June 2011

This version of the Indian Point Unit 2 Updated Final Safety Analysis Report (UFSAR) is the licensee's version 22, submitted to the NRC on October 6, 2010, with information current through April 2010, with certain redactions of sensitive information by staff of the Nuclear Regulatory Commission (NRC) to allow release to the public. The redactions are made under 10CFR2.390(d)(1). The material included within is classified as publicly available information. As of June 2011, this is the latest UFSAR revision submitted to the NRC.

The redactions were made due to meeting the NRC's criteria on sensitive information, as specified in SECY-04-0191, "Withholding Sensitive Unclassified Information Concerning Nuclear Power Reactors from Public Disclosure", dated October 19, 2004, ADAMS ML042310663, as modified by the NRC Commissioners Staff Requirements Memorandum on SECY-04-0191, dated November 9, 2004, ADAMS ML043140175.

The following information was redacted by NRC staff:
Figures 5.1-2, 5.1-3, 5.1-4, 5.1-5, 5.1-6, 5.1-7, which were supplied as plant drawings 9321-2501, 2502, 2503, 2506, 2507, 2508.

Any other material that is listed as "deleted" was deleted by the licensee as part of their continuous update process for the UFSAR.

Instructions and Key to Indian Point 2 UFSAR for Revision 22

Current Changes - Revision 22:
The 2010 revision to the UFSAR is Revision 22. The current changes made in this update that have been sent to the NRC are highlighted in a gray shaded background.

Historical Information:
This UFSAR has marked some parts of the UFSAR as "Historical Information" following the guidance in NEI 98-03, Rev. 1, "*Guidelines for Updating Final Safety Analysis Reports*". Information that is highlighted with a green background is designated as "Historical Information" and 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.

Deleted Information and Figures:
Deleted information will have the word "Deleted" in yellow highlighting.

NRC Orders and License Conditions:
Information relocated to the UFSAR from the Technical Specifications or included by NRC Order, become a Licensing Condition and cannot be removed from the UFSAR without NRC approval. Where these are identified, there will be a *[Note:]* with **bold text and purple highlighting**.

Fission Product Barrier Design Basis Limits for IP2:
Information contained in the UFSAR that represent fission product barrier design basis limits for the plant are highlighted with an aqua shaded background, or aqua lettering if the change is contained in a gray shaded background. These are fundamental to the fission product barrier and cannot be exceeded, changed or altered without NRC approval.

UFSAR Figures and Plant Drawings:
In an effort to improve the accuracy of the figures in the UFSAR and to reduce redundant work, Revision 19 replaced UFSAR Figures, where possible, with references to the current plant drawings. Each current revision has placed "snapshots" of the referenced plant drawings for use by the NRC and for user convenience.

Searching the UFSAR:
The UFSAR has been reformatted in Adobe Acrobat. Use the Adobe search feature to find the information you are looking for.

INDIAN POINT ENERGY CENTER (IPEC)
UNIT 2
UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR)

Table of Contents

CHAPTER 1
INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

[Historical Information] The Final Safety Analysis Report (FSAR) was submitted in support of an application by Consolidated Edison Company of New York, Inc. (Con Edison) for a license to operate the Indian Point Unit 2 nuclear power plant (AEC Docket 50-247, Permit No. CPPR-21). It provided pertinent technical information in accordance with Section 50.34 of 10 CFR 50 requirements to obtain a nuclear power plant operating license. Westinghouse Electric Corporation was the primary contractor and had turnkey responsibility for the design, construction, testing, and initial startup of the facility. Westinghouse had contracted with United Engineers and Constructors as architect-engineer to provide engineering assistance in the design of and construction of the structural and civil works.

This document is the Updated FSAR and is submitted in accordance with Section 50.71(e) of 10 CFR Part 50. The revision history is summarized in Section 1.9.2.

The following paragraphs contain a summary of the report's scope:

The unit employs a pressurized water reactor nuclear steam supply system (NSSS) furnished by Westinghouse Electric Corporation.

The reactor was originally licensed at a maximum thermal power of 2758 MW. By letter dated September 30, 1988 Con Edison initiated a request to authorize an increase in the licensed thermal power level to 3071.4 MW. The NRC documented their acceptance of this request in a Safety Evaluation Report dated March 7, 1990. By letter dated December 12, 2002, Entergy Nuclear Operations, Inc. initiated a request for a Measurement Uncertainty Recapture power uprate of 1.4%, for a licensed thermal power level of 3114.4 MW. The NRC documented their acceptance of this request in a Safety Evaluation Report dated May 22, 2003.

By letter dated January 29, 2004, Entergy Nuclear Operations, Inc. initiated a request for a Stretch Power Uprate, for licensed Thermal power level of 3216MW. The NRC documented their acceptance if this request in a Safety Evaluation Report dated October 27, 2004 and issued License Amendment #241.

The reactor is presently licensed to operate until September 28, 2013. The reactor power corresponds to an electric output from the turbine-generator of approximately 1078 MW.

The plant heat removal systems have been designed for the equivalent guarantee rating of 3071.4 MWt; some of the portions of the safety analysis dependent on heat removal capacity of plant and safeguards systems have assumed the maximum calculated power of 3216 MWt as have the evaluations of activity release and radiation exposure.

By NRC order dated August 27, 2001 (Reference 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 Chapter 1 of this report summarizes the principal design features and parameters of the plant, pointing out the similarities and differences with respect to other pressurized water nuclear power plants presently in operation. A general description of the plant is included as well as a statement and summary of all the General Design Criteria.

Chapter 2 contains a description and evaluation of the site and environs, supporting the suitability of the site for a reactor of the size and type described. Chapters 3 and 4 describe and evaluate the reactor and the reactor coolant system; Chapter 5, the containment system; Chapter 6, the engineered safety features; Chapter 7, plant instrumentation and control; Chapter 8, the electrical system; Chapter 9, the auxiliary and emergency system; Chapter 10, the steam and power conversion system; Chapter 11, radioactive waste disposal and radiation protection. Chapter 12 and 13 are conduct of operations and initial test and operations, respectively. They describe plant organization, training programs, and startup administrative procedure. Chapter 14 is a safety evaluation summarizing the analyses, which demonstrate the adequacy of the reactor protection system and the containment and engineered safety features, and which show that the consequences of various postulated accidents are within applicable limits.

REFERENCES FOR SECTION 1.1

1. NRC letter to Consolidated Edison, Indian Point Nuclear Generating Unit No.s 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 PLANT 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. These data were used in evaluating the effects of gaseous discharges from the plant during normal operations and during the postulated loss-of-coolant accident. In addition, data supplied by the U.S. Weather Bureau at the Bear Mountain Station, regarding the meteorological conditions during periods of precipitation, were used to evaluate the rainout of fission gases into surface water reservoirs following the postulated loss-of-coolant accident. The evaluations indicate that the site meteorology provides adequate diffusion and dilution of any released gases.

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 plant 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 plant. Although it is hard, the jointed limestone formation is permeable to water. Thus, if water from the plant should enter the ground (an improbable event since the plant is designed to preclude any leakage into the ground) it 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 plant 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 plant design is to treat the river water as if it were used for drinking and thus to reduce radioactive discharges, by dilution with ordinary plant 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 plant 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 for nearly 20 years in association with the 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 containment design and the engineered safety features included in the plant design lead to the conclusion of appropriate suitability of the site for safe operation of the Indian Point Unit 2 nuclear power plant. Accident

analyses presented in Chapter 14 verify that the maximum expected doses at or beyond the site boundary are within applicable limits.

1.2.2 Plant Description

The unit incorporates a closed-cycle pressurized water nuclear steam supply system, a turbine generator and their necessary auxiliaries. A radioactive waste disposal system, fuel handling system and all auxiliaries, structures, and other onsite facilities required for a complete and operable nuclear power plant are provided for the unit.

The general arrangement of the plant is shown on historical Figures 1.2-1 and 1.2-4, and 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 Nuclear Steam Supply System (NSSS)

The nuclear steam supply system consists of a pressurized water reactor, reactor coolant system, and associated auxiliary fluid systems. The reactor coolant system is arranged as four closed reactor coolant loops, each containing a reactor coolant pump and a steam generator, connected in parallel to the reactor vessel. An electrically-heated pressurizer is connected to one of the loops.

The reactor core is composed of uranium-dioxide pellets enclosed in zircaloy tubes with welded end plugs. The tubes are supported in assemblies by a spring clip grid structure. The mechanical control rods consist of clusters of stainless steel clad absorber rods and guide tubes located within the fuel assembly. The core was initially loaded in three regions of different enrichments with new fuel being introduced into the outer region at successive refuelings and discharged to spent fuel storage, following burnup.

The steam generators are vertical U-tube units employing Inconel tubes. Integral separating equipment reduces the moisture content of the steam leaving the steam generators to 0.25-percent or less.

The reactor coolant pumps are vertical, single stage, centrifugal pumps equipped with controlled leakage shaft seals.

Auxiliary systems are provided to perform the following functions:

1. Charge the reactor coolant system.
2. Add makeup water.
3. Purify reactor coolant water.
4. Provide chemicals for corrosion inhibition and reactor control.
5. Cool system components.
6. Remove residual heat when the reactor is shut down.
7. Cool the spent fuel storage pool.
8. Sample reactor coolant water.
9. Provide for emergency core cooling.
10. Collect reactor coolant drains.
11. Provide containment spray.
12. Provide containment ventilation and cooling.
13. Dispose of liquid, gaseous and solid wastes.

1.2.2.2 Reactor and Plant Control

The reactor is controlled by a coordinated combination of chemical shim and mechanical control rods. The control system allows the unit to accept step load changes of ± l0-percent and ramp load changes of ± 5-percent per min over the load range of 15-percent to, but not exceeding, 100-percent power under nominal operating conditions subject to xenon limitations.

1.2.2.3 Turbine and Auxiliaries

The turbine is a tandem-compound, comprising one high pressure and three low pressure cylinders, 1800 rpm unit having 45-in. exhaust blading in the low pressure cylinders. There are four moisture preseparators located at the four high-pressure turbine exhaust lines and six combination moisture separator-reheater units that are employed to dry and superheat the steam between the high and low pressure turbine cylinders. The turbine generator is capable of a 50-percent loss of external electrical load without turbine or reactor trip. The turbine auxiliaries include deaerating surface condensers, steam jet air ejector, turbine-driven main feedwater pumps, motor-driven condensate pumps, and six stages of feedwater heating.

The original turbine generator had a guaranteed capability of 1,021,793 kWe at 1.5-in. Hg absolute exhaust pressure with zero percent makeup and six stages of feedwater heating.

1.2.2.4 Electrical System

The main generator feeds electrical power through an isolated phase bus to two half-sized main power transformers. Station auxiliaries receive power during normal operation from either the station auxiliary transformer (i.e., offsite power) or the unit auxiliary transformer (i.e., unit main power transformers).

The auxiliary electrical system provides power to those auxiliary components required to operate during normal or emergency conditions of operation. Standby power required during plant startup, shutdown, and after reactor trip is supplied to the station auxiliary transformer from the Con Edison 138-kV system by either of two separate overhead lines from the Buchanan substation approximately 0.50 mile from the plant. Alternate feeds from the 13.8-kV system are also available for immediate manual connection to the auxiliary buses. [Deleted]

Emergency power supply for vital instruments and controls is from four 125-V station batteries.

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

1.2.2.5 Control Room

The plant is provided with a reactor and turbine-generator control room containing all necessary instrumentation for the operation of the plant under normal and accident conditions.

Adequate shielding and air conditioning facilities permit occupancy during all operating or accident conditions.

1.2.2.6 Diesel Generators

Three diesel-generator sets supply emergency power for shutdown or essential safeguards operation in the event of a loss of all other alternating current auxiliary power.

1.2.2.7 Waste Disposal System

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

1.2.2.8 Fuel Handling System

The fuel handling system provides the ability to fuel and refuel the reactor core. Carefully established administrative procedures plus the design of the system minimizes the probability of potential fission product release during the refueling operation.

The system also includes the following features:

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

1.2.2.9 Engineered Safety Features

The engineered safety features for this plant have sufficient redundancy of component and power sources such that under the conditions of a hypothetical loss-of-coolant, the system can, even when operating with partial effectiveness, maintain the integrity of the containment and keep the exposure of the public within applicable limits.

The major engineered safeguards systems are as follows:

1. The containment system, which incorporates continuously pressurized and monitored penetrations and liner weld channels and a seal water injection system, which provides a highly reliable, essentially leaktight barrier against the escape of radioactivity, which might be released to the containment atmosphere.

2. The safety injection system (which constitutes the emergency core cooling system) provides borated water to cool the core in the event of a loss-of-coolant accident.

3. The containment air recirculation cooling system provides a heat sink to cool the containment atmosphere.

4. The containment spray system provides a spray of cool, borated water to the containment atmosphere that is a heat sink and also provides iodine removal capability.

1.2.2.10 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 of the reactor containment interior components arrangement are shown on Plant Drawings 9321-2501, 9321-2502, 9321-2503, 9321-2506, 9321-2507, 9321-2508 [formerly UFSAR Figures 5.1-2 through 5.2-7]. General layouts and interior components arrangement of the primary auxiliary building, control building, fuel storage building, and holdup tank building were removed from the UFSAR due to security reasons following September 11, 2001 and can be viewed on plant drawings.

1.2.2.11 Containment

The reactor containment is a steel-lined reinforced concrete cylinder with a hemispherical dome and a flat base. The containment is designed to withstand the internal pressure accompanying a loss-of-coolant accident, is virtually leaktight, and provides adequate radiation shielding for both normal operation and accident conditions.

When required, the containment isolation valve seal water system permits automatic rapid sealing of pipes, which penetrate the containment so that in the event of any loss-of-coolant accident, leakage from containment to the environment is minimal.

Ground accelerations for the operational basis earthquake used for containment design purposes and all seismic Class I structures (Section 1.11) 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.

1.2 FIGURES

Figure No.	**Title**
Figure 1.2-1	Indian Point Nuclear Generating Units 1 & 2 [Historical]
Figure 1.2-2	Deleted
Figure 1.2-3	Deleted
Figure 1.2-4	Cross Section of Plant [Historical]
Figure 1.2-5	Deleted
Figure 1.2-6	Deleted
Figure 1.2-7	Deleted
Figure 1.2-7	Deleted
Figure 1.2-8	Deleted
Figure 1.2-9	Deleted

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.

More recently, 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). 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 Plant REQUIREMENTS (GDC 1 - GDC 5)

All systems and components of the facility are classified according to their importance. Those items vital to safe shutdown and isolation of the reactor or whose failure might cause or increase the severity of a loss-of-coolant accident or result in an uncontrolled release of excessive amounts of radioactivity are designated Class I. Those items important to reactor operation but not essential to safe shutdown and isolation of the reactor or control of the release of substantial amounts of radioactivity are designated Class II. Those items not related to reactor operation or safety 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
Reactor Coolant System;	
Design Bases	4.1
Containment System Structures;	
Design Bases	5.1.1
Electrical Systems;	
Design Bases	8.1
Introduction & Summary;	
Design Criteria for Structures and Equipment	1.11

Fire prevention in all areas of the nuclear electric plant 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 fire fighting equipment is provided with capacities proportional to the energy that might credibly be released by fire.

Reference sections:

Section Title	Section
Instrumentation & Control;	
Operating Control Stations	7.7
Auxiliary & Emergency Systems;	
Facility Service System	9.6

The only structures, systems, or components important to safety that are shared between Units 2 and 3 are:

1. The cooling water discharge channel, which carries the service water discharge to the river.
2. The Emergency Fuel Supply to the Emergency Diesel Generators.

Since the channel is designed to handle the discharge flow from both operating units, sharing of this structure will not impair the ability of Unit 2 safety systems to perform their safety functions. Units 2 and 3 are required by Technical Specification to maintain a designated amount of fuel reserve onsite or at the Buchanan substation, which is dedicated for emergency diesel generator use at that unit. Therefore sharing the Emergency Fuel Supply to the Emergency Diesel Generators will not impair the ability of Unit 2 safety systems to perform their safety functions.

Reference sections:

Section Title	Section
Electrical Systems;	
Emergency Fuel Supply	8.2.3.2
Auxiliary & Emergency Systems;	
Service Water System	9.6.1

A complete set of facility plant 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 plant, as defined by the quality assurance program, is retained.

Reference sections:

Section Title	Section
Conduct of Operations;	
Records	12.4
Initial Tests and Operation	13
Introduction & Summary;	
Quality Assurance Program	1.10

1.3.2 Protection By Multiple Fission Product Barriers (GDC 6 - GDC 10)

The reactor core, with its related control and protection system, is designed to function throughout its design lifetime without exceeding acceptable fuel damage limits. The core design, together with reliable process and decay heat removal systems, provides for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and anticipated transient situations.

The reactor control and protection instrumentation is designed to actuate a reactor trip for any anticipated combination of plant conditions when necessary to ensure DNBR remains at or

above the applicable safety analysis DNBR limit and fuel center temperature below the melting point of uranium-dioxide.

Reference sections:

Section Title	Section
Reactor;	
Design Bases,	3.1
Reactor Design	3.2
Instrumentation and Control;	
Protective Systems	7.2
Safety Analysis	14

The design of the reactor core and related protection systems ensures that power oscillations, which could cause fuel damage in excess of acceptable limits are not possible or can be readily suppressed.

Low frequency spatial xenon oscillations may occur in the axial dimension. However, the core is expected to be (and has proven to be) stable to xenon oscillations in the X-Y dimension. Ex-core instrumentation is provided to monitor any xenon induced oscillations. Incore instrumentation is used periodically to calibrate and verify the information provided by the ex-core instrumentation.

The moderator temperature and overall power coefficients in the power operating range were maintained negative by inclusion of burnable poison shims in the first core loading. The overall power coefficient in the power operating range is always negative (as discussed in Section 14.1.11.2).

Reference sections:

Section Title	Section
Reactor;	
Design Bases,	3.1
Reactor Design	3.2

The reactor coolant system in conjunction with its control and protective provisions is designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions, and maintain the stresses within applicable code stress limits. The materials of construction of the pressure retaining boundary of the reactor coolant system are protected by control of coolant chemistry from corrosion phenomena, which might otherwise reduce the system structural integrity during its service lifetime.

System conditions resulting from anticipated transients or malfunctions are monitored and appropriate action is initiated automatically to maintain the required cooling capability and to limit system conditions so that continued safe operation is achieved.

The system is protected from overpressure by means of pressure relieving devices, as required by Section III of the ASME Boiler and Pressure Vessel Code and by an overpressure protection system intended to ensure compliance with 10 CFR 50, Appendix G.

Isolable sections of the system are provided with overpressure relieving devices discharging to closed systems such that the system code allowable relief pressure within the protected section is not exceeded.

Reference section:

Section Title	Section
Reactor Coolant System;	
Design Bases	4.1

The design pressure of the containment exceeds the peak pressure occurring as the result of the complete blowdown of the reactor coolant through any pipe rupture of the reactor coolant system up to and including the hypothetical severance of a reactor coolant system pipe.

All piping systems, which penetrate the vapor barrier are anchored so that the penetration is structurally adequate to resist the piping loads and the vapor barrier will not be breached due to a hypothesized pipe rupture. The lines (with the exception of sample tubing) connected to the reactor coolant system that penetrate the vapor barrier are restrained near the secondary shield walls and are each provided with at least one valve between the shield wall and the reactor coolant system. These restraints are designed to withstand the thrust moment and torque resulting from a hypothesized rupture of the attached pipe.

All isolation valves are supported to withstand, without impairment of valve operability, the combined loading of the design-basis accident and design seismic conditions.

Reference section:

Section Title	Section
Containment System Structures;	
Design Bases	5.1.1

1.3.3 Nuclear And Radiation Controls (GDC 11 - GDC 18)

The plant is equipped with a control room, which contains all controls and instrumentation and facilities necessary for continuous operation of the reactor and turbine generator under normal and accident conditions.

Sufficient shielding, ventilation control and filtration, and containment integrity are provided to ensure that control room personnel will not be subjected to doses under postulated accident conditions during occupancy of the control room, which in the aggregate, would exceed the applicable limits.

Instrumentation and controls essential to avoid undue risk to the health and safety of the public are provided to monitor and maintain neutron flux, reactor coolant pressure, flow rate, temperature, and control rod positions within prescribed operating ranges.

