAPM) 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 QAPM applies to all activities associated with structures, systems, and components that are safety related or controlled by 10 CFR 72. The QAPM also applies to transportation packages controlled by 10 CFR 71. The methods of implementation of the requirements of the QAPM are commensurate with the item's or activity's importance to safety. The applicability of the requirements of the QAPM to other items and activities is determined on a case-by-case basis. The QAPM implements 10 CFR 50 Appendix B, 10 CFR 71 Subpart H, and 10 CFR 72 Subpart G. All items and activities affecting safety addressed in Regulatory Guide 1.29 "Seismic Design Classification" revision 3, September 1978, are also governed by the Quality Assurance Program. A list of safety related items is maintained. Elements of the Quality Assurance Program are also applicable to activities and items affecting safety as defined in Licensing commitments. (Reference 1.6-1)

1.6.3 Organization and Responsibilities

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

REFERENCES FOR SECTION 1.6

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

1.7 DESIGN CRITERIA FOR STRUCTURES AND COMPONENTS

1.7.1 Definition of Seismic Design Classifications

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

1. TID-7024, "Nuclear Reactors and Earthquakes," August 1963 and,

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

Class I

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

Class II

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

Class III

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

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

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

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

2. Primary steady state stresses when combined with the seismic stress resulting from the response to a ground acceleration of 0.10g acting in the vertical and 0.15g acting in the

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

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

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

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

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

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

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

1. The natural period of vibration of the structure or component is determined.

2. The response acceleration of the component to the seismic motion is taken from the response spectrum curve at the appropriate period.

3. Stresses and deflections resulting from the combined influence of normal loads and the seismic load due to the design earthquake (0.05g acting in the vertical and 0.10g acting in the horizontal planes simultaneously) are calculated and checked against the limits imposed by the design standard.

4. Stresses and deflections resulting from the combined influence of normal loads and the seismic loads due to the maximum potential earthquake (0.10g acting in the vertical and 0.15g acting in the horizontal planes simultaneously) are calculated and checked to verify that deflections do not cause loss of function and that stresses do not produce rupture.

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

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

1.7.2 Classification of Particular Structures and Equipment

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

Item	Class
Buildings and Structures	
Containment	III
Spent fuel pit	I
Control Building	III
Diesel Generator Building	III
Intake structure (to the extent that water is always available to the service water pumps)	III
Service water screenwell	III
Primary Auxiliary Building	III
Turbine Building	III
Buildings containing conventional facilities Such as the Maintenance and Outage Building	III
Original Steam Generator Storage Facility	III
Equipment, Piping, and Supports	

[***Note*** *- Class I components (equipment, piping, instrumentation, etc.) located in or supported on a Class II structure are protected from earthquake damage or are backed up by other Class I components located in or supported by a Class I structure.*]

Item	Class
Radiation monitoring system	III
Process instrumentation and controls	III
Fuel assemblies	I
Refueling Water Storage Tank	I
Auxiliary building ventilation system	III
Component cooling loop	III
Instrument air system	III

Sampling system	III
Spent fuel pit cooling loop	III
Standby power supply system	III
Diesel generator and fuel oil storage tank DC power supply system Power distribution lines to equipment Control panel board Motor control centers	III
Waste disposal system	III
Containment crane	III
Manipulator and other cranes	III
Conventional equipment, tanks and piping, other than Classes I and II	III
Auxiliary boiler feed and service water pumps and piping	III
Chemical and volume control system	III

1.7.3 Design Criteria for Seismic Class I Structures and Equipment

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

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

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

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

With a ground acceleration of 0.15g horizontal, the spectral acceleration corresponding to the maximum point on the 0.5-percent critical damping response curve was used to calculate an equivalent static force imparted to the pipe at its support points. This resulted in a seismic design load approximately equal to 0.6W horizontally and 0.4W vertically taken simultaneously, where W is the weight of the pipe including static forces. The sum of the resulting additional stress plus the normal stresses was limited to 1.2 times the B31.1 code allowable. The stresses in the pipe supports and hangers were likewise limited to 1.2 times the B31.1 code allowable.

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

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

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

1.7.3.1 Piping, Vessels, and Supports

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

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

1.7.4 Models and Methods for Seismic Class I Design

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

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

1.7.4.1 Containment Building

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

1.7.4.1.1 Steel

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

1.7.4.1.2 Concrete

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

1.7.4.2 Control Building

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

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

1.7.4.3 Diesel Generator Building

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

1.7.4.4 Intake Structure

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

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

1.7.4.5 Waste Holdup Tank Pit

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

1.7.4.6 Spent Fuel Pit

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

1.7.4.7 Primary Water Storage Tank and Refueling Water Storage Tank Foundation

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

1.7.4.8 Class I Piping Systems

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

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

$$1.3^{*}\, K\, S_s + S_n \leq 1.8\, S_a$$

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

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

where:

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

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

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

Solving, the K-factor becomes:

$$K = 5$$

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

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

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

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

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

1. Representatives from Westinghouse and United Engineers and Constructors, Inc., visited the Indian Point Unit 2 site and inspected the Class I piping systems.

2. Based upon their best engineering judgment, representative worst-case lines were selected for detailed dynamic analyses.

3. In exercising their engineering judgment, these representatives looked for the following characteristics, which would indicate possible sources of problems.

    a. Amplification due to the location and elevation in building.

    b. Large concentrated masses such as overhung motor-operated valves, particularly in what appear to be flexible sections of the pipe.

    c. Complexity of configuration of the piping system itself such that application of the original design criteria would be difficult.

    d. Manual excitation of the pipe by pushing or kicking indicated excessive flexibility either in the pipe excited or the piping attached to it.

4. The results of the dynamic analyses were compared with original design values to determine whether the design approach was conservative. Portions of the following systems were analyzed:

    a. Service water (Historical Classification).
    b. Component cooling (Historical Classification).

1.7.4.8.1 Design Approach

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

Two fundamental principles underlie the design approach, namely:

1. The system be designed such that its fundamental natural frequency does not coincide with the exciting frequency.

2. The maximum seismic stresses in piping be less than the USAS B31.1 code allowable value. The seismic stresses were limited to 0.2 S allowable (3000 psi). This is extremely conservative since the longitudinal pressure stress accounts for approximately 0.5 S allowable leaving a margin of safety of 0.5 S allowable, which is unused. (Note-this is based on a maximum allowable of 1.2 S_a)

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

1.7.4.8.2 Analysis Approach

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

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

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

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

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

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

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

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

1. The maximum longitudinal stress due to seismic excitation was limited to 0.2S rather than the usual 0.7S.

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

3. A low value of the fraction of critical damping was adopted (0.5-percent). Dr. N. M. Newmark recommended a value of 2-percent for vital piping at or just below the yield point. This would reduce the maximum amplification of the ground acceleration.

4. The maximum longitudinal stresses due to pressure, deadweight, and seismic loads were presumed to occur at the same cross-section and some point in the cross-section.

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

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

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

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

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

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

1.7.4.9 Service Water Lines

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

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

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

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

1.7.4.10 Masonry Walls

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

1.7.5 Wind Effects

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

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

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

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

1.7.6 Structural Effects

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

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

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

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

1.7.6.1 Seismic Analysis of the Indian Point Unit 2 Turbine Building

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

1. The stack can withstand a tornado wind load of approximately 300mph prior to buckling failure of the stack steel shell.

2. The maximum stress in the stack at the critical vortex-shedding frequency wind velocity is 7660 psi, which provided a 3.64 factor of safety against stack failure by this mode.

3. The maximum combined dead and seismic stress for the earthquake parameters defined for the site is 19,140 psi, which provides a 1.46 factor of safety against stack failure by this mode.

1.7.6.3.1 Load Case 1 - Tornado

I. Load Criteria

Wind = 300 mph
L = D + W'

where:

L = Total load
D = Dead load
W'= Tornado load

II. Method of Load Analysis

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

III. Allowable Stress Criteria

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

where:

σ_a = allowable stress (psi)

E = modulus of elasticity (psi)

t = shell thickness (in.)