The non-nuclear regulating, process and containment instrumentation measures temperatures, pressures, flows, and levels in the reactor coolant system, steam systems, containment, and other auxiliary systems.

The quantity and types of process instrumentation provided ensures safe and orderly operation of all systems and processes over the full operating range of the plant.

The operational status of the reactor is monitored from the control room. When the reactor is subcritical, the neutron source multiplication is continuously monitored and indicated by proportional counters located in instrument wells in the primary shield adjacent to the reactor

vessel. The source detector channels are checked prior to operations in which criticality may be approached by the use of an incore source. Any appreciable increase in the neutron source multiplication, including that caused by the maximum physical boron dilution rate, is slow enough to give ample time to start corrective action (boron dilution stop and/or emergency boron injection) to prevent the core from becoming critical (as discussed in Sections 14.1.5.2.3 and 14.1.5.3).

When the reactor is critical, means for showing the relative reactivity status of the reactor is provided by display of all rod bank positions in the control room. Periodic samples of the coolant boron concentration are taken, the variation of which provides a further check on the reactivity status of the reactor including core depletion during life.

Instrumentation and controls provided for the protective systems are designed to trip the reactor when necessary to prevent or limit fission product release from the core, to limit energy release, to signal closure of containment isolation valves, and to control the operation of engineered safety features equipment.

During reactor operation in the startup and power modes, redundant safety limit signals will automatically actuate two reactor trip breakers, which are in series with the rod drive mechanism coils. This action would interrupt power and initiate reactor trip.

Reference sections:

Section Title	Section
Instrumentation and Control;	
General Design Criteria,	7.1
Protective Systems,	7.2
Nuclear Instrumentation,	7.4
Operating Control Stations	7.7

The reactor protection system receives signals from plant instrumentation, which are indicative of an approach to an unsafe operating condition, actuates alarms, prevents control rod motion, initiates load cutback, and/or opens the reactor trip breakers, depending on the severity of the condition.

The basic reactor tripping philosophy is to define a region of power and coolant temperature conditions allowed by the primary tripping functions (e.g., overpower ΔT trip, overtemperature ΔT trip, and nuclear flux trips). The allowable operating region within these trip settings is shown to prevent any combination of power, temperature, and pressure, which would result in DNB with all reactor coolant pumps in operation. Additional tripping functions such as low and high pressurizer pressure trips, high pressurizer level trip, loss of flow trip, steam and feedwater flow mismatch trip, steam generator low-low level trip, turbine trip, safety injection trip, nuclear flux trips (source, intermediate, and high range), and manual trip are provided as backup to the primary tripping functions for specific accident conditions and mechanical failures.

Intermediate Range and Power Range rod stops, Overtemperature ΔT and Overpower ΔT rod stops, are provided to prevent abnormal power conditions which could result from excessive control rod withdrawal initiated by operator violation of administrative procedures.

Reference sections:

Section Title	Section
Engineered Safety Features;	
Safety Injection System	6.2
Instrumentation and Control;	
Protective Systems	7.2

Positive indications in the control room of leakage of coolant from the reactor coolant system to the containment are provided by equipment, which permits continuous monitoring of containment air activity and humidity, and of runoff from the condensate collecting pans under the cooling coils of the containment air recirculation units. The basic design criterion is the detection of deviations from normal containment environmental conditions including air particulate activity, radiogas activity, humidity, condensate runoff and in addition, in the case of gross leakage, the liquid inventory in the process systems and containment sump.

The containment atmosphere, the plant vent, the containment fan-coolers service water discharge, the waste disposal system liquid effluent, and the component cooling loop are monitored for radioactivity concentration during all normal operations, anticipated transients, and accident conditions.

For the case of leakage from the reactor containment under accident conditions, the plant area radiation monitoring system supplemented by portable survey equipment to be kept in the control room provide adequate monitoring of accident releases.

Monitoring and alarm instrumentation is 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 and initiate containment isolation.

Reference sections:

Section Title	Section
Engineered Safety Features;	
Safety Injection System,	6.2
Isolation Valve Seal Water System,	6.5
Leakage Detection and Provisions for the Primary and Auxiliary Coolant Loops	6.7
Auxiliary & Emergency Systems;	
Auxiliary Coolant System	9.3
Waste Disposal & Radiation Protections System;	
Radiation Protection	11.2

1.3.4 Reliability And Testability Of Protection Systems (GDC 19 - GDC 26)

Upon a loss of power to the magnetic-type control rod drive mechanisms, the rod cluster control (RCC) assembly is released and falls by gravity into the core. The reactor internals, fuel assemblies, RCC assemblies and drive system components are designed as Class I equipment. The RCC assemblies are fully guided through the fuel assembly and for the maximum travel of

the control rod into the guide tube. Furthermore, the RCC assemblies are never fully withdrawn from their guide thimbles in the fuel assembly. Due to this and the flexibility designed into the RCC assemblies, abnormal loadings and misalignments can be sustained without impairing operation of the RCC assemblies.

All reactor protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power. The analog channel trip bistables have the capability to be bypassed for surveillance testing. This “test in bypass” feature enables analog logic relays in a protective channel to be bypassed while testing actuation of the associated bistable. Bistable testing does not preclude the protective action provided by concurrent channels.

Two reactor trip breakers are provided to interrupt power to the rod drive mechanisms. The breaker main contacts are connected in series with the power supply to the mechanism coils. Opening either breaker interrupts power to all full length rod mechanisms. Each breaker is opened through an undervoltage trip coil. Each protection channel actuates two separate trip logic trains, one for each reactor trip breaker undervoltage trip coil. The protection system is thus inherently safe in the event of a loss of rod control power.

Channel independence is carried throughout the system extending from the sensor to the relay providing the logic. In most cases, the safety and control functions when combined are combined only at the sensor. A failure in the control circuitry does not affect the safety channel. This approach is used for pressurizer pressure and water level channels, steam generator water level, T_{avg} and ΔT channels, steam flow, and nuclear power range channels. The power supplied to the channels is fed from four vital instrument buses. All four of the buses are supplied by static inverters.

The initiation of the engineered safety features provided for loss-of-coolant accidents (e.g., high head safety injection, residual heat removal pumps, and containment spray systems) is accomplished from redundant signals derived from reactor coolant system and containment instrumentation. The initiation signal for containment spray comes from coincidence of two sets of two-out-of-three high-high containment pressure signals. On loss of voltage to the safety features equipment buses, the emergency diesel generators will be automatically started and connected to their respective buses.

Trip signals for the containment isolation valves are derived from either a high-high containment pressure signal (phase B), and/or a safety injection signal (phase A), and/or a containment ventilation isolation signal.

The components of the protection system are designed and laid out so that the mechanical and thermal environment accompanying any emergency situation in which the components are required to function does not interfere with that function.

Each protection channel in service at power is capable of being calibrated and tripped independently by simulated signals for test purposes to verify its operation.

Redundancy is provided in that there are three emergency diesel-generator sets capable of supplying separate 480-V buses. The minimum complement of safety features equipment is supplied from any two of the three emergency diesel generators.

The ability of the emergency diesel-generator sets to start within the prescribed time and to carry load is periodically checked.

An open circuit or loss of reactor trip channel power causes the system to go into its trip mode. In a two-out-of-three circuit, the three channels are equipped with separate primary sensors and each channel is energized from independent electrical buses.

The signal for containment isolation is developed from a two-out-of-three circuit in which each channel is separate and independent and which signals for containment isolation upon loss of power. The failure of any channel to de-energize when required does not interfere with the proper functioning of the isolation circuit.

Diesel engine cranking is accomplished by a stored energy system supplied solely for the associated emergency diesel generator. The undervoltage relay scheme is designed so that loss of 480-V power does not prevent the relay scheme from functioning properly.

Reference sections:

Section Title	Section
Instrumentation and Control; Protection Systems	7.2
Electrical Systems	8

1.3.5 Reactivity Control (GDC 27 - GDC 32)

Reactivity control is achieved by two independent systems, the rod cluster control assemblies and the chemical and volume control system, which regulates the concentration of boric acid solution neutron absorber in the reactor coolant system. The system is designed to prevent, under anticipated system malfunction, uncontrolled or inadvertent reactivity changes, which might stress the system beyond design limits.

The reactivity control systems provided are capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes. The maximum excess reactivity expected for the core occurs for the cold, clean condition at the beginning-of-life (as discussed in Section 14.1.11.2).

The full length rod cluster control assemblies are divided into two categories comprising control banks and shutdown banks. The control bank of the rod cluster control assemblies is used to compensate for short-term reactivity changes at power produced due to variations in reactor power or in coolant temperature. The chemical shim control is used to compensate for the more slowly occurring changes in reactivity throughout core life such as those due to fuel depletion and fission product buildup and decay.

The shutdown banks are provided to supplement the control banks of the rod cluster control assemblies to make the reactor at least 1-percent subcritical (^{k}eff = 0.99) following trip from any credible operating condition to the hot, zero power condition assuming the most reactive rod cluster control assembly remains in the fully withdrawn position.

Boron injection from the safety injection system supplements rod insertion and prevents exceeding core safety limits in the event of the maximum credible steam break, namely opening of a safety valve. This is accomplished with maximum worth rod fully withdrawn.

Any time that the plant is at power, the quantity of boric acid retained in the boric acid storage tanks and ready for injection always exceeds that quantity required for the normal cold shutdown.

The boric acid solution is transferred from the boric acid storage tanks into the reactor coolant by the boric acid transfer pumps and charging pumps, which can be operated from emergency diesel-generator power on loss of primary power. Boric acid can be injected by either boric acid transfer pumps and one of the three charging pumps to shut down the reactor from full power with no rods inserted. In addition, boric acid can be injected to compensate for xenon decay (xenon decay below the equilibrium operating level will not actually begin until approximately 20 hr after shutdown). Additional boric acid is employed if it is desired to bring the reactor to cold shutdown conditions.

The reactor protection systems will limit reactivity transients such that DNBR remains at or above the applicable safety analysis DNBR limit due to any single malfunction in the reactor coolant deboration controls.

Limits, which include considerable margin, are placed on the maximum reactivity worth of control rods by limiting position of insertion as a function of power and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals so as to lose capability to cool the core.

The rod cluster drive mechanisms are wired into preselected groups and are prevented from being withdrawn in other than their respective groups. The rod drive mechanism is of the magnetic latch type and the coil actuation is sequenced to provide variable speed rod travel. The maximum reactivity insertion rate is analyzed in the detailed plant analysis assuming two of the highest worth banks to be accidentally withdrawn at maximum speed, yielding reactivity insertion rates of the order of 8×10^{-4} Δk/sec, which is well within the capability of the overpower-overtemperature protection circuits to prevent core damage (as discussed in Section 14.1.1).

Reference sections:

Section Title	Section
Reactor;	
Design Bases	3.1
Instrumentation and Control;	
Protective Systems,	7.2
Regulating Systems	7.3
Auxiliary & Emergency Systems;	
Chemical and Volume Control System	9.2

1.3.6 Reactor Coolant Pressure Boundary (GDC 33 - GDC 36)

The reactor coolant pressure boundary is shown to be capable of accommodating without further rupture, the static and dynamic loads imposed as a result of a sudden reactivity insertion such as rod ejection (as discussed in Section 14.2.6.10).

The operation of the reactor is such that the severity of an ejection accident is inherently limited. Since the rod cluster control assemblies are used to control load variations only and core depletion is followed with boron dilution, only the rod cluster control assemblies in the controlling

groups are inserted in the core at power, and at full power these rods are only partially inserted. A rod insertion limit monitor is provided as an administrative aid to the operator to ensure that this condition is met.

By using the flexibility in the selection of control rod groupings, radial locations and positions as a function of load, the design limits the maximum fuel temperature for the highest worth ejected rod to a value, which precludes any resultant damage to the primary system pressure boundary.

The failure of a rod mechanism housing causing a rod cluster control assembly to be rapidly ejected from the core is evaluated as a theoretical, though not a credible, accident. The analysis is discussed in Section 14.2.6.

In the core region of the reactor vessel, the V-notch toughness of the material will change during operation as a result of fast neutron exposure, which results in a shift in the nil ductility transition temperature (NDTT). This is factored into the operating procedures in such a manner that full operating pressure is not obtained until the affected vessel material is above the increased design transition temperature (DTT) and in the ductile material region. The pressure during startup and shutdown at the temperature below NDTT is maintained below the threshold of concern for safe operation.

The DTT is a minimum NDTT plus 60^{o}F and dictates the procedures to be followed in the hydrostatic test and in station operations to avoid excessive cold stress. The value of the DTT is increased during the life of the plant as required by the expected shift in the NDTT, and as confirmed by the experimental data obtained from irradiated specimens of reactor vessel materials during the plant lifetime.

The design of the reactor vessel and its arrangement in the system provide the capability for accessibility during service life to the entire internal surfaces of the vessel and the external zones of the vessel including the nozzle to reactor coolant piping welds and the top and bottom heads. The reactor arrangement within the containment provides sufficient space for inspection of the external surfaces of the reactor coolant piping, except for the area of piping within the primary shielding concrete.

Determination of the NDTT of the core region plate forgings, weldments, and associated heat treated zones is performed in accordance with ASTM E-185, Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors. Samples of reactor vessel plate material have been retained and catalogued in case future engineering development shows the need for further testing.

The material properties surveillance program includes not only the conventional tensile and impact tests, but also fracture mechanics specimens. The observed shifts in NDTT of the core region materials with irradiation will be used to confirm the calculated limits to startup and shutdown transients.

To define permissible operating conditions below DTT, a pressure range is established, which is bounded by a lower limit for pump operation and an upper limit, which satisfies reactor vessel stress criteria. Since the normal operating temperature of the reactor vessel is well above the maximum expected DTT, brittle fracture during normal operation is not considered to be a credible mode of failure.

Reference sections:

Section Title	Section
Reactor Coolant System;	
System Design and Operation,	4.2
Safety Limits and Conditions,	4.4
Inspections and Tests,	4.5
Determination of Reactor Pressure Vessel NDTT	Appendix 4A

1.3.7 Engineered Safety Features (GDC 37 - GDC 65)

The design, fabrication, testing and inspection of the core, reactor coolant pressure boundary, and their protection systems give assurance of safe and reliable operation under all anticipated normal, transient, and accident conditions. However, engineered safety features are provided in the facility to back up the safety provided by these components.

These engineered safety features have been designed to cope with any size reactor coolant pipe break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends as discussed in Section 14.3.3.3. They are also designed to cope with any steam or feedwater line break up to and including the main steam or feedwater headers as discussed in Section 14.2.5. The total loss of all offsite power is assumed concurrent with these accidents.

The release of fission products from the reactor fuel is limited by the safety injection system, which by cooling the core and limiting the fuel clad temperature, keeps the fuel in place and substantially intact with its essential heat transfer geometry preserved and limits the metal-water reaction to an insignificant amount.

The basic design criteria for ensuring that the core geometry remains in place and substantially intact so that effective cooling of the core is not impaired following a loss-of-coolant accident:

1. The cladding temperature is to be less than:
    a. The melting temperature of Zircaloy-4
    b. The temperature at which gross core geometry distortion, including fragmentation, may be expected.

2. The total core metal-water reaction will be limited to less than 1-percent.

The safety injection system (which constitutes the emergency core cooling system) consists of high and low head centrifugal pumps driven by electric motors, and passive accumulator tanks, which are self-energized and which act independently of any actuation signal or power source.

The release of fission products from the containment is limited in three ways:

1. Blocking the potential leakage paths from the containment. This is accomplished by:
    a. A steel-lined concrete reactor containment with continuously pressurized double-sealed penetrations and liner weld channels, which form a virtually leaktight barrier to the escape of fission products should a loss-of-coolant accident occur.

    b. Isolation of process lines by the containment isolation system, which imposes double barriers in each line penetrating the containment.

2. Reducing the fission product concentration in the containment atmosphere. This is accomplished by containment spray, which removes elemental iodine vapor and particulates from the containment atmosphere by washing action.

3. Reducing the containment pressure and thereby limiting the driving potential for fission product leakage. This is accomplished by cooling the containment atmosphere by the following independent systems, each with adequate heat removal capacity:
   a. Containment spray system
   b. Containment air recirculation cooling system.

A comprehensive program of plant testing is performed for all equipment systems and system control vital to the functioning of engineered safety features. The program consists of performance tests of individual pieces of equipment in the manufacturer's shop and integrated tests of the system as a whole, and periodic tests of the actuation circuitry and mechanical components to assure reliable performance, upon demand, throughout the plant lifetime. In the event that one of the components should require maintenance as a result of failure to perform during the test according to prescribed limits, the necessary corrections or minor maintenance will be made and the unit retested immediately.

The plant is supplied with emergency power sources as follows:

1. Three independent emergency diesel generators, located in the Diesel Generator Building adjacent to the Primary Auxiliary Building, supply emergency power to the engineered safety features buses in the event of a loss of AC auxiliary power. There are no automatic bus ties associated with these buses. Each diesel generator is started automatically on a safety injection signal or upon the occurrence of an undervoltage condition on any vital 480-V switchgear bus. The system is sufficiently redundant such that any two diesels have adequate capacity to supply the engineered safety features for the design basis accident concurrent with a loss of offsite power. One diesel is adequate to provide power for a safe and orderly plant shutdown in the event of a loss-of-offsite electrical power.

2. Emergency power for vital instrumentation and control and for emergency lighting is supplied from the 125 VDC system via four independent DC channels. The station batteries supply emergency power to the instrumentation and control systems when their associated battery chargers are not available.

For such engineered safety features as are required to ensure safety in the event of an accident, protection from dynamic effects or missiles is considered in the layout of plant equipment and missile barriers.

Layout and structural design specifically protect injection paths leading to unbroken reactor coolant loops against damage as a result of the maximum reactor coolant pipe rupture. Injection lines penetrate the main missile barrier, and the injection headers are located in the missile-protected area between the missile barrier and the containment outside wall. Individual injection lines, separated to the maximum extent practicable, are connected to the injection header, pass through the barrier and then connect to the loops. Movement of the injection line, associated with rupture of a reactor coolant loop, is accommodated by line flexibility and by the design of the pipe supports such that no damage outside the missile barrier is possible.

In 1989, the NRC approved changes to the design basis with respect to dynamic effects of postulated primary loop ruptures, as discussed in Section 4.1.2.4.

Each engineered safety feature provides sufficient performance capability to accommodate any single failure of an active component and still function in a manner to avoid undue risk to the health and safety of the public.

Under the hypothetical accident conditions, the containment air recirculation cooling system, and the containment spray system are designed and sized so that either system, operating alone at its rated capacity, is able to supply the necessary postaccident cooling capacity to reduce rapidly the containment pressure following blowdown and cooling of the core by safety injection. Together these two systems provide the single failure protection for the containment cooling function as analyzed in Chapter 14.

All active components of the safety injection system (exception: injection line isolation valves) and the containment spray system are located outside the containment and not subject to containment accident conditions.

Instrumentation, motors, cables, and penetrations located inside the containment are selected to meet the most adverse accident conditions to which they may be subjected. These items are either protected from containment accident conditions or are designed to withstand, without failure, exposure to the worst combination of temperature, pressure, radiation, and humidity expected during the required operational period.

The reactor is maintained subcritical following a primary system pipe rupture accident. Introduction of borated cooling water into the core results in a net negative reactivity addition (as discussed in Section 14.3.2). The control rods insert and remain inserted, except for the large break loss of coolant accident analysis where it is conservatively assumed that the control rods do not insert as discussed in Section 14.3.3.2. The delivery of cold safety injection water to the reactor vessel following accidental expulsion of reactor coolant was evaluated to ensure that this does not cause further loss of integrity of the reactor coolant system boundary (as discussed in Section 14.3.4.3.3).

Design provisions have been made to the extent practical to facilitate access to the critical parts of the reactor vessel internals, injection nozzles, pipes, valves, and safety injection pumps for visual or boroscopic inspection for erosion, corrosion, and vibration wear evidence, and for non-destructive test inspection where such techniques are desirable and appropriate.

Design provisions are made so that active components of the safety injection system can be tested periodically for operability and functional performance. The safety injection pumps can be tested periodically during plant operation using the minimum flow recirculation lines provided. The residual heat removal pumps are used every time the residual heat removal loop is put into operation.

An integrated safety injection system test is performed at refueling outage intervals. This test does not introduce flow into the reactor coolant system but demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry upon initiation of safety injection.

The accumulator tank pressure and level are continuously monitored during plant operation and flow from the tanks can be checked at any time using test lines.

The accumulators and the safety injection piping up to the final isolation valve are maintained sufficiently filled of borated water at refueling water concentration while the plant is in operation to ensure the system remains operable and performs properly. Flow in each of the high head injection headers and in the main flow line for the residual heat removal pumps is monitored by flow and pressure instrumentation.

The design provided for capability to test initially, to the extent practical, the full operational sequence up to the design conditions for the safety injection system to demonstrate the state of readiness and capability of the system. These functional tests provided information to confirm valve operating times, pump motor starting times, the proper automatic sequencing of load addition to emergency diesel generators, and delivery rates of injection water to the reactor coolant system.

The following general criteria were followed to ensure conservatism in computing the required containment structural load capacity:

1. In calculating the containment pressure, rupture sizes up to and including a double-ended severance of reactor coolant pipe were considered.

2. In considering postaccident pressure effects, various malfunctions of the emergency systems were evaluated. Contingent mechanical or electrical failures were assumed to disable one of the emergency diesel electric generators, two of the fan coolers and one of the containment spray pumps (as discussed in Section 14.3.5.3.7).

3. The pressure and temperature loading obtained by analyzing various loss-of-coolant accidents, when combined with operating loads and maximum wind or seismic forces, does not exceed the load-carrying capacity of the structure, its access opening or penetrations.

Discharge of reactor coolant through a double-ended rupture of the main loop piping, followed by operation of only those engineered safety features, which can run simultaneously with power from two of the three emergency onsite diesel generators, results in a sufficiently low radioactive materials leakage from the containment structure that there is no undue risk to the health and safety of the public.

The concrete containment is not susceptible to a low temperature brittle fracture. The containment liner is enclosed within the containment and thus is not exposed to the temperature extremes of the environs. The containment ambient temperature during operation is between 90°F and 130°F, which is well above the NDTT + 30°F for the liner material. Containment penetrations, which can be exposed to the environment are also designed to the NDTT + 30°F criterion.

The reactor coolant pressure boundary does not extend outside of the containment. Isolation valves for all fluid system lines penetrating the containment provide at least two barriers against leakage of radioactive fluids to the environment in the event of a loss-of-coolant accident. These barriers, in the form of isolation valves or closed systems, are defined on an individual line basis. In addition to satisfying containment isolation criteria, the valving is designed to facilitate normal operation and maintenance of the systems and to ensure reliable operation of other engineered safety features.

After completion of the containment structure and installation of all penetration and weld channels, an initial integrated leakage rate test was conducted at the peak calculated accident pressure, maintained for a minimum of 24 hr, to verify that the leakage rate was not greater than 0.1-percent by weight of the containment volume per day. This leakage rate test was performed using the reference vessel method.

A leak rate test at the peak calculated accident pressure using the same method as the initial leak rate test can be performed at any time during the operational life of the plant, provided the plant is not in operation and precautions are taken to protect instruments and equipment from damage.

Penetrations are designed with double seals so as to permit pressurization of the interior of the penetration whenever a leak test is required. The system utilizes a supply of clean, dry, compressed air which places all the penetrations under an internal pressure above the peak calculated accident pressure (Peak calculated accident pressure is discussed in Section 14.3.5.1.1). Leakage from the system is checked by measurement of the integrated makeup air flow. In the event excessive leakage is discovered, each penetration can then be checked separately.

Capability is provided to the extent practical for testing the functional operability of valves and associated apparatus during periods of reactor shutdown.

Initiation of containment isolation employs coincidence circuits, which allow checking of the operability and calibration of one channel at a time.

The main steam and feedwater barriers and isolation valves in systems, which connect to the reactor coolant system are hydrostatically tested to measure leakage.

Design provisions are made to the extent practical to facilitate access for periodic visual inspection of all important components of the containment air recirculation cooling and containment spray systems.

The containment pressure-reducing systems are designed to the extent practical so that the spray pumps, spray injection valves, spray nozzles can be tested periodically and after any component maintenance for operability and functional performance.

Permanent test lines for the containment spray loop are located so that all components up to the isolation valve at the spray nozzle may be tested. These isolation valves are checked separately.

The air test lines for checking that spray nozzles are not obstructed connect upstream of the isolation valve. Air flow through the nozzles is verified by periodic testing in accordance with the Technical Specifications.

Capability is provided to test initially to the extent practical the operational startup sequence beginning with transfer to alternate power sources and ending with near design conditions for the containment air recirculation cooling and containment spray systems.

Reference sections:

Section Title	Section
Containment System	5
Engineered Safety Features	6
Electrical Systems;	
Design Bases,	8.1
Electrical Systems Design	8.2

1.3.8 Fuel And Waste Storage Systems (GDC 66 - GDC 69)

The new and 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, normally at a similar concentration to that used in the reactor cavity and refueling canal during refueling operations. 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 ≤0.95 when filled with water borated ≥ 2000 ppm boron.

During reactor vessel head removal and while loading and unloading fuel from the reactor, the boron concentration is maintained at not less than that required to maintain the reactor subcritical by 5-percent $\Delta k/k$ with all the rods inserted. The refueling water boron concentration is periodically checked to ensure the proper shutdown margin.

The design of the fuel handling equipment incorporating built-in interlocks and safety features, the use of detailed refueling instructions and observance of minimum operating conditions provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety. The refueling system interlocks are verified to be functioning each refueling shutdown prior to refueling operations.

The refueling water provides a reliable and adequate cooling medium for spent fuel transfer. Heat removal from the refueling water is provided by an auxiliary cooling system.

Adequate shielding for radiation protection is provided during reactor refueling 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 operating personnel. Pit water level is alarmed and water to be removed from the pit must be pumped out as there are no gravity drains when the pit is isolated from the refueling canal. 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 refueling canal and spent fuel storage pit are reinforced concrete structures with a seam-welded stainless steel plate liner. These structures are designed to withstand any anticipated earthquake loadings as seismic Class I structures so that the liner should prevent leakage even in the event the reinforced concrete develops cracks.

Reference sections:

Section Title	Section
Auxiliary & Emergency Systems;	
Sampling System	9.4
Waste Disposal & Radiation Protection System;	
Waste Disposal System	11.1
Radiation Protection Systems	11.2

1.3.9 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 operation, gases are discharged intermittently at a controlled rate from these tanks through the monitored plant 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	11.1

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.

2. Deleted

1.4 DESIGN PARAMETERS AND PLANT COMPARISON

1.4.1 Design Highlights

The original design of the plant is based upon proven concepts, which have been developed and successfully applied in the construction of pressurized water reactor systems. In subsequent paragraphs, the original design features of the plant are discussed.

1.4.1.1 Power Level

The initial license application power level for Indian Point Unit 2 was 2758 MWt. The increase in this power rating over 1473 MWt for Connecticut Yankee was achieved by a 44-percent increase in heat transfer surface area and a 31-percent increase in average heat flux. The increased heat transfer surface area is due to 22-percent more fuel rods, each 20-percent longer.

The increase in maximum heat flux and the 13.4 kW/ft linear heat generation rate (LHGR) resulting are justified by the results of incore experiments by Westinghouse and others.

1.4.1.2 Reactor Coolant Loops

The reactor coolant system for the Indian Point Unit 2 consists of four loops as compared with three loops for San Onofre and four loops for Connecticut Yankee. The use of four loops in the Indian Point Unit 2 for the production of 2758 MWt requires an attendant increase in the size and capacity of the reactor coolant system components such as the reactor vessel, reactor coolant pumps, piping, and steam generators. These increases represent reasonable engineering extrapolations of existing and proven designs.

1.4.1.3 Peak Specific Power

Based on values for hot channel factors, reactivity coefficients, and other design parameters, which were established in the PSAR and are supported by the previous experience with other plants of the same type, this reactor can be operated safely at power levels at least as high as the license application rating.

1.4.1.4 Fuel Cladding

The fuel rod design for the plant employs zircaloy as a cladding material. This clad has proven successful in numerous operating facilities. The fuel rod dimensions are identical to those in Ginna, Salem, and Zion Station Units 1 and 2.

1.4.1.5 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. Extensive out-of-pile and in-pile tests have been performed on this concept; operating experience is available from numerous facilities.

1.4.1.6 Moderator Temperature Coefficient of Reactivity

The reactor has a negative moderator temperature coefficient of reactivity at operating temperature at all times throughout core life (as discussed in Section 14.1.11.2).

1.4.2 IP2 - IP3 Design Differences

An NRC Confirmatory Order (Reference 1) required the Licensees (Consolidated Edison and the Power Authority) to jointly review and identify the significant differences between Indian Point Units 2 and 3 and evaluate these differences in light of the regulatory standards and requirements in existence at the time. Consolidated Edison determined, evaluated, and provided justification for each design difference in submittals to the NRC for acceptance (References 2, 3). These design differences were found acceptable in an NRC Safety Evaluation Report (Reference 4).

REFERENCES FOR SECTION 1.4

1. Letter from Harold R. Denton, NRC, to Consolidated Edison, "Confirmatory Order", dated February 11, 1980.

2. Letter from William J. Cahill, Consolidated Edison, to Harold R. Denton, NRC, "Confirmatory Order", dated May 9, 1980.

3. Letter from John D. O'Toole, Consolidated Edison, to Steven A. Varga, NRC, "Confirmatory Order", dated May 27, 1982.

4. Letter from Steven A. Varga, NRC, to John D. O'Toole, Consolidated Edison, "Confirmatory Order", dated December 1, 1982.

1.5 RESEARCH AND DEVELOPMENT REQUIREMENTS

Research and development were conducted relating to finalization of core design details and parameters, air recirculation system halogen filters, failure of core cooling systems and means to ameliorate consequences, emergency core cooling system, control rod ejection analysis, and reactor coolant pump controlled leakage seals.

The detailed final core design and thermal-hydraulics and physics parameters have been completed. The nuclear design including fuel configuration and enrichments, control rod pattern and worths, reactivity coefficients and boron requirements are presented in Section 3.2.1 and the final thermal-hydraulics design parameters are in Section 3.2.2. Section 3.2.3 presents the final fuel, fuel rod, fuel assembly, and control rod mechanical design. The core design incorporates fixed burnable poison rods[1] in the initial loading to ensure a negative moderator reactivity temperature coefficient at operating temperature. This improves reactor stability and lessens the consequences of a rod ejection or loss-of-coolant accident.

Core stability has been analyzed[2-4] and design provisions for detection and control of potential xenon oscillations have been finalized[3]. The original core design incorporated part-length control rods for controlling these xenon oscillations and shaping the axial power distribution.

These have since been found unnecessary and removed from the reactor. X-Y control is not required and therefore not provided. Tests in operating reactors demonstrate the ability of the out-of-core instrumentation to give accurate indication of power redistribution and provide the operator information necessary to monitor redistributions and control axial oscillations by moving the rods in a prescribed pattern[3]. This capability was verified during startup tests in the Indian Point Unit 2 Plant.

Full-size filter tests were conducted for the Connecticut Yankee Atomic Power Company to demonstrate the efficiency for iodine removal under the most extreme conditions anticipated in the postaccident containment environment. The results of these tests[5] filed with the former U. S. Atomic Energy Commission (now the U. S. Nuclear Regulatory Commission) under Docket No. 50-213, are directly applicable to the charcoal filter system originally employed in this plant. The charcoal filters are no longer required by the Technical Specifications and the radiological consequences analysis presented in Section 14.3.6 does not credit the filters.

A program for development of a crucible system design, which would contain the reactor core assuming failure of the core cooling system to prevent a core meltdown, was undertaken. A scheme for containing the molten core in a water submerged high melting point refractory lined steel crucible resulted. Refractory materials and crucible physical design were investigated along with analysis of the temperature distribution expected with the molten core and crucible refractory, and steam and water recirculation paths. As a result of uncertainties in material properties at the high application temperature, the lack of experimental proof that the boiling core mass would dissipate its heat upward through the water cover, and the possibility of violent liquid metal-water reactions, it became apparent that the proper emphasis for research and development on the loss-of-coolant accident should be increased emphasis placed on research and development for emergency core cooling system improvement in order to eliminate need for a crucible. This is supported by the conclusions of the Report of Advisory Task Force on Power Reactor Emergency Cooling, "Emergency Core Cooling," USAEC.

This additional development effort on emergency core cooling system design resulted in the modification of the system to include pressurized accumulator tanks for rapid core reflooding. This increased flooding capability limits the clad temperature after a loss-of-coolant accident to well below the melting temperature of Zircaloy-4, minimizes metal-water reaction and ensures that the core remains in place and intact thereby ensuring preservation of essential heat transfer geometry. The system design incorporates redundancy of components such that the minimum required water addition rates can be met assuming any active component to fail concurrent with the loss-of-coolant accident or, over the long term period of postaccident core decay heat removal a passive or active component failure in either the safety injection or service water systems, or an active failure in the component cooling water system. Details of system design and operation are given in Chapters 6 and 9, and analysis of the loss-of-coolant accident is presented in Section 14.3. Because of the incorporation of this revised emergency core cooling system, the reactor pit crucible was deleted from the plant design. Although it is not required to provide cooling for molten fuel in the bottom of the reactor vessel with the upgraded emergency core cooling system performance, the clearance between the insulation and the instrumentation penetrations with the pressure relief holes in the insulation at the top of the vessel provide assurance that water in the flooded reactor vessel cavity will be in contact with the vessel. No other provisions for direct vessel cooling are provided or required.

A control rod ejection analysis was performed for the final core design, rod worths, rod position limits, and moderator reactivity temperature coefficient. As mentioned above, the addition of the burnable poison rods eliminates power operation with a positive moderator temperature

coefficient and reduces the severity of the ejected rod accident, hence, lessening the need for research and development on this subject. The analysis is presented in Section 14.2.6.

The reactor coolant pump controlled leakage seal design for this plant has been fully developed. A full scale mock-up of this seal was operated for over 100 hr to confirm that seal deflection and leak rate under load were acceptable. The full scale mock-up has been used during the development of the controlled leakage seal to provide information related to long-term performance. One of the two seals used in this plant was operated about 300 hr and the other about 100 hr, each in its pump motor unit. During hot functional testing in the plant, before the core was loaded, additional operation brought the total operating time for each seal to well over 500 hr. Successful operation of similar seals had previously been demonstrated with over 5000 hr total running time in San Onofre and over 3000 hr in Haddam Neck.

REFERENCES FOR SECTION 1.5

1. P. M. Wood, E. A. Bassler, et al., Use of Burnable Poison Rods in Westinghouse Pressurized Water Reactors, WCAP-7113, October 1967.

2. P. M. Wood, J. M. Gallagher, R. M. Metz, R. A. Dean, Use of Part-Length Absorber Rods in Westinghouse Pressurized Water Reactors, WCAP-7072, June 1967.

3. Westinghouse Electric Corporation, Power Distribution Control of Westinghouse Pressurized Water Reactors, WCAP-7208, October 1968.

4. Westinghouse Electric Corporation, Power Maldistribution Investigations, WCAP-7407-L (proprietary), January 1970.

5. Connecticut Yankee Charcoal Filter Tests, CYAP-101, December 1966.

1.6 IDENTIFICATION OF CONTRACTORS [Historical Information Only]

The Indian Point Unit 2 was designed and built by the Westinghouse Electric Corporation as prime contractor for Con Edison. Westinghouse undertook to provide a complete, safe, and operable nuclear power plant ready for commercial service. The project was directed by Westinghouse from the offices of its Atomic Power Division in Pittsburgh, Pennsylvania, and by Westinghouse representatives at the plant site during construction and plant startup. Westinghouse engaged United Engineers and Constructors of Philadelphia, Pennsylvania, to provide the design of certain portions of the plant.

The plant construction was under the general direction of Westinghouse through United Engineers and Constructors, which was responsible for the management of all site construction activities and either performed or subcontracted the work of construction and equipment erection. Preoperational testing of equipment and systems at the site and initial plant operation was performed by Con Edison personnel under the technical direction of Westinghouse.

1.7 PROJECT REORGANIZATION - DECEMBER 1969 [Historical Information Only]

This section describes a reorganization in project management, which was implemented by Westinghouse in December 1969.

Westinghouse formed a wholly-owned subsidiary corporation, called WEDCO Corporation, to perform certain functions at the Indian Point site of Con Edison. Westinghouse remained the prime contractor and continued to exercise overall control and to have full responsibility for the Indian Point 2 project. WEDCO performed, under Westinghouse, project management, engineering, quality assurance, construction, and procurement functions for Indian Point Unit 2. These functions were previously carried out by Westinghouse or United Engineers and Constructors (UE&C).

The entire Westinghouse senior management organization, which prior to the advent of WEDCO, was responsible for the Westinghouse effort at Indian Point Unit 2, remained responsible. All other personnel within Westinghouse senior management who, prior to WEDCO, carried any responsibility in any area for Indian Point Unit 2 continued to carry those responsibilities, regardless of the formation of WEDCO or changes in title or designation. Furthermore, WEDCO had behind it the full organization and strength of the Westinghouse Electric Corporation: Westinghouse engineering, legal, and other personnel continued to be available for the project.

The functional relationships among Con Edison, Westinghouse, WEDCO, and UE&C are shown in Historical Figure 1.7-1.

Westinghouse WEDCO-UE&C Relationship.

Westinghouse retained UE&C as its architect-engineer-constructor to perform certain work and services in connection with the plant. Initially, UE&C performed services within its scope in the following areas:

1. Design and Engineering

2. Procurement

3. Construction Management and Construction

4. Quality Assurance (including Home Office Quality Control Engineering, Vendor Surveillance and Onsite Quality Control)

Westinghouse removed items (2) and (3) from the scope of work to be performed by UE&C and assigned these functions to WEDCO. In these areas, however, UE&C provided qualified personnel to assist in effectuating the transition of work to Westinghouse and WEDCO.

UE&C continued to have responsibility for all of the design and engineering functions and all of the quality assurance functions, including home office quality control engineering, vendor surveillance and onsite quality control, for which it had responsibility prior to the advent of WEDCO. UE&C continued to have direct corporate responsibility to Westinghouse for all of the work within its scope.

In its organizational structure, WEDCO exercised a high level quality and engineering reliability function. This function included the activities previously performed by the Nuclear Power Service Staff Resident Quality Assurance Engineer, and in addition included the centralization and overall management for quality assurance activities previously performed by various organizations. This function was carried out by a Reliability Manager who was located at the site. The Reliability Manager was responsible for surveillance visits to selected shops or suppliers. This function was previously delegated to the Westinghouse Nuclear Power Services Group. In addition, the Reliability Manager continually audited the quality assurance efforts of UE&C. In effect, a new reliability management function over and above those previously set forth was established while all existing organizational functions and responsibilities for quality assurance were maintained.

The quality control functions previously performed at various Westinghouse organizational levels continued to be performed. At the Westinghouse headquarters level, the staff quality assurance audit team reviewed periodically the quality control program for Indian Point Unit 2 as it had done in the past. At the Westinghouse PWR Systems Division level, the quality control functions performed by that division for the nuclear steam supply system continued as before.

Con Edison.

The project reorganization described did not in any way alter the ultimate responsibility of Con Edison for the quality assurance program. There was no basic change in the Con Edison program. However, the following minor procedural changes were made in view of the existence of WEDCO:

1. Con Edison's monitoring function included monitoring the activities of WEDCO.

2. Con Edison forwarded the United States Testing Company quality assurance reports to Westinghouse and/or WEDCO.

3. Con Edison contacted Westinghouse and/or WEDCO for necessary corrective action.

1.7 FIGURES

Figure No.	**Title**
Figure 1.7-1	Functional Relationships [Historical]

1.8 PROJECT REORGANIZATION - MARCH 1970 [Historical Information Only]

This section describes a change, which was implemented by Westinghouse in the spring of 1970 in the project organization.

The changes made in December 1969 (see Section 1.7) involved the creation of WEDCO and the delegation to WEDCO by Westinghouse of certain functions at the Indian Point site previously carried out by Westinghouse or United Engineers and Constructors (UE&C). Following the December 1969 reorganization, UE&C retained the following functions within its scope of work as architect-engineer-constructor:

1. Design and Engineering.

2. Quality Assurance (including Home Office Quality Control Engineering, Vendor Surveillance, and Onsite Quality Control).

The change consisted of the removal of the vendor surveillance and onsite quality control portions of item (2) from the scope of work to be performed by UE&C, and the assigning of these functions to WEDCO.