ν = Poisson's ratio

r = radius of stack (in.)

IV. Stress Determination

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

where:

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

I = moment of inertia of stack (in.^{4})

A = cross sectional area of stack (in.^{2})

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

1.7.6.3.2 Load Case 2 - Seismic

I. Load Criteria

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

b) Damping 7-percent.

c) Ground response curve - Figure 1.7-2.

$$L = D + E'_{h'} = E'_V$$

where:

E'_h = load resulting from horizontal earthquake component

E'_V =load resulting from vertical earthquake component

II. Method of Load Analysis

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

III. Allowable Stress Criteria

See Load Case 1, item III.

IV. Stress Determination

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

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

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

$= 1.46$

where:

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

1.7.6.3.3 Load Case 3 - Vortex-Shedding

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

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

C_L = Lift coefficient for a stationary circular cylinder

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

D = Average stack diameter (ft)

L = Length of stack (ft)

δ = Logarithmic decrement

ρ = Air density (0.0023385 lb - sec^2/ft^4)

v = F1 x Vc

Vc = Critical vortex-shedding velocity (fps)

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

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

S = Stronhal number

f = Fundamental frequency (cps)

II. Pertinent parameters

CL = 0.1

MF = 4.0

D = 20-ft

L = 334.5-ft

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

Vc = 42.7 fps

F1 = 1.2

S = 0.27

f = 0.576 cps

III. Stress criteria

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

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

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

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

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

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

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

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

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

$$(\ddot{U}_{rn})\,\text{Max} = (\ddot{A}_n)\,\text{Max}\ \varnothing_{rn}$$

$$(\ddot{A}_n)\,\text{Max} = \Gamma_n\, Sa_n$$

$$\Gamma_n = \frac{\sum_r M_r \varnothing'_{rn}}{\sum_r M_r \varnothing^2_{rn}}$$

Where:

n = Refers to mode n

r = Refers to mass r

$Ø'_{rn}$ = Component of $Ø_{rn}$ in the direction of the earthquake

$Ø_{rn}$ = Component of mode shape n for mass r

M_r = Mass lumped at point r

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

Sa_n= Spectral acceleration for mode n from response curve for 7-percent damping

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

Γ_n = Modal participation factor for mode n

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

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

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

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

1.7.6.5 Evaluation of Structural Modifications

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

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

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

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

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

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

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

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

REFERENCES FOR SECTION 1.7

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

2. NRC Branch Technical Position MEB 3-1, Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment.

3. Letter (Attachment I) from S. Bram, Con Edison, to NRC, Subject: Indian Point Unit No. 2 Spent Fuel Storage Capacity Increase, dated January 19, 1990.

4. Letter (Attachment B) from S. Bram, Con Edison, to NRC, Subject: Request for License Amendment to Technical Specification Modifying Spent Fuel Storage Requirements, dated June 20, 1989.

5. Letter from Cahill, Con Edison, to A. Schwencer, Director of Nuclear Reactor Regulation NRC, Subject: Supplemental Response to IE Bulletins 79-02 and 79-07, dated November 27, 1979.

6. Letter from Steven A. Varga, NRC to John D. O'Toole Con Edison, Subject: Completion of IE Bulletin 80-11, "Masonry Wall Design" for Indian Point Nuclear Generating Unit No. 2 (IP2), (Safety Evaluation Report included) dated October 19,1983.

TABLE 1.7-1
Damping Factors

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

TABLE 1.7-2
Loading Combinations and Stress Limits

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

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

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

TABLE 1.7-3
Dynamic Characteristics of the Turbine Building

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

TABLE 1.7-4
Relative Stiffness Percentages

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

TABLE 1.7-5
Inertial Loads

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

TABLE 1.7-6
Frequencies

Frequencies for First and Second Analysis
(Units: cps)

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

1.7 FIGURES

1.8 Control of Heavy Loads

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