There was little change of personnel involved in the transfer of the onsite quality control function. WEDCO employed a Manager of Vendor Surveillance and other personnel for this work. The transition in this respect was gradual. New personnel were phased in and the UE&C personnel were used during the transition period to ensure continuity of the surveillance program. The transfer was made on a purchase-order-by-purchase-order basis, with UE&C personnel working with new personnel in performing the surveillance during the transition.

To assure that a level of quality assurance review was not lost, the organization of the WEDCO reliability group was structured to provide for an independent, internal audit of the two quality assurance functions transferred to WEDCO. The Vendor Surveillance Group and Onsite Quality Control Group each reported directly to the Reliability Manager.

The activities of both the Vendor Surveillance Group and the Onsite Quality Control Group were audited by a Systems Reliability Group. The Systems Reliability Group reported directly to the Reliability Manager to ensure its functional independence. Historical Figure 1.8-1 shows these organizational relationships in chart form.

1.8 FIGURES

Figure No.	**Title**
Figure 1.8-1	Organization Chart WEDCO Reliability Group [Historical]

1.9 SUPPLEMENTS AND REVISIONS TO ORIGINAL FSAR

1.9.1 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 Tave to 565^{o}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.

1.9.2 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 1	May 2003
Revision 18,	October 2003
Revision 19,	May 2005
Revision 20,	November 2006
Revision 21,	October 2008
Revision 22,	October 2010

Revision 2, in addition to reflecting required changes, incorporated a major editorial effort to standardize the format of the updated FSAR and to correct typographical, grammatical, and syntax errors, so that the material is presented in a more uniform and clear manner. The changes of technical content and some major editorial changes were marked in the margins with the numeral 2. The majority of editorial changes were minor and were not marked individually. A changed page, however, was indicated by the label Revision 2 at the lower right hand corner.

1.10 QUALITY ASSURANCE PROGRAM

1.10.1 General

Entergy's Quality Assurance Program (QAP) for Indian Point Unit 2 is in accordance with the quality assurance requirements of 10 CFR 50 Appendix B. The QAP is described in a Quality Assurance Program Manual, which satisfies the criteria of Appendix B. Changes to the program description are submitted to the NRC in accordance with the provisions of 10 CFR 50.54(a)(3).

1.10.2 Scope

The Quality Assurance Program provides control for activities affecting the quality of structures, systems, and components of the plant and their operation to the extent consistent with their

importance to safety. Those structures, systems, and components of the plant that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public are designated "Class A" as described in the Quality Assurance Program. 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 Class A items is maintained. Elements of the Quality Assurance Program are also applicable to activities and items affecting safety as defined in Licensing commitments.[2]

It is recognized that not every portion of each of the listed systems and components affect the safety related function. Therefore, allowance is made for subcomponents of systems to be declassified. When such is the case, the agreement is appropriately documented identifying the parts or subcomponents concerned and showing appropriate concurrences.

1.10.3 Organization And Responsibilities

The major organizations or groups participating in the Quality Assurance Program are: Nuclear Quality Assurance and Oversight, Nuclear Power, Nuclear Power Engineering, and the Safety Review Committee. The duties and responsibilities of the individuals participating in the Quality Assurance Program are described in procedures, or manuals.

REFERENCES FOR SECTION 1.10

1. Deleted
2. 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.

TABLE 1.10-1
DELETED

1.11 DESIGN CRITERIA FOR STRUCTURES AND COMPONENTS

1.11.1 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 and components including instruments and controls whose failure might cause or increase the severity of a loss-of-coolant accident or result in an uncontrolled release of radioactivity causing more than 10 rem to the thyroid or 10

rem whole body to the average adult beyond the nearest site boundary. Also included are those structures and components vital to safe shutdown and isolation of the reactor.

Class II

Class II is defined as those structures and components, which are important to reactor operation but not essential to safe shutdown and isolation of the reactor and whose failure could result in the release of radioactivity causing more than 1.0 rem to the thyroid or 0.5 rem whole body dose to the average adult beyond the nearest site boundary.

Class III

Class III is defined as those structures and components, which are not directly related to reactor operation or containment. In Indian Point Unit 2, the only portions of the plant, which are not seismic Class I and which might carry substantial radioactivity because of required safeguards operation or requirements for safe shutdown and isolation of the reactor are portions of the chemical and volume control system and waste disposal system.

The specific components in the chemical and volume control system are the volume control tank, holdup tank, and the concentrates holding tank with associated piping, valves and supports. These components are all seismic Class I. In addition, the design of the system tanks and their location were based upon the commitment that a vessel rupture would not cause doses in excess of 10 CFR 20 limits at the exclusion radius.

The specific components in the waste disposal system are the gas decay tanks with their associated piping, valves and supports. These components are all seismic Class I. In addition, the gas decay tanks of the waste disposal system have been designed such that the failure of any tank will not exceed 10 CFR 20 doses at the exclusion radius.

The analysis showing that the rupture of the volume control tank or a gas decay tank does not exceed the special dose limits selected for Indian Point Unit 2 is found in Section 14.2.3 of the FSAR.

Those components of the chemical and volume control system that are not seismic Class I are as follows: batching tank, monitor tanks, monitor tank pumps, chemical mixing tank, and resin fill tank. In addition, the boric acid evaporator and the condensate demineralizer are not seismic Class I.

Those components of the waste disposal system, which are not seismic Class I include: waste condensate tank and pumps.

Failure of these components will not result in offsite doses in excess of 10 CFR 20 limits at the site exclusion radius.

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 1.11-1 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 1.11-1 and 1.11-2.

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 1.11.4.

1.11.2 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 (including all penetrations and airlocks, the concrete shield, the liner, and the interior structures)	I
Spent fuel pit	I
Control Building	I
Diesel Generator Building	I
Intake structure (to the extent that water is always available to the service water pumps)	I
Service water screenwell	I
Primary Auxiliary Building	I
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.*]	
Reactor control and protection system	I
Radiation monitoring system	I
Process instrumentation and controls	I

Reactor Vessel and its supports	I
Vessel internals Fuel assemblies Rod cluster control assemblies and drive mechanisms Supporting and positioning members Incore instrumentation structure	
Reactor coolant system Piping and valves (including safety and relief valves) Steam generators Pressurizer Reactor coolant pumps Supporting and positioning members	I
Main Steam system, up to and including the isolation valve	I
Engineered safety features Safety injection system (including safety injection and residual heat removal pumps, refueling water storage tank, accumulator tanks, residual heat removal heat exchangers and connecting piping and valving) Containment spray system (including spray pumps, spray headers, and connecting piping and valving) Containment air recirculation cooling system (including fans, coolers, ducts, valves, and demisters)	I
Auxiliary building ventilation system	I
Condensate storage tanks	I
Pressurizer relief tank	II
Residual heat removal loop	I
Containment penetration and weld channel pressurization system	I
Component cooling loop	I
Instrument air system (essential sections)	I
Isolation valve seal water system	I
Sampling system	II
Spent fuel pit cooling loop	II
Fuel transfer tube	I

Emergency power supply system	I
Diesel generators and fuel oil storage tank	
DC power supply system	
Power distribution lines to equipment required for transformers and switchgear supplying the engineered safety features	
Control panel boards	
Motor control centers	
Control Equipment, facilities and lines necessary for the above seismic Class I items	I
Waste disposal system	I
Chemical drain tank	
Waste holdup tanks	
Gas decay tanks	
Reactor coolant drain tank	
Compressors	
Waste holdup tank pumps	
Interconnecting waste gas piping	
Waste disposal system	II or III
All elements not listed as seismic Class I	
Containment crane	I
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	I
The chemical and volume control system is considered seismic Class I except for the components listed below:	
Batching tank	II
Monitor tanks	III
Monitor tank pumps	III
Chemical mixing tank	II
Resin fill tank	III

1.11.3 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, systems required to be leaktight will remain leaktight, and components required to respond actively (such as valves and relays) will respond actively.

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 and condensate storage tank were designed in accordance with the stress limitations of American Water Works Association Standard D100.All components of the reactor coolant system and associated systems are designed to the standards of the applicable ASME code or USAS code. 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 1.11-2.

Table 1.11-2 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 1.11-2 have been modified in certain instances in accordance with NRC guidance given in References 3 and 4. Generic Letter 87-11 allows for the elimination of pipe whip restraints and jet impingement shields, which were installed to mitigate the effects of arbitrary intermediate pipe ruptures, provided certain criteria are met.

These design criteria have also been modified in certain instances by the application of "leak before break" technology, as discussed in Section 4.1.2.4.

To be able to perform their function, i.e., allow core shutdown and cooling the reactor vessel, internals must satisfy deformation limits, which are more restrictive than the stress limits shown in Table 1.11-2. For this reason the reactor vessel internals are treated separately.

1.11.3.1 <u>Piping, Vessels, and Supports</u>

The reasoning for selection of the load combinations and stress limits given in Table 1.11-2 is as follows. For the design earthquake, the nuclear steam supply system is designed to be capable of continued safe operation, i.e., for the combination of normal loads and design earthquake loading. Critical equipment and supports needed for this purpose are required to operate within normal design limits as shown in line 2 of Table 1.11-2.

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., shut the plant down and maintain it in a safe condition. This capability is ensured by maintaining the stress limits as shown in line 3 of Table 1.11-2. No rupture of a seismic Class I pipe can be caused by the occurrence of the maximum potential earthquake. With respect to the seismic design of the piping supports, relative displacement between anchor points has been considered in the seismic analysis of the main steam lines, where largest relative displacements are expected. Analysis indicates that the stress at the highest seismically stressed point is affected by less than 10-percent when relative anchor displacements are considered. The seismic supports installed have been verified to agree with the design location and therefore the locations used in the analyses.

Careful design and thorough quality control during manufacture and construction and periodic inspection during plant life, ensures that the independent occurrence of a reactor coolant pipe rupture is extremely remote. If it is assumed that a reactor coolant pipe ruptures, the stresses in the unbroken leg will be as noted in line 4 of Table 1.11-2.

1.11.3.2 <u>Reactor Vessel Internals</u>

1.11.3.2.1 Design Criteria for Normal Operation

The internals and core are designed for normal operating conditions and subjected to loads of mechanical, hydraulic, and thermal origin. The response of the structure under the design earthquake is included in this category.

The stress criteria established in Section III of the ASME Boiler and Pressure Vessel Code, Article 4, have been adopted as a guide for the design of the internals and core with exception of those fabrication techniques and materials, which are not covered by the code, such as the fuel rod cladding. Seismic stresses are combined in the most conservative way and are considered primary stresses.

The members are designed under the basic principles of: (1) maintaining distortions within acceptable limits, (2) keeping the stress levels within acceptable limits, and (3) prevention of fatigue failures.

1.11.3.2.2 Design Criteria for Abnormal Operation

The abnormal design condition assumes blowdown effects due to a reactor coolant pipe double-ended break.

For this condition the criteria for acceptability are that the reactor be capable of safe shutdown and that the engineered safety features are able to operate as designed. Consequently, the limitations established on the internals for these types of loads are concerned principally with

the maximum allowable deflections. The deflection limits for critical internals structures under abnormal operation are presented in Table 14.3-14.

1.11.3.3 Reactor Vessel

The criteria for movement of the reactor vessel, under the worst combination of loads, i.e., normal plus the maximum potential earthquake or normal plus reactor coolant pipe rupture loads, ensure that the radial movement of the reactor vessel will not exceed the clearance between the reactor coolant piping and the surrounding concrete.

The relative motions between reactor coolant system components are controlled by the structures, which are used to support the reactor vessel, the steam generators, the pressurizer and the reactor coolant pumps.

The maximum movement of the reactor vessel under the worst combination of loads, i.e., normal plus maximum potential earthquake or normal plus reactor coolant pipe rupture loads comprises an end deflection on the safety injection piping, which is small even in comparison with that resulting from thermal growth during plant heatup, and is well within the flexibility of the design of the piping system.

The supports are designed to limit the stresses in the pipe to the stress limits given in Table 1.11-2.

1.11.4 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 the reactor, recirculating pumps, and 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 1.11-2 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.

1.11.4.1 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.

1.11.4.1.1 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.

1.11.4.1.2 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.

1.11.4.2 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.

1.11.4.3 Diesel Generator Building

Due to the light weight of the structure, the wind load controlled the design.

1.11.4.4 Fan House

One hundred 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 with 5-percent critical damping was used for the concrete structure and the corresponding 2.5-percent was used for the steel superstructure. A one-third increase in allowable working stresses was allowed.

1.11.4.5 Boric Acid Evaporator Building

One hundred percent of the dead load was considered. For method of design, see fan house. Without allowing a one-third stress increase for seismic design, the controlling factor for reinforcing design was the minimum temperature steel requirements of the ACI-318 Building Code.

1.11.4.6 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.

1.11.4.7 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.

1.11.4.8 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 $\varnothing f_y$. The effects of water in the pool are accounted for in this design approach. Ground acceleration of 0.15g was used. In 1990, new high density spent fuel storage racks were installed. Prior to their installation, the spent fuel pit was reanalyzed (Ref. 6). The new racks were also analyzed (Ref. 5 and 6).

1.11.4.9 Electrical Penetration Tunnel

The peak of the response curve for 0.15g ground acceleration with 5-percent critical damping was considered using working stress design limits. The load was considered to act at 2/3 L, where L = the height of the tunnel. Temperature of steel considerations controlled the design of the concrete while service loads controlled the structural steel.

1.11.4.10 Pipe Penetration Tunnel

One hundred percent of the dead load, plus 50-percent of the live load were considered. The peak of the response curve for 0.15g ground acceleration with 5-percent damping was used to find the shear, which was considered as a concentrated load applied at the top slab of the tunnel. A one-third increase on working stress allowables was used in the design.

1.11.4.11 Electrical Cable Tunnel

One hundred percent of the dead load, 50-percent of the surcharge, and 50-percent of live load in the tunnel were considered. The Modified Rayleigh Method was used to determine the natural period and the loads were distributed as described in the Containment Design Report. The response curve for 0.15g ground acceleration with 5-percent critical damping was used. A one-third stress increase was permitted on working stress allowables when considering the effect of seismic loads.

1.11.4.12 Shield Wall

The peak of the response curve for 0.15g ground acceleration with 5-percent critical damping was used. The pipe break loads controlled the design.

1.11.4.13 Retaining Wall At Equipment Entrance

The wall was designed for soil pressure including a 1000 psf surcharge applied during reactor loading. The load combination that includes a seismic factor governs the design. It has been shown that the wall design was adequate.

1.11.4.14 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.

1.11.4.15 Condensate Water Storage Tank Foundation

The seismic loads on the spread footing foundation were those supplied by the tank manufacturer. The shear forces from the earthquake on the water in the tank were applied at 3/4 L above the footing, where L = the height of the tank. The shear force from the earthquake on the tank was applied at L/2 above the top of the footing. The stresses were limited to working stress design limits.

A multi degree-of-freedom modal analysis was performed on all Class I structures for Indian Point Unit 3. The results indicated that all structures except the containment structure were rigid. The only significant differences between the structural design of Units 2 and 3 seismic Class I buildings are the control building and the steel structural portion of the primary auxiliary building for Indian Point Unit 2, which are flexible steel structures. On Unit 3 they are rigid concrete structures. All seismic Class I structures on Indian Point Unit 2 except control building and containment shell are rigid and move with zero period ground acceleration. However, the design of all seismic Class I structures on Unit 2 were standardized and based on the peak acceleration of the ground response spectrum, which is extremely conservative for rigid structures.

In the preceding designs, limits have been placed on stresses to ensure that all structures will respond elastically to the earthquake. If for some reason inelastic response were to occur, the period of the structure would be expected to increase. Since the majority of the structures were designed for the peak of the response curve, the effect of any change in period would be to decrease the coefficient of spectral acceleration and thus lower all shears and moments.

Mathematical models were not used for seismic design of instrumentation. Ability to withstand the seismic condition is determined by actual vibration type testing of typical instrumentation equipment under simulated seismic accelerations to demonstrate its ability to perform its functions. The seismic testing is reported in Westinghouse report WCAP-7397-L (Reference 1).

The locations of protection and safeguards control and electrical equipment in Indian Point Unit 2 have been identified. The most adverse location, seismically, is the control building floor at

elevation 53-ft, which supports the nuclear instrumentation, radiation monitoring, process instrumentation, and safeguards logic racks. Dynamic analyses of this building for the plant design basis earthquake of 0.15g show that the significant horizontal and vertical accelerations of this floor are within the specified low seismic test envelopes given in WCAP-7397-L (Reference 1).

Seismic analysis of Class I equipment including heat exchangers, pumps, tanks, valves, motors, and electrical equipment components was performed using one of the following four methods:

1. Equipment, which is rigid and rigidly attached to its support structure, was analyzed for a "g" loading equal to the peak acceleration of the supporting structure at the appropriate elevation.

2. Equipment, which is not rigid and therefore a potential for response to the support motion exists, was analyzed for the peak of the floor response curve for appropriate damping values.

3. In some instances nonrigid equipment was analyzed using a multi degree-of-freedom modal analysis. All contributing modes were considered. In addition, a sufficient number of masses was included in the mathematical models to ensure that coupling effects of members within the component were properly considered. The results of these analyses indicated that the models contained more masses than necessary, and that future analyses of comparable equipment could be considerably simplified by considering fewer masses. The method of dynamic analysis used a proprietary computer code called WESTDYN. This code used as input the inertia values, member sectional properties, elastic characteristics, support and restraint data characteristics, and the appropriate seismic response spectrum. Both horizontal and vertical components of the seismic response spectrum were applied simultaneously. The modal participation factors were combined with the mode shapes and the appropriate seismic response spectra to obtain the structural response for each mode. The internal forces and moments were computed for each mode from which the modal stresses are determined. The stresses were then summed using the root mean square method.

4. Type testing of selected electrical equipment has been conducted to demonstrate seismic design adequacy as described in WCAP-7397-L (Reference 1).

For the analysis of equipment to resist the vertical seismic component, two-thirds of the horizontal response spectrum curves were used to determine the acceleration appropriate to the vertical frequency.

Engineered safeguards tanks, e.g., boric acid, accumulator and surge tanks, were analyzed using method 3 above for combined horizontal and vertical seismic excitation occurring simultaneously and in conjunction with normal loads without exceeding allowable stresses. Hydrodynamic analyses of these tanks have been performed using the methods described in Chapter 6 of the "U.S. Atomic Energy Commission - TID 7024." The stresses for these components due to the above-mentioned load combinations were found to be within allowable limits. Heat exchangers associated with the engineered safeguard systems, e.g., component cooling and residual heat removal, were analyzed using method 3 above, and the results show that stresses and deflections are within allowable limits.

Selected critical engineered safeguards valves were analyzed using method 3 above and the results indicated that their fundamental natural frequency was sufficiently separated from the building frequency. The results further indicated that the total stress, considering all modes, was far below the allowable stress limit.

Appendages, such as motors attached to motor-operated valves, were included in the mathematical models.

1.11.4.16 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. Besides analysis of the reactor coolant loop, portions of the following systems were analyzed:

   a. Safety injection.
   b. Main steam.
   c. Residual heat removal.
   d. Service water.
   e. Accumulator discharge.

f. Containment spray.
g. Component cooling.

1.11.4.16.1 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.

1.11.4.16.2 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 9.

1.11.4.17 Reactor Coolant System Analysis for Combination Loading of Design-Basis Earthquake and Design-Basis Accident [Historical Information Only]

The Indian Point Unit 2 reactor coolant system was not committed to be designed for the combination of the seismic and blowdown loads. However, an analysis for this combination of loadings was performed for the original configuration of the Indian Point Unit 3 reactor coolant system, which was identical to the original Unit 2 configuration.

The analysis was performed as outlined below:

1. A lumped mass dynamic mathematical model of the primary coolant loop and support system was developed.

2. This dynamic model was subjected to multiple simultaneous time history hydraulic forcing functions for the blowdown analysis. The double-ended ruptures were located at places of large change in flexibility. Time history response of the total structure to these conditions was computed and reduced to time history stresses.

3. The dynamic model was then subjected to a floor response spectra earthquake analysis.

4. The loads as determined above were used for an evaluation of the stresses along the piping system.

5. The stresses as determined from the basis described above were lower than the allowable stresses calculated by using the approach described in WCAP-5890, Revision 1 (Reference 2) and the following parameters:

    a. 20-percent of the uniform strain on the allowable membrane and average strain.

    b. 23,100 psi as the at-temperature yield in the axial direction. This value was based on the minimum value of the at-temperature yield in the loop direction as measured with samples from the Unit 2 piping, and increased by 10-percent for the increase in strength in going from the loop to the axial direction. The tensile tests on the Unit 2 piping material at-temperature yielded at a minimum value of 20,900 psi, a maximum of 29,700 psi, and an average of eleven samples of 23,300 psi.

Based on the above analysis, it was concluded that the Unit 2 reactor coolant system can stand the combination of blowdown and seismic loads within acceptable stress limits.

In 1989, the NRC approved changes to the design bases with respect to dynamic effects of postulated primary loop pipe ruptures, as discussed in Section 4.1.2.4. In 2000, an analysis of the Unit 2 reactor coolant loop and its component supports, which incorporates the NRC approved changes, was performed with the replacement steam generators and sixteen of the original twenty-four steam generator support frame hydraulic snubbers removed (Reference 11). In line with the older analysis for Unit 3 described above, the Unit 2 analysis of 2000 included the effects of the now controlling feedwater line break at the steam generator nozzle in a similar fashion as described for the Unit 3 analysis. Based on this revised analysis, it was concluded that the Unit 2 reactor coolant system can withstand the combination of blowdown and seismic loads within acceptable stress limits.

This 2000 analysis has since been updated to include a power uprate to a core power level of 3216 MWt. Combination of blowdown and seismic loads were not considered in this latest evaluation.

1.11.4.18 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.

1.11.4.19 Seismic Evaluation of the Fan Cooler and Passive Hydrogen Recombiner Systems

The seismic analysis of the fan cooler system was completed in two parts.

1. Analysis of the structural steel enclosure of the fan cooler units to include the effect of supported equipment.

   The structural analysis considering simultaneous incident pressures and earthquake forces was conducted on particular members, plates, and connections, which are, considered critical to the structural performance of the reactor containment fan coolers. This analysis included consideration of the mass of all components supported partially or wholly by the enclosure. The fan, fan motor, and fan motor heat exchanger, although entirely within the enclosure, are independently supported from the concrete floor that makes up the base of each unit.

Earthquake loadings were treated using the response spectra techniques. A horizontal force of 0.6W and a vertical force of 0.4W, where W is the weight of the member including static forces, were concentrated at the center of gravity of each member. A negative differential pressure of 1.5 psig was applied to the portion of the unit from the inlet up through the fan compartment; a negative differential pressure of 6.3 psig was applied to the charcoal filter compartment. Both of these values are consistent with unit geometry and containment environment following a loss of coolant accident or main steam line break. The charcoal filter compartment pressure is limited by a pressure equalization device installed during the 1997/1998 Maintenance Outage. All loadings were assumed to act simultaneously and comparison was made with allowable stresses consistent with specifications for installed materials. An increase in allowables was considered for loads associated with accident conditions. Where applicable, allowable concrete stresses were taken from ACI Specifications.

Results of the analysis on the enclosure demonstrated that the design is adequate.

2. Evaluation of the fan motor system and its foundation.

The fan motor and its supporting structural system was evaluated using acceleration values for a maximum hypothetical earthquake. These values are 0.6g for the horizontal direction and 0.4g for the vertical direction. These accelerations were assumed to act simultaneously.

The failure modes considered for the motor unit were excess deflection of the rotor shaft, which results in rubbing against the housing or by bearing failure. The failure modes for the fan are failure of the fan shaft support bearings or deflection of the fan housing and fan wheel causing binding. In addition, the potential for shear and overturning failure of the motor fan assembly at the foundation anchorage was evaluated.

Based on analyses made on similar fan motor systems, it was concluded the fan cooler units in the containment are adequately designed to resist the seismic loading defined for the site and supporting building structure.

The two hydrogen recombiners are located in the containment at the 95-ft elevation. The hydrogen recombiners are as shown on Figure 6.8-1. The Passive Hydrogen Recombiners (PHRs) are seismic class I, and have undergone qualification testing in accordance with IEEE 344-1987. On April 14, 2005, NRC issued IP2 License Amendment 243 which eliminated the requirement for hydrogen recombiners to provide any combustible gas control function.

1.11.4.20 <u>Masonry Walls</u>

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 10. As a result of this evaluation, certain walls in the control building,

the Unit No. 1 Superheater building, the boric acid evaporator building, the fan house, and the fuel storage building have been reinforced.

1.11.5 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 does not necessarily include associated equipment.

1.11.6 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 I structures and components that could be endangered by failure of Class III structures are the control building, main steam piping, and feedwater piping, which could be endangered by failure of the Class III turbine building. No special provisions were provided in the original plant design except in the case of the main steam and feedwater lines up to the isolation valves, which are protected by the shield wall and the structural frame at the north end of the shield wall. Evaluations were performed and bracing was added to the turbine building during construction, as described in section 1.11.6.5, to preclude such catastrophic failures.

The only Class III crane whose failure could endanger any Class I function is the fuel storage building crane. Failure of this crane will not impair a safe and orderly shutdown. 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 Class III manipulator crane in the containment building is restrained from overturning and will not endanger Class I structures.

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.

1.11.6.1 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 1.11-2.

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 1.11-4.

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.

1.11.6.2 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 1.11-2.
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 1.11-3 and 1.11-4. 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.

1.11.6.3 Seismic and Wind Analysis of the Superheater Stack of Indian Point Unit 1

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.

1.11.6.3.1 Load Case 1 - Tornado

I. Load Criteria

Wind = 300 mph
L = D + W'

where:

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

II. Method of Load Analysis

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

III. Allowable Stress Criteria

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

where:

$\sigma_a =$ allowable stress (psi)

E = modulus of elasticity (psi)

t = shell thickness (in.)

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})

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

1.11.6.3.2 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 1.11-2.

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

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 1.11-5. 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

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

σ= 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

1.11.6.3.3 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.

1.11.6.4 Seismic and Tornado Evaluation of the Superheater Building at Indian Point Unit 1

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 1.11-2 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:

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

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

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

Where:

n = Refers to mode n

r = Refers to mass r

$\varnothing'_{rn}$ = Component of $\varnothing_{rn}$ in the direction of the earthquake

$\varnothing_{rn}$ = 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 1.11-5 and 1.11-6. A typical column line modeled for STRUDL to determine overall column line stiffness and permit determination of resultant seismic stresses is shown in Figure 1.11-7. In Figure 1.11-8 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.

1.11.6.5 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 1.11-5, 1.11-6, and 1.11-7 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.

1.11.7 Seismic Qualification For Safe Shutdown

In response to NRC Generic Letter (GL) 87-02 and Supplement No. 1 to GL 87-02, “Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue A-46,” Con Edison committed to implement Generic Implementation Procedure (GIP-2) including the clarifications, interpretations, and exceptions in the NRC’s Supplemental Safety Evaluation Report (SSER-2). The NRC accepted Consolidated Edison’s response and commitments regarding this issue as documented in their Safety Evaluation Report (SER) dated November 19, 1992. Consolidated Edison has verified the seismic capabilities of equipment required for safe shutdown, as documented in Reference 7. The verification utilized the Generic Implementation Procedure developed by the Seismic Qualification Utility Group and approved by the NRC. A Summary Report, including the Seismic Evaluation and Relay Evaluation Reports, was submitted to the NRC on December 31, 1996[7]. Indian Point 2 site specific SER, dated November 8, 2000, Reference 15, provides NRC conclusions that the licensee may revise its licensing basis by incorporating the SQUG methodology.

Revision 3 of the Generic Implementation Procedure (GIP-3), Reference 12, as modified and supplemented by the NRC’s Supplemental Safety Evaluation Report No.2 (SSER-2), Reference 13, and No.3 (SSER-3), Reference 14, may be used as an alternative to existing methods for the seismic design and verification of modified, new, and replacement equipment. Only those portions of GIP-3 as described in Section 5 of “Implementation Guidelines for Seismic

Qualification of New and Replacement Equipment/Parts (NARE) using the Generic Implementation Procedure", Reference 16, shall apply to the seismic design and verification of mechanical and electrical equipment, electrical relays, tanks and heat exchangers, and cable and conduit raceway systems.

1.11.8 Protection from Flooding of Equipment Important to Safety

In response to NRC Guidelines for Protection from Flooding of Equipment Important to Safety, Consolidated Edison identified the potential sources of flooding outside containment that could affect safety-related equipment. The areas containing safety-related equipment that could be subject to flooding from postulated failure of water systems that are not seismic Class I were evaluated. The plant is designed so as to minimize or eliminate the vulnerability of safety-related equipment to this flooding. Modifications were made to install water level alarm switches in the adjoining Unit 1 condenser pit area, and add flap panels in doors from the primary auxiliary building and the auxiliary feed pump room. A later modification was made to install a flood control drain line and valve from the PAB to a manhole in the transformer yard. These modifications along with the implementation of an alternate safe shutdown capability, as discussed in Section 8.3, serve to mitigate the consequences of the postulated flooding. Additionally, operator action would be taken in the event of flooding from the circulating water system to prevent damage to the 480 volt switchgear in the control building.

In their Safety Evaluation Report (SER) dated December 18, 1980, the NRC determined that design features and operating procedures provide assurance that the plant can be safely shut down in the event of flooding outside containment from a non-seismic component or pipe and that their guidelines (contained in Appendix A to the SER) have been satisfied.[8] The Fire Protection piping added later in the PAB was analyzed for seismic loads and restrained to prevent breakage.

REFERENCES FOR SECTION 1.11

1. E. L. Vogeding, Topical Report - Seismic Testing of Electrical and Control Equipment, WCAP-7397-L, Westinghouse Electric Corporation, January 1970.

2. Westinghouse Electric Corporation, Ultimate Strength Criteria to Ensure No Loss of Function of Piping and Vessels Under Earthquake Loading, WCAP-5890, Revision 1.

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

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

5. 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.

6. 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.

7. Letter from Quinn, Con Edison, to NRC, Subject: Summary Report for Resolution of USI-A-46, Seismic Qualification, dated December 31, 1996.

8. Letter from S. A. Varga, NRC, to J. D. O'Toole, Con Edison, Subject: Safety Evaluation Report Susceptibility of Safety-Related Systems to Flooding from Failure of Non-Category I Systems, dated December 18, 1980.

9. 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.

10. 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.

11. Altran Corporation, Technical Report No. 00222-TR-001, Rev. 1, Reactor Coolant Loop Analysis for Replacement Steam Generators and Snubber Reduction

12. Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment, Revision 3, Updated 05/16/97 (GIP-3). Prepared by SQUG and sent to the NRC by letter dated May 16,1997.

13. Supplement No.1 to Generic Letter (GL) 87-02 that transmits Supplemental Safety Evaluation Report No.2 (SSER-2) on SQUG Generic Implementation Procedure Revision 2, as Corrected on February 14, 1992 (GIP-2), May 22, 1992.

14. NRC letter to SQUG dated December 4, 1997, Supplemental Safety Evaluation Report No.3 (SSER-3), on the Review of Revision 3 to the Generic Implementation Procedure for Seismic Verification of Nuclear Power Plant Equipment, Updated 05/16/97 (GIP-3)

15. Indian Point Nuclear Generating Station No. 2 – "Plant Specific Safety Evaluation Report for Unresolved Safety Issue A-46 Program Implementation", November 8, 2000.

16. "Implementation Guidelines for Seismic Qualification of New and Replacement Equipment/Parts (NARE) Using the Generic Implementation Procedure (GIP)", Revision 4, by MPR Associates

TABLE 1.11-1
Damping Factors

COMPONENT	PERCENT OF CRITICAL DAMPING
Containment structure	2.0
Concrete support structure of reactor vessel	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 1.11-2
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	$P_m \leq S_m$ $P_L + P_B \leq 1.5\ S_m$	$P_m \leq 1.2S$ $P_L + P_B \leq 1.2S$	1-1/3 working stresses or applicable factored load design values
3. Normal + maximum potential earthquake loads	$P_m \leq 1.2S_m$ $P_L + P_B \leq 1.2\left(1.5S_m\right)$	$P_m \leq 1.2S$ $P_L + P_B \leq 1.2\left(1.5S\right)$	Deflections and stresses of supports limited to maintain supported equipment within their stress limits
4. Normal + pipe rupture loads	$P_m \leq 1.2S_m$ $P_L + P_B \leq 1.2\left(1.5S_m\right)$	$P_m \leq 1.2S$ $P_L + P_B \leq 1.2\left(1.5S\right)$	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

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

TABLE 1.11-3
DELETED

TABLE 1.11-4
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 1.11-5
Relative Stiffness Percentages

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

TABLE 1.11-6
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 1.11-7
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

1.11 FIGURES

Figure No.	Title
Figure 1.11-1	Ten Percent of Gravity Response Spectra
Figure 1.11-2	Fifteen Percent of Gravity Response Spectra
Figure 1.11-3	Fuel Storage Building North-South Model [Historical]
Figure 1.11-4	Fuel Storage Building East-West Model [Historical]
Figure 1.11-5	Indian Point Unit 1 Superheater Building North-South Section
Figure 1.11-6	Indian Point Unit 1 Superheater Building East-West Section
Figure 1.11-7	Column Line “G”
Figure 1.11-8	Representation of Lumped Mass Model of Superheater Building Used in Dynamic Analysis

1.12 INSERVICE INSPECTION AND TESTING PROGRAMS

1.12.1 General

The ISI Program complies with the requirements of 10 CFR 50.55a and is based upon the requirements set forth in ASME Boiler and Pressure Vessel Code, Section XI and by the applicable Code year. This program is also responsive to pertinent provisions of applicable Regulatory Guides.

The Indian Point Unit 2 Inservice Inspection (ISI) and Testing (IST) Programs for the ten year interval are controlled by Program Plans and plant procedures.

1.12.2 Application

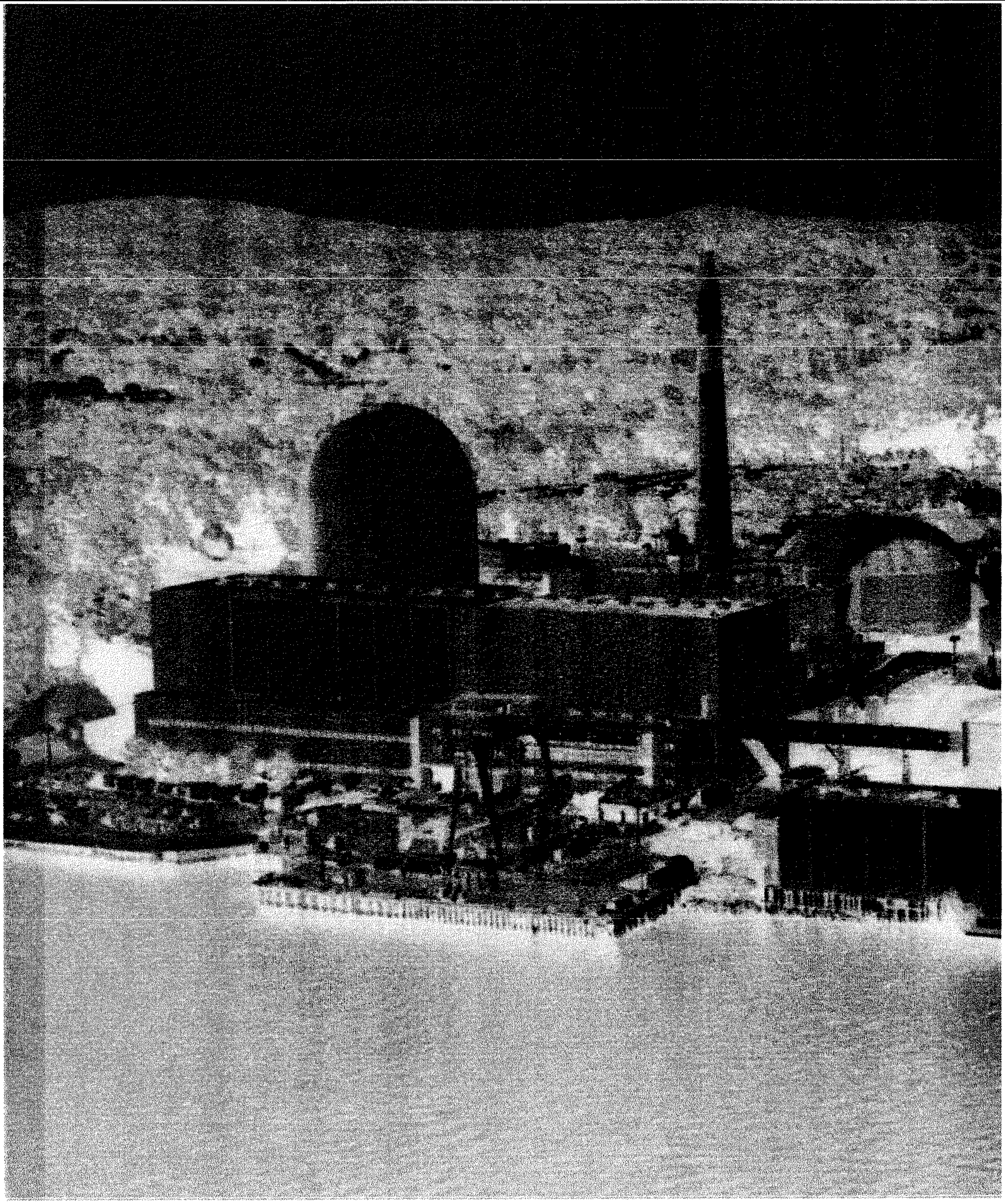
The ISI program applies to Quality Groups A, B, and C systems, components (including supports), and pumps and valves as classified in accordance with Regulatory Guide 1.26, Revision 3.

1.12.3 Program Summary

The ISI and IST programs identify the specific systems, components, or parts thereof to be examined and the specific pumps and valves to be tested.

1.13 CONTROL OF HEAVY LOADS

In response to a December 22, 1980 Generic Letter and to NRC Staff guidelines provided in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," Con Edison performed evaluations of provisions for the handling and control of heavy loads in the vicinity of irradiated fuel or safe shutdown equipment. Control of heavy loads in the Fuel Storage Building is addressed in section 9.5.6. The NRC documented their acceptance of Con Edison's assessments in a Safety Evaluation Report dated February 19, 1985.

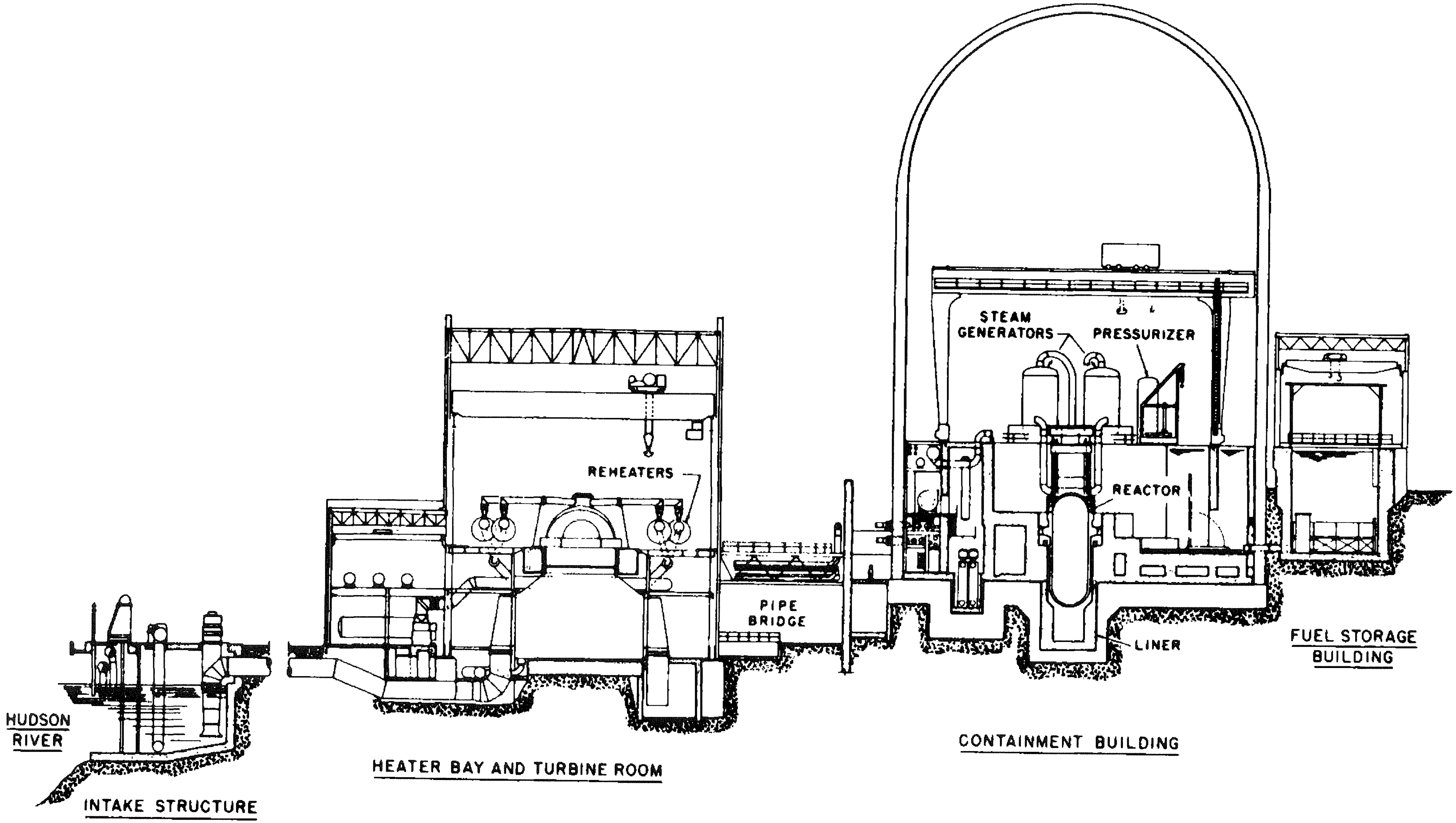


INDIAN POINT UNIT No. 2

UFSAR FIGURE 1.2-1

INDIAN POINT
NUCLEAR GENERATING STATION
UNITS 1 & 2

MIC. No. 1999MC3559	REV. No. 17A

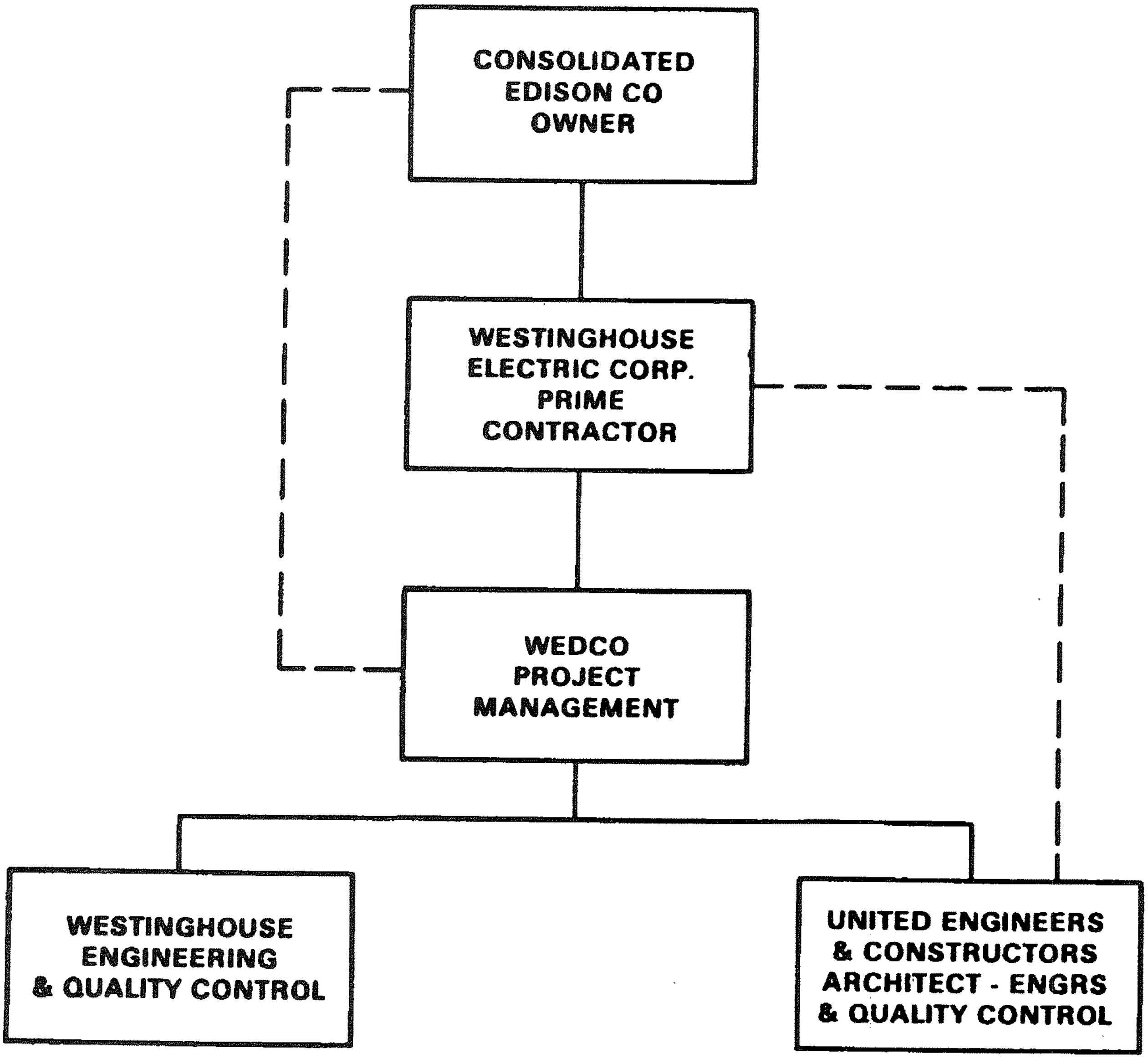


INDIAN POINT UNIT No. 2

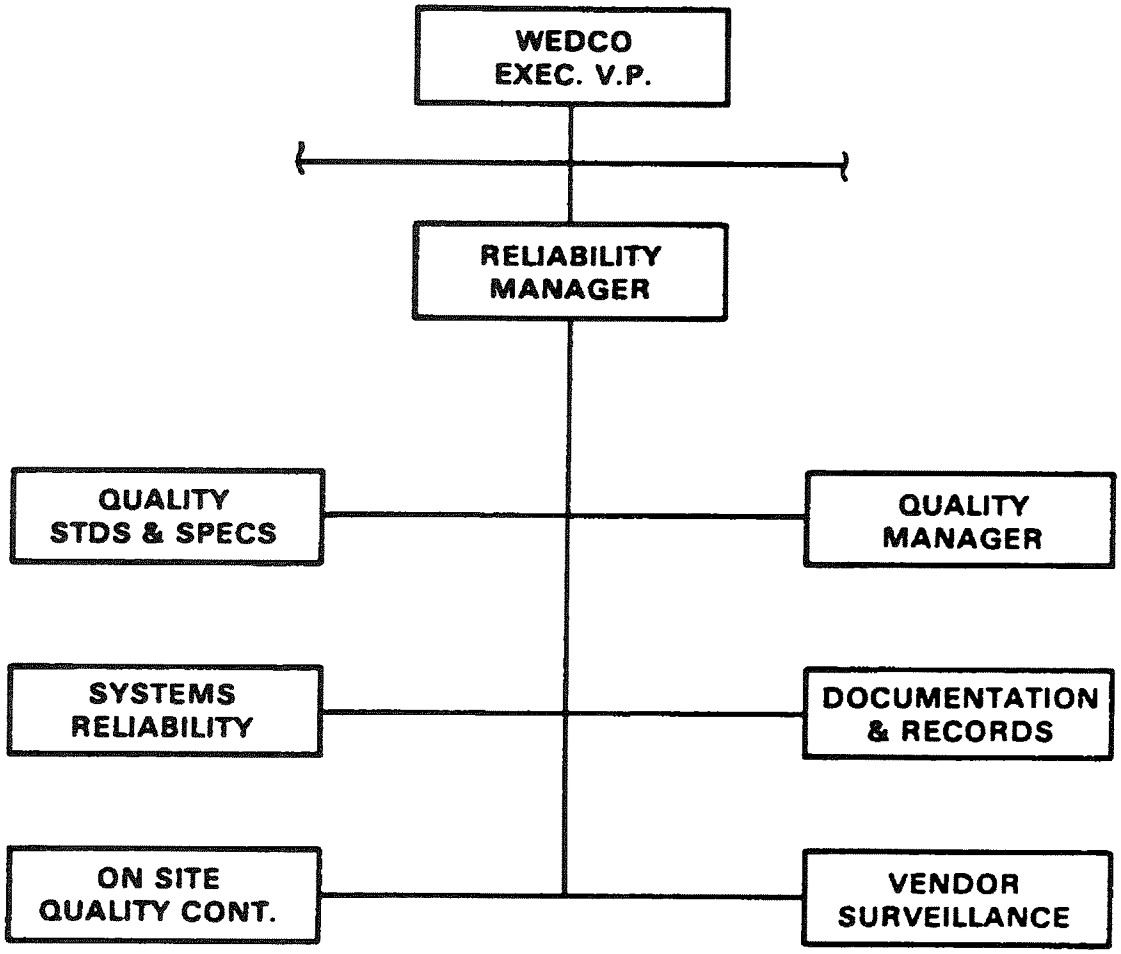
UFSAR FIGURE 1.2-4

CROSS SECTION OF PLANT

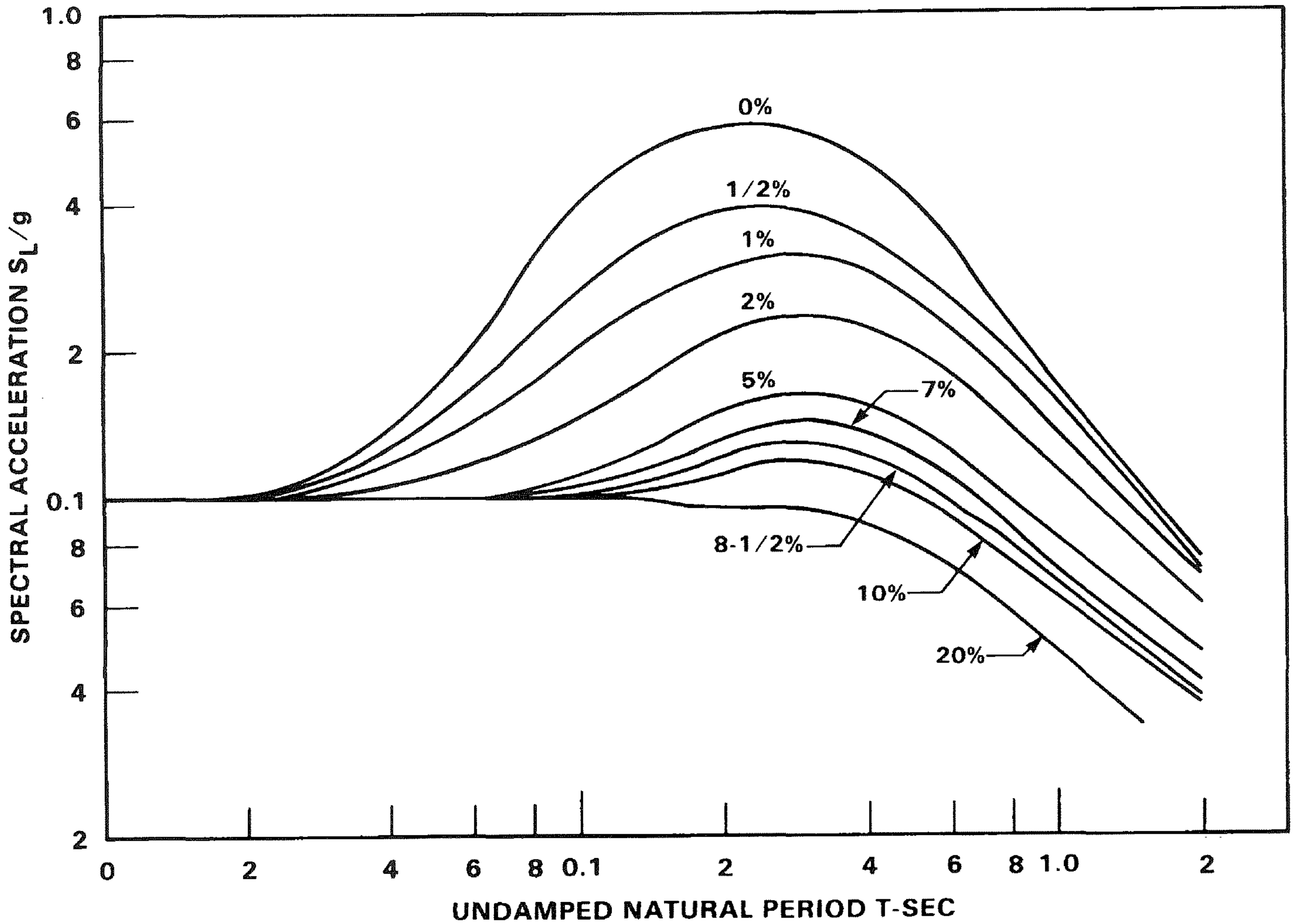
MIC. No. 1999MC3561 | REV. No. 17A



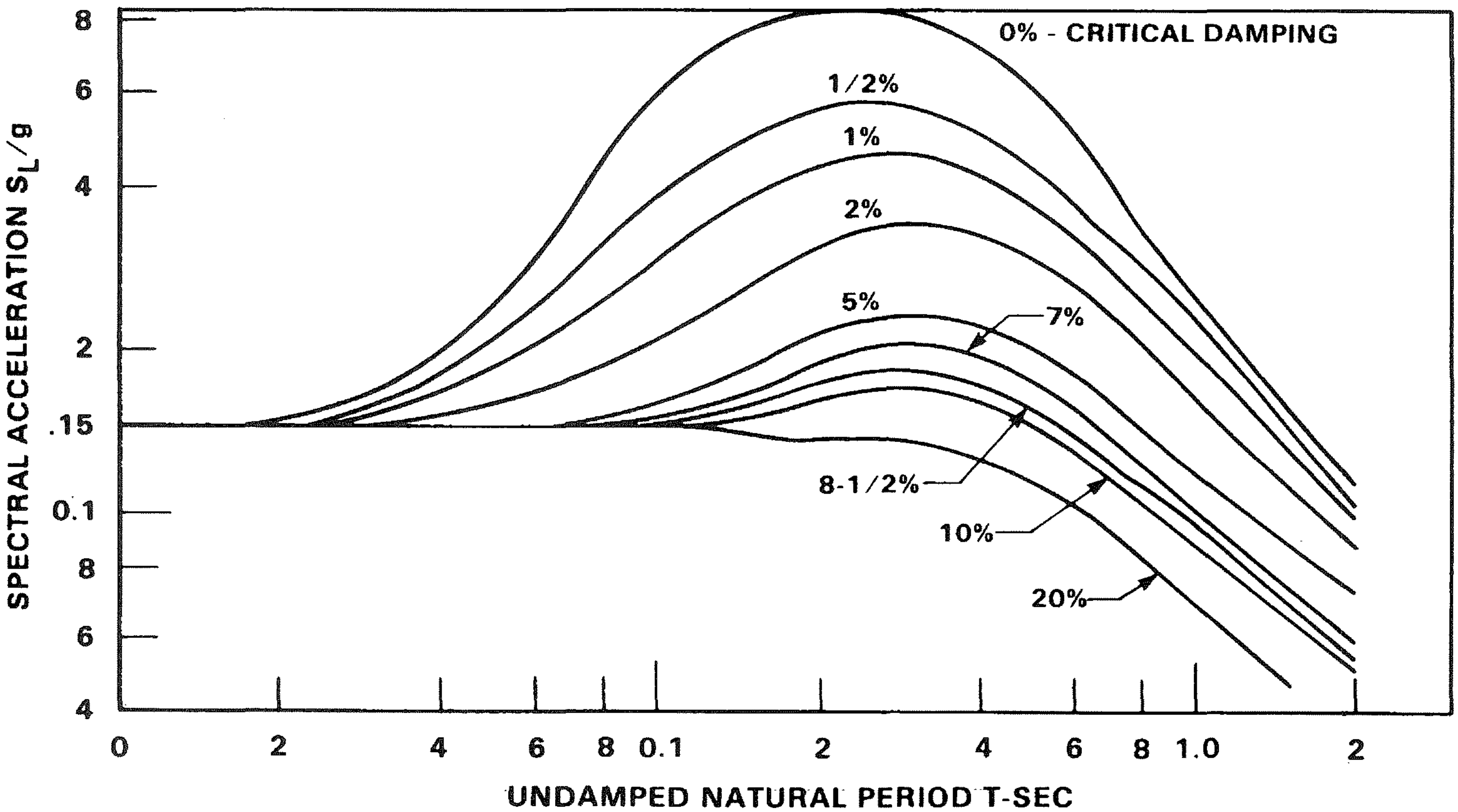
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 1.7–1	
FUNCTIONAL RELATIONSHIPS	
MIC. No. 1999MC3570	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 1.8-1 ORGANIZATIONAL CHART WEDCO RELIABILITY GROUP	
MIC. No. 1999MC3571	REV. No. 17A



INDIAN POINT UNIT No. 2
UFSAR FIGURE 1.11-1 TEN PERCENT OF GRAVITY RESPONSE SPECTRA
MIC. No. 1999MC3562 | REV. No. 17A

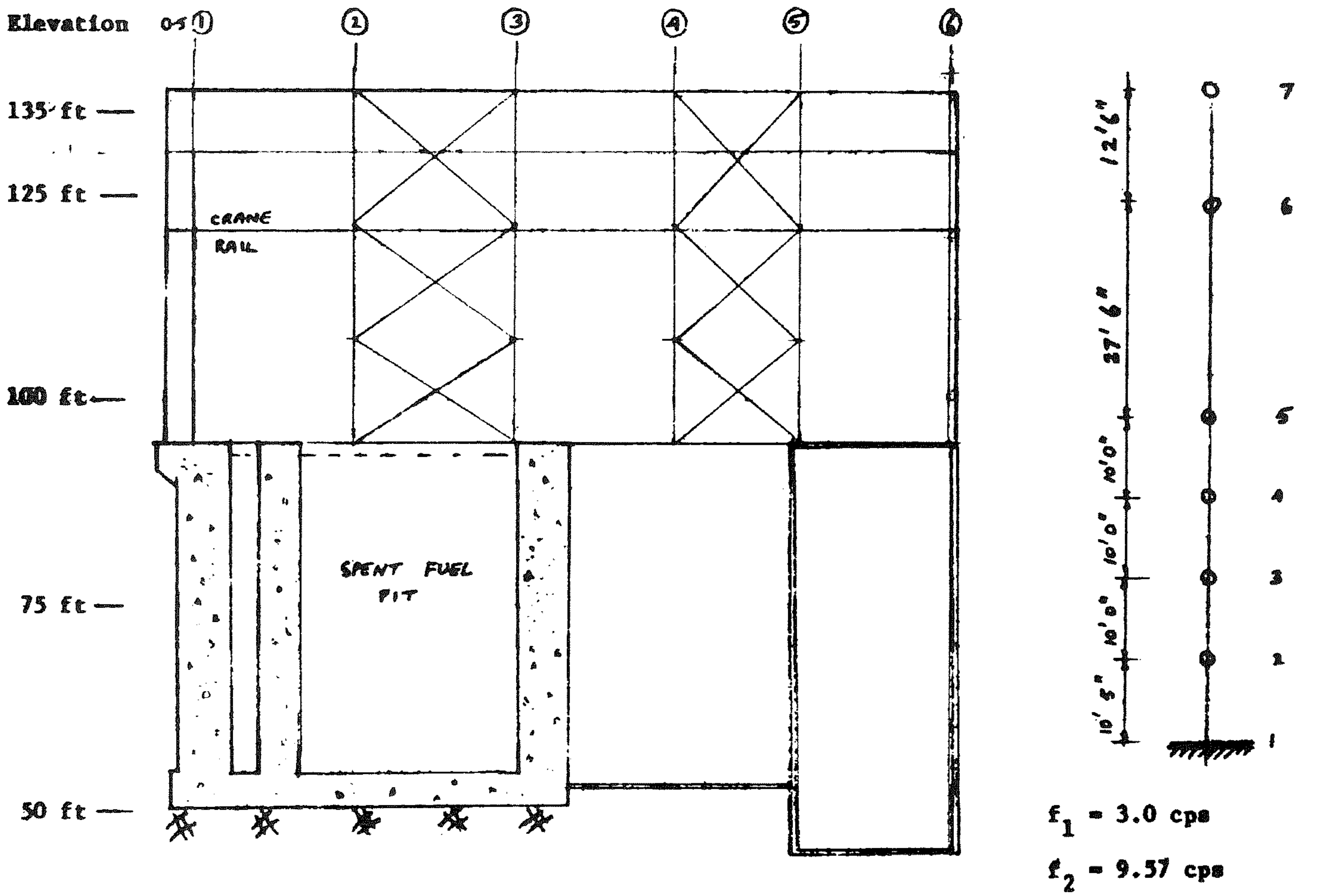


INDIAN POINT UNIT No. 2

UFSAR FIGURE 1.11-2

FIFTEEN PERCENT OF GRAVITY RESPONSE SPECTRA

MIC. No. 1999MC3563	REV. No. 17A

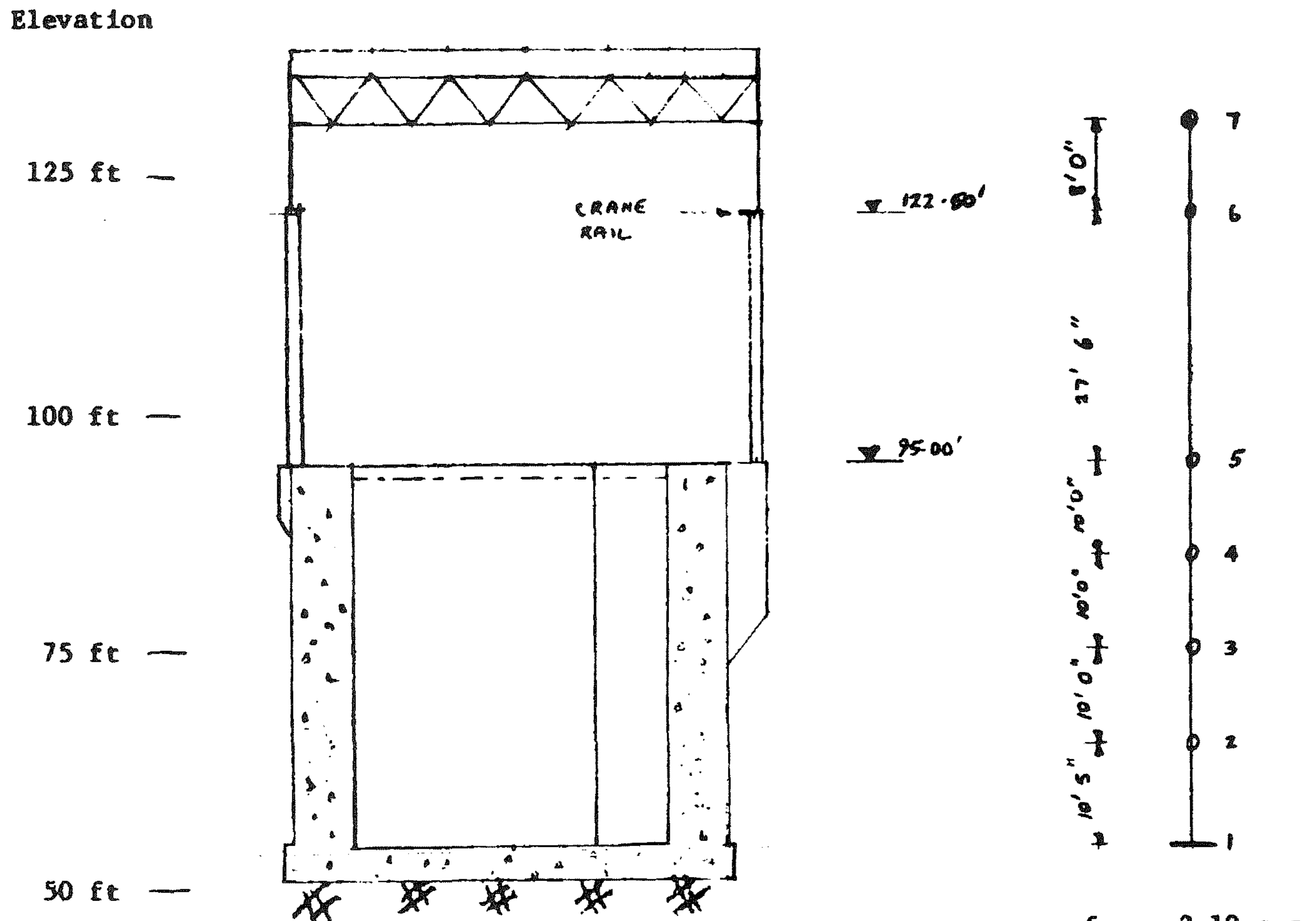


INDIAN POINT UNIT No. 2

UFSAR FIGURE 1.11-3

FUEL STORAGE BUILDING NORTH-SOUTH MODEL

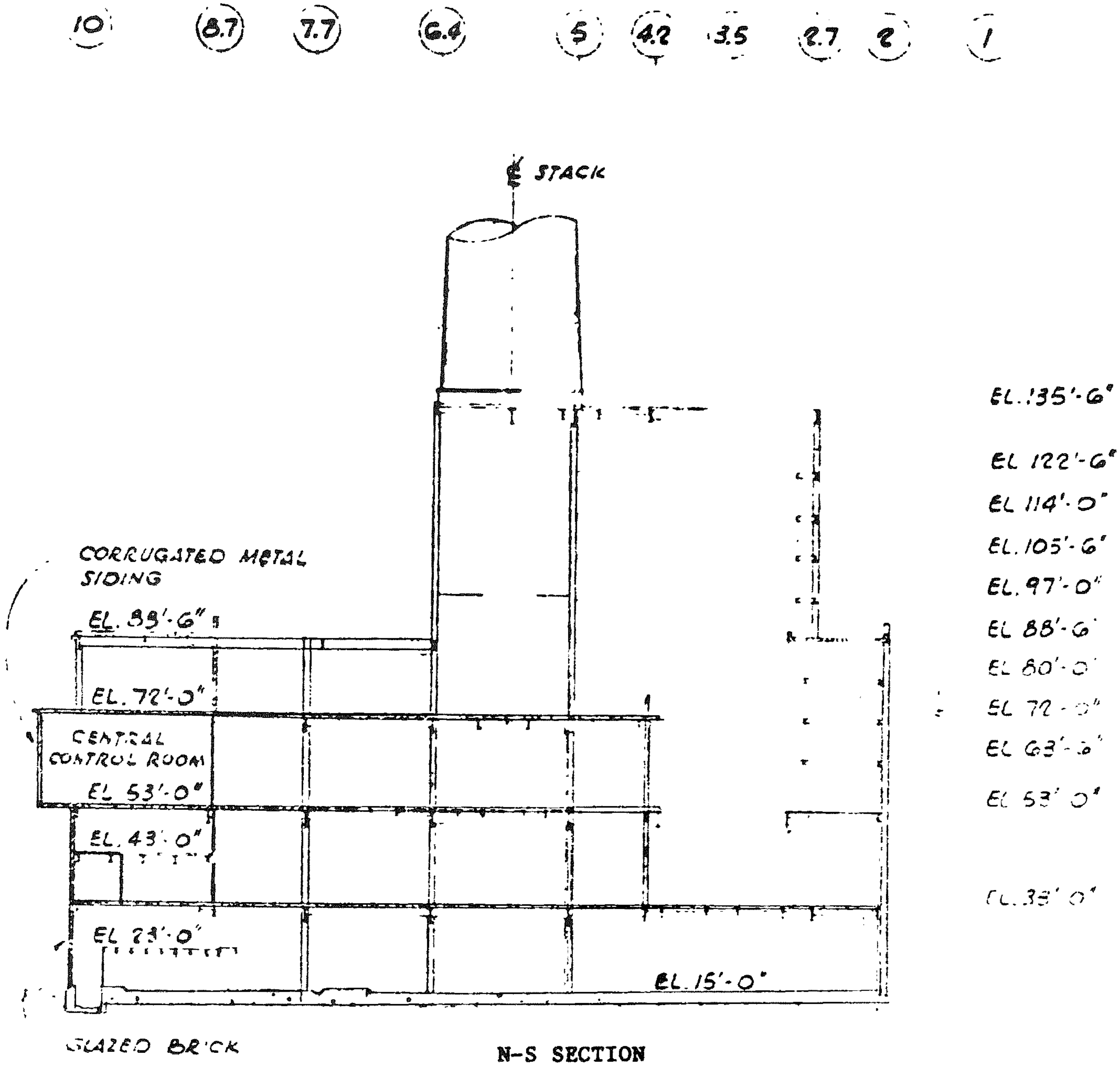
MIC. No. 1999MC3564 | REV. No. 17A



f_1 = 2.10 cps

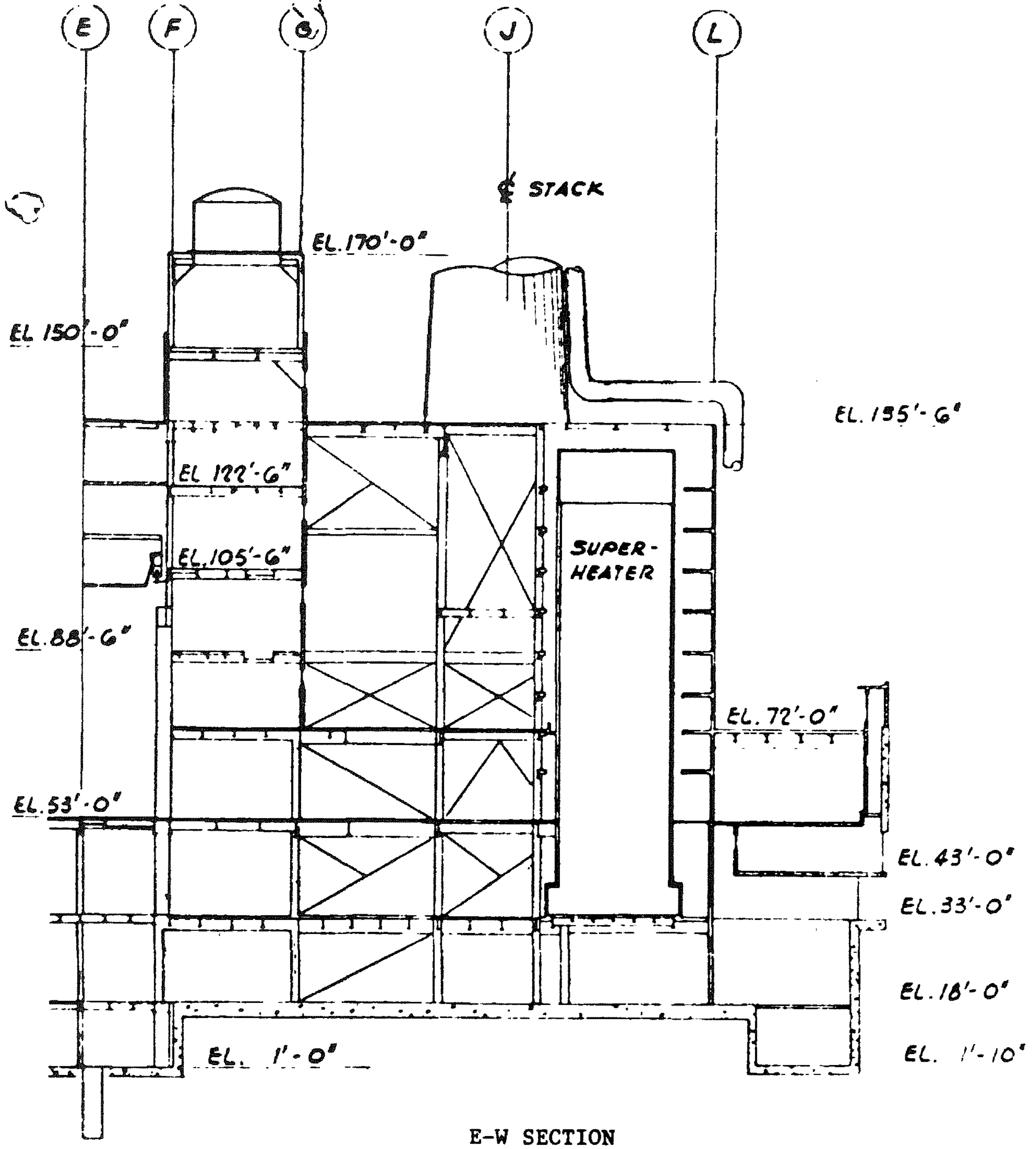
f_2 = 12.97 cps

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 1.11-4 FUEL STORAGE BUILDING EAST-WEST MODEL	
MIC. No. 1999MC3565	REV. No. 17A



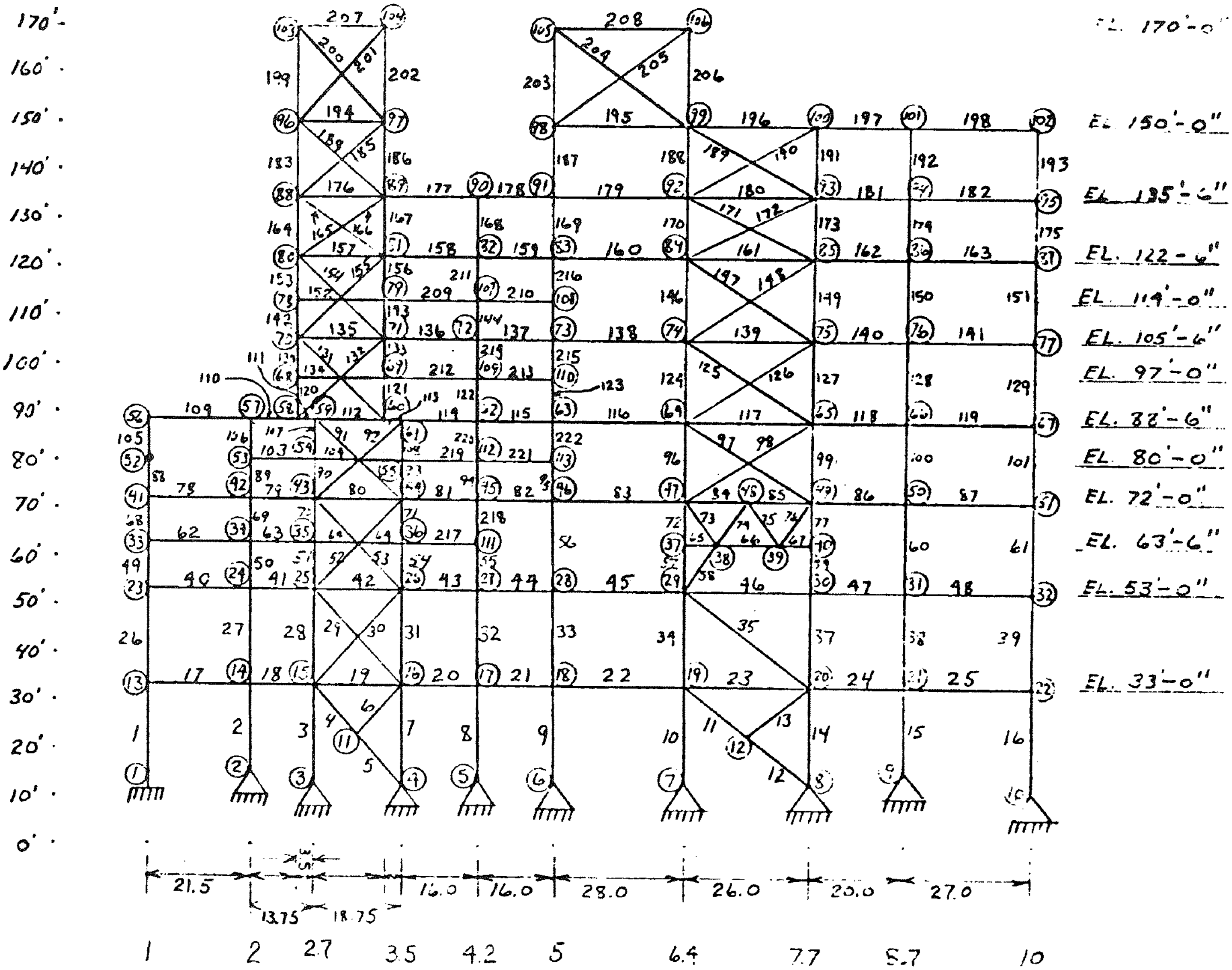
N-S SECTION

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 1.11–5	
INDIAN POINT UNIT 1 SUPERHEATER BUILDING NORTH–SOUTH SECTION	
MIC. No. 1999MC3566	REV. No. 17A



E-W SECTION

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 1.11-6 INDIAN POINT UNIT 1 SUPERHEATER BUILDING EAST-WEST SECTION	
MIC. No. 1999MC3567	REV. No. 17A

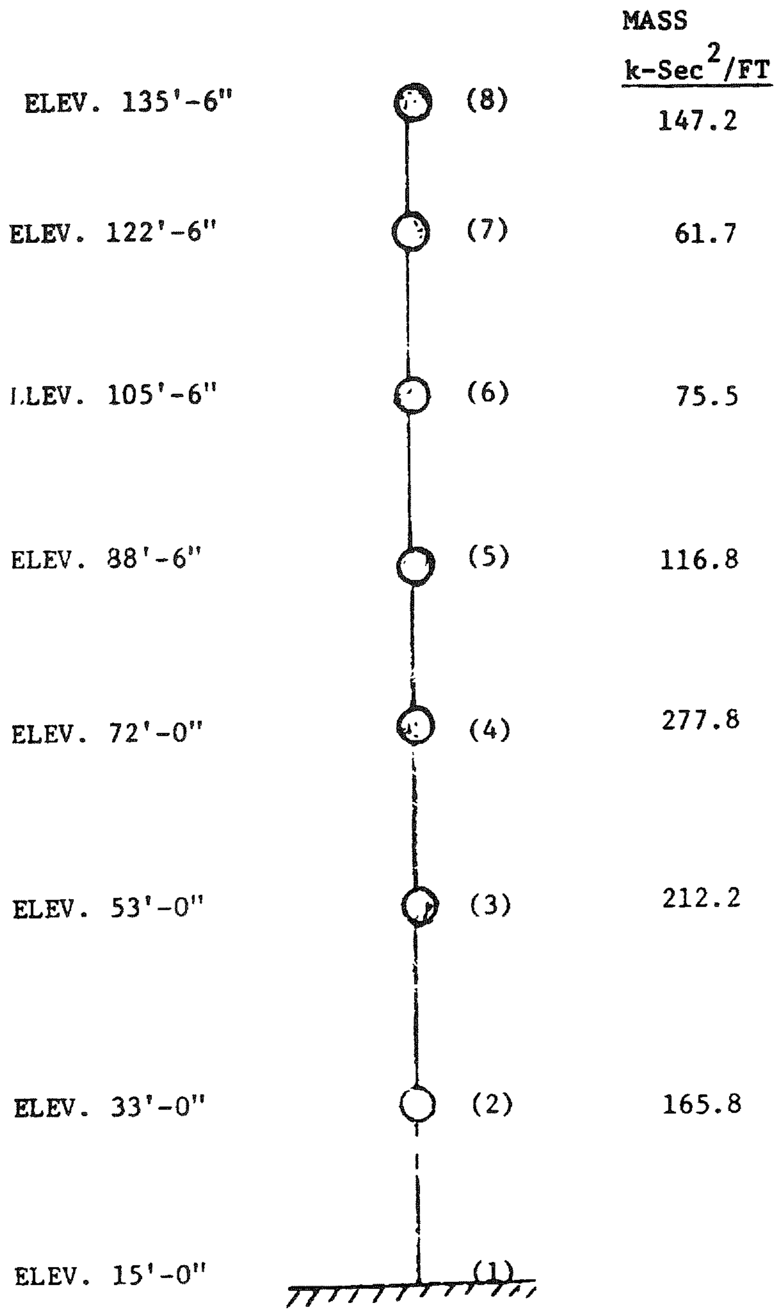


INDIAN POINT UNIT No. 2

UFSAR FIGURE 1.11-7

COLUMN LINE "G"

MIC. No. 1999MC3568	REV. No. 17A



NOTE: Stack Lumped at Mass Point

Mode	CPS N-S Freq.	CPS E-W Freq.
1	0.72	0.95
2	1.58	2.07
3	3.48	4.07
4	4.65	5.18
5	6.0	7.0
6	7.15	8.0
7	8.25	9.7

NOTE: STIFFNESS MATRICES USED TO DEFINE THE STIFFNESS RELATIONSHIP BETWEEN MASS POINTS IN THE E-W AND N-S DIRECTION

INDIAN POINT UNIT No. 2

UFSAR FIGURE 1.11-8

REPRESENTATION OF LUMPED MASS MODEL OF SUPERHEATER BUILDING USED IN DYNAMIC ANALYSIS

MIC. No. 1999MC3569 | REV. No. 17A

CHAPTER 2
SITE AND ENVIRONMENT

2.1 SUMMARY AND CONCLUSIONS

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

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

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

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

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

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

Geologically, the site consists of a hard limestone in a jointed condition that provides a solid bed for the plant foundation. The bedrock is sufficiently sound to support any loads that could be expected up to 50 tons/ft^2, which is far in excess of any load that may be imposed by the plant. Although it is hard, the jointed limestone formation is permeable to water. Thus, if water from the plant should enter the ground (an improbable event since the plant is designed to preclude any leakage into 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 plant 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 plant design is to treat the river water as if it were used for drinking and thus to reduce radioactive discharge, by dilution with ordinary plant effluent, to concentrations that would be tolerable for drinking water. There is a very low probability of flooding at the site.

[Historical Information] Seismic activity in the Indian Point area is limited to low-level microseismicity. Detailed field investigations[1-3] 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 sediment overlying fault traces or in cores obtained in the vicinity of Indian Point that might support a conclusion that displacement has occurred along major fault systems within the New York Highlands, the Ramapo, or its associated branches during Quaternary time (the last 1.5 million years). In the vicinity of Indian Point, evidence that no displacement has occurred in the last 65 million years (since the Mesozoic) along specific major structures has been observed.

The plant 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 plant's design-basis earthquake.

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

These data have been used in evaluating the effects of gaseous discharges from the plant during normal operations and during the postulated loss-of-coolant accident. The evaluations indicate that the site meteorology provides adequate diffusion and dilution of any released gases.

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

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

REFERENCES FOR SECTION 2.1

1. Ratcliffe 1976.
2. Ratcliffe 1980.
3. Dames & Moore 1977.

4. Hollis S. Ingraham, Consolidated Edison Indian Point Reactor Environmental and Post Operational Survey - August, 1965, Division of Environmental Health Services, New York State Department of Health.
5. Hollis S. Ingraham, Consolidated Edison Indian Point Reactor Environmental and Post Operational Survey - July, 1966, Division of Environmental Health Services, New York State Department of Health.

2.2 LOCATION

2.2.1 General

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

2.2.2 Access

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

2.2.3 Site Ownership And Control

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

The 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 Entergy to determine all activities within the right-of-way in order to ensure safe operation of the units.

Units 1, 2, and 3 have a security fence surrounding the "protected" areas. Access to the protected areas is controlled via security buildings that are manned on a 24-hr basis. In addition, spaces within the protected area designated as "vital areas" are provided with additional access control. All roads within the site are continuously patrolled by security personnel. A site plot plan is shown in Historical Figure 2.2-2.

2.2.4 Activities On The Site

The principal activities on the site are the generation, transmission, and distribution of electrical energy; associated service activities; activities relating to the controlled conversion of the nuclear energy of fuel to heat energy by the process of nuclear fission; and the storage, use, and production of special nuclear source and byproduct materials.

2.2 FIGURES

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

2.3 TOPOGRAPHY

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

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

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

2.3 FIGURES

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

2.4 POPULATION AND LAND USE

2.4.1 Overview

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

2.4.2 Population And Land Use

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

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 plant on the west side of the River.

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

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

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

2.4.3 Low-Population Zone

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

CFR 50.67. The low-population zone population in the year 2010 is projected to be approximately 88.

2.4.4 Exclusion Area

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

2.4.5 Population Data Sources

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

1. 1978 Official Population Projections for New York State Counties, prepared by the Economic Development Board, New York State Department of Commerce.

2. Population by Municipality 1970-2000, prepared by the Westchester County Department of Planning, October 1979.

3. Population of Rockland County, Capacity and Forecast, 1970-2000, prepared by the Rockland Planning Board, April 1978.

4. Population Estimate and Projections, Orange County, New York, prepared by the Orange County Planning Department, March 1980.

5. Putnam County Population Projections, prepared by the Putnam County Planning Board, 1977.

6. New Jersey Revised Total and Interim Age and Sex Population Projections, 1980-2000, prepared by the New Jersey Department of Labor and Industry, Division of Planning and Research, Office of Demographic and Economic Analysis, April 1979.

7. State of Connecticut Population Projections for Connecticut Municipalities and Regions to the Year 2000, prepared by the Office of Policy and Management, Comprehensive Planning Division, February 1980.

8. Pennsylvania Projection Series, Summary Report, Employment by Labor Market Area, and Population and Labor Force by County for 1980, 1985, 1990, 1995 and 2000, Report No. 78, PPS-1, prepared by the Office of State Planning and Development, State Economic and Social Research Data Center, June 1978.

TABLE 2.4-1
Sector and Zone Designators for Population Distribution Map[1]

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

Notes:
1. An area is identified by a sector and zone alphanumeric designator (refer to Figure 2.4-1). Thus, area A1 is that area, which lies between 348.75- and 11.25-degrees true north from the facility out to a radius of 1-mile. Area G4 would be that area between 123.75- to 146.25-degrees and the 3- and 4-mile arcs from the facility.
2. The letters I and 0 have been omitted from sector designators so as to eliminate possible confusion between letters and numbers.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Notes:

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

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

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

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

Counties in Con Ed Study Area			Intensive				Low Intensive	
State	In RPA Region	Outside RPA Region	Residential	Community Industrial/ Commercial	Facilities Community	Parks Recreation	Public Rights-Of Way	Open
N.Y.	Suffolk		279 (199)$_2$	50 (35)	92	84	72	172
		Sullivan	[8]$_1$	[4]	[117]	[106]	[90]	[217]
		Ulster	[53]	[12]	[207]	[188]	[160]	[386]
	Westchester		162	31	53	48	42	99
	Bronx		25	4	3	3	2	5
	Kings		42	7	4	4	4	8
	New York		14	2	1	1	1	3
	Queens		65	11	7	7	6	13
	Richmond		39	7	3	2	2	5
P.A.		Pike	[7]	[1]	[76]	[69]	[59]	142
	Total RPA Region$_3$		2040	368	794$_3$*	724$_3$*	617$_3$*	1482$_3$*
	Total Consolidated Edison Area		2078	383	1385	1261	1073	2583

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

2.4 FIGURES

2.5 HYDROLOGY

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

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

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

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

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

The average downstream flow (for a 17-year period preceding 1930) is in excess of:[2]

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

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

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

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

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

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

1. Flooding resulting from runoff generated by a probable maximum precipitation over the entire Hudson River drainage basin upstream of the site.

2. Flooding caused by the occurrence of any upstream dam failure concurrent with heavy runoff generated by a standard project flood.

3. Flooding due to the occurrence of a probable maximum hurricane concurrent with a spring high tide in the Hudson River.

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

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

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

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

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

Within a 5-mile radius of the plant 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 plant 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 plant, Queensboro Lake, supplies water to a state park area only. The location of these reservoirs, and others within a 15-mile radius of the site, are shown on Figure 2.5-2. The city of New York's Chelsea Pumping Station is located about 1-mile north of Chelsea, New York, on the east bank of the Hudson River, about 22-miles upstream of the site. Water will be pumped from intakes in the river at the rate of up to 100 million gal per day into the city reservoir system as required to supplement the primary supply from watersheds during severe drought conditions. This source, however, was not used during the recent 1981 drought.

The discharge of any contaminant into a tidal estuary will result in its distribution throughout the estuary. Factors affecting this distribution include tide amplitude and current, river geometry, salinity distribution, and freshwater discharge. Quirk, Lawler, and Matusky investigated for Con Edison the influence of these factors and determined the effect of radioactive discharges on overall river concentrations, and specifically conditions at Chelsea Pumping station, as discussed in Section 11.1. During normal operations, the plant discharge will not exceed its maximum permissible concentration. Compliance with regulatory release limits is further discussed in Section 11.1.

REFERENCES FOR SECTION 2.5

1. Letter report of Karl L. Kennison, Civil and Hydraulic Engineer, to G. R. Milne, Con Edison, November 28, 1955.

2. Metcalf and Eddy Engineers, Hydrology of Indian Point Site and Surrounding Area, October 1965.

3. Quirk, Lawler, and Matusky Engineers, Transport of Contaminants in the Hudson River above Indian Point Station, May 1966.

4. Quirk, Lawler, and Matusky Engineers, Evaluation of Flooding Conditions at Indian Point Nuclear Generating Unit No. 3, April 1970.

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

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

Component Flow at Indian Point	Elevation at the Battery - Datum MSL (ft)	Flow at Indian Point (million cfs)	Elevation at Indian Point - Datum-MSL (ft)	Elevation at Indian Point Including Local Oscillatory Wave Height Datum MSL (ft)
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

2.5 FIGURES

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

2.6 METEOROLOGY

2.6.1 General

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

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

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

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

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

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

The atmospheric dispersion factors required for the safety analysis of Chapter 14 have been computed for the worst possible meteorological conditions that could prevail at the Indian Point site.

A search of the records indicates that the most protracted consecutive period during which the wind direction was substantially from the same direction was 5 days. The winds in this case

were from the northwest and speeds ranged from 15 to 30 mph. Therefore, this case does not represent the most conservative meteorology associated with the loss-of-coolant accident.

The most frequent wind flow at low heights under inversion conditions is down the axis of the valley. This direction, roughly 10- to 30-degrees, is also the direction of maximum wind frequency. Because of the relatively high frequency of inversion conditions associated with this wind direction, the safety analysis assumes that the distribution of wind speed and thermal stability during the hypothetical accident is exactly that measured at the 100-ft tower level for the 5- to 20-degree wind direction.

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

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

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

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

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

months (September and October) and is undoubtedly conservative with varying degrees of conservatism for the remainder of the year.

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

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

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

REFERENCES FOR SECTION 2.6

1. New York University, Research Division, A Micrometeorological Survey of the Buchanan, N.Y., Area, NYU Technical Report 372.1, November 1955, which was Exhibit L-1, Docket 50-3, given in its entirety. The topography of the area surrounding the site is described and the effects of the topography on meteorological conditions are discussed. The types of data collected, the methods and frequencies of collection, the description and location of the equipment, and the general scope of the meteorological program are indicated in this report. Seasonal wind characteristics, including speeds, directions, and frequencies are tabulated.

2. New York University, Research Division, Evaluation of Potential Radiation Hazard Resulting From Assumed Release of Radioactive Wastes to the Atmosphere From Proposed Buchanan Nuclear Power Plant, Sections 1, 2, and 3 of NYU Technical Report 372.3, April 1957. This report was submitted to the NRC in its entirety as Exhibit L-5, Docket 50-3. These sections discuss the diffusive conditions and the climatological data of the site. The basis for evaluating the diffusion parameters selected for the safety analysis is given on pages 19 to 21. Section 3 contains tables of frequency distribution of diffusion classes and wind directions, and also wind roses.

3. New York University, Research Division, Summary of Climatological Data at Buchanan, N.Y., 1956-1957, NYU Technical Report 372.4, March 1958, was Exhibit L-6, Docket 50-3. This report summarizes the final meteorological testing at Indian Point.

2.6 FIGURES

2.7 GEOLOGY AND SEISMOLOGY

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

In 1982, a report by Woodward-Clyde Consultants was done to update Section 2.7 of the FSAR. The previous studies are listed in the reference list of the report. The report is included in Appendix 2B.

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

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

REFERENCES FOR SECTION 2.7

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

2.8 ENVIRONMENTAL RADIOACTIVITY

Monitoring for environmental radioactivity in the vicinity of the Indian Point Station began in 1958, approximately 4 years before the operation of Unit 1. Measurements since that time have indicated that the present operation of Units 2 and 3 and the past operation of Unit 1 have had no significant effect on the environment. The monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Measurements of radioactivity in the environment are summarized in the Annual Radiological Environmental Operating Report, which is submitted annually as required by the plant's Technical Specifications.

Determinations of radioactivity in the environment are made regularly. Samples include drinking water from two nearby reservoirs and the New York City Aqueduct, river water, sediments, fish, shellfish, lake and river aquatic vegetation, land vegetation, soil from locations in the vicinity of the site, shoreline soil, air, and milk along with direct gamma radiation measurements in various locations.

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 plant.

2. To ensure that the effluent releases are kept as low as is reasonably achievable (ALARA) and within allowable limits in accordance with 10 CFR 20.

3. To verify projected and anticipated radioactivity concentrations in the environment and related exposure from releases of radioactive material from the Indian Point site.

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

Environmental surveys have also been confirmed by Dr. Merril Eisenbud, Director of Environmental Radiation Laboratory, Institute of Environmental Medicine, New York University Medical Center, who has found that the levels of environmental radioactivity are associated with natural background and fallout of nuclear weapons testing.[2]

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

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

REFERENCES FOR SECTION 2.8

1. New York State Department of Health, Division of Environmental Health Services, Consolidated Edison Indian Point Reactor, Post Operational Survey, August 1965.

2. New York State Department of Health, Division of Environmental Health Services, Consolidated Edison Indian Point Reactor, Post Operational Survey, July 1966.

3. New York University Medical Center Institute of Environmental Medicine, Ecological Survey of the Hudson River: Progress Report No. 2, submitted to

Division of Radiological Health, USPHS, Contract PHS 86-95, Neg. 141, December 1966.

4. F. Cosolito, et al., Radioactivity in the Hudson River, Symposium on Hudson River Ecology, Hudson River Valley Commission of New York, October 4-5, 1966.

NOTE: This information is classified as Historical Information

APPENDIX 2A

FACILITY SAFETY ANALYSIS REPORT (FSAR)

CONSOLIDATED EDISON COMPANY
OF NEW YORK, INCORPORATED

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

METEOROLOGICAL UPDATE

SEPTEMBER, 1981

YSC PROJECT + 01-4122

prepared by:

EDWARD J. KAPLAN
Senior Scientist

BRUCE R. WUEBBER
Staff Meteorologist

ATMOSPHERIC SERVICES DEPARTMENT

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1.0 GENERAL

1.1 HISTORICAL BACKGROUND

1.1.1 Introduction

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

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

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

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

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

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

With respect to Indian Point on-site meteorological measurements, there were for the purpose of the reports that have been cited, three different meteorological towers at three different locations. These are specifically delineated in Figure 1 of Halitsky, Kaplin and Laznow (1971).

The meteorological data collected during the 1955-1957 were from the 300 Foot Tower designated as IP1. Meteorological data collected and reported by Halitsky, Kaplin and Laznow (1971) and Kaplin and Laznow (1972) were from the 100 Foot Tower designated as IP3. The base of IP3 Tower was about 200 feet from the base location of the original IP1 Tower.

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

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

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

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

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

Indian Point and inland to the east. This network consisted of new anemometer stations in addition to the Indian Point 400 Foot Meteorological Tower, the Bowline 350 Foot Meteorological Tower and the U.S. Department of Commerce, NWS Station at Westchester County. These wind data were digitalized and evaluated by New York Services Corporation under contract to Con Edison for the purposes of defining surface air flow patterns within a 10-15-mile range of Indian Point with emphasis on generating refined estimated of southward movements. As completed, ten consecutive months (March 1, 1980 - December 31, 1980) were evaluated and a total of 7,264 eight-hour parcel trajectories were created objectively using appropriate local one-hour wind velocity averages on a real time basis. The results of this study were submitted to Con Edison: Kaplin, Edward J. and B. Wuebber (1981).

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

1.1.2 Tower Siting and Instrumentation

1.1.2.1 Hudson River

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

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

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

1.1.2.2 General Topography

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

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

The Hudson River, flowing southward through the valley, resembles a gourd with its curved ¾-mile thick northern neck nestling at the base of Dunderberg Mountain while the bulbous three mile thick body fills the southern part of the valley between South Mountain and Prickly Pear Hill. The Indian Point peninsula lies in the hollow of the curved neck.

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

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

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

1.1.2.3 Site Configuration

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

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

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

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

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

* Intermittent usage.

TABLE 1 (Cont'd)

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

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

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

2.0 122M METEOROLOGICAL TOWER

2.1.1 Siting

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

2.1.2 Instrumentation

2.1.2.1 Sensor Configuration

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

2.1.2.2 Instrumentation Specifications

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

2.1.2.2.1 Climatronics F460 Wind Speed Transmitter

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

2.1.2.2.2 Climatronics F460 Wind Direction Transmitter

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

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

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

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

Temperature:

Range:	-34 to +50°C
Accuracy:	± 0.2°C
Time Constant:	10 sec. To 63% (in TS-10 Shield)
Linearity:	± 0.2%

Delta Temperature:

Range:	± 10°F
Sensitivity:	0.02°F
Accuracy:	0.1°F or ± 5% of delta-T not to exceed 0.3°F
Response Time:	10 sec. To 63% in TS-10 Shield

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

Range:	-40° to 42°C
Accuracy:	± 0.5°C
Response Time:	1°C/min.

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

Accuracy:	± 1% up to 3"/hr
Resolution:	0.01"
Size:	8" diameter x 24" height
Conversion Accuracy:	± 0.2%

2.1.2.2.7 Data Collection Systems

Analog:

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

Digital:

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

2.1.3 Meteorological Support System

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

2.2 DATA LOG

2.2.1 Indian Point Tower IP3

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

2.2.2 122 Meter Meteorological Tower (IP4)

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

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

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

TABLE 2
VALID DATA LOG*

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

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

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

3.0 ANALYSES DATA

3.1 INDIAN POINT TOWER IP3

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

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

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

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

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

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

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

3.2 122M METEOROLOGICAL TOWER (IP4)

3.2.1 <u>October 1, 1973 to August 31, 1974</u>

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

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

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

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

TABLE 3
COMPARISON OF ANNUAL PERCENT
OCCURRENCE OF STABILITY CATEGORIES

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

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

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

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

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

3.2.2 August 1, 1978 - July 31, 1979

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

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

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

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

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

3.2.2.1 Surface Air Trajectories Analyses - Summary

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

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

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

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

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

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

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

TABLE 4
SUMMARY OF TRAJECTORY END-POINTS

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

( ) =Total number of trajectories generated.

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

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

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

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

					% TOTAL
HR 1		35	164	133	46.0
		26	8	15	6.8
		232	84	25	47.2
	% TOTAL	40.6	35.4	24.0	99.0

PATTERN 1 / PATTERN 2

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

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

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

					% TOTAL
		25	52	86	23.1
HR 1		10	10	27	6.7
		154	72	271	70.3
	% TOTAL	26.7	19.0	54.3	98.1

	PATTERN 1					PATTERN 2			
				% TOTAL					% TOTAL
	22	21	58	20.8		24	21	56	20.8
HR 2	14	5	16	7.2		12	5	16	6.8
	133	41	176	72.0		100	45	207	72.4
% TOTAL:	34.8	13.8	51.4	67.5	% TOTAL	28.0	14.6	57.4	67.5
				% TOTAL					% TOTAL
	17	4	19	17.7		14	4	21	17.3
HR 4	7	0	3	4.4		7	0	3	4.4
	67	16	92	77.8		57	14	106	78.3
% TOTAL:	40.4	8.9	50.7	31.3	% TOTAL	34.5	8.0	57.4	31.5
				% TOTAL					% TOTAL
	16	2	7	15.9		15	2	8	15.9
HR 6	2	1	0	3.2		2	1	0	1.9
	46	8	73	80.9		39	8	82	82.1
% TOTAL:	40.8	7.0	51.0	21.9	% TOTAL	35.7	7.0	57.3	21.9
				% TOTAL					% TOTAL
	9	1	4	12.4		6	0	4	8.7
HR 8	3	1	0	3.5		3	1	0	3.5
	34	10	51	84.1		29	9	63	87.8
% TOTAL:	40.7	10.6	48.7	15.8	% TOTAL	33.0	8.7	58.3	16.1

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

TABLE 7
SUMMATION TRAJECTORY OCCURRENCES
SOUTH OF INDIAN POINT

August, 1978

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

JANUARY, 1979

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

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

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

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

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

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

3.2.3 March 1980 - December 1980

3.2.3.1 General

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

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

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

3.2.3.2 Wind Frequency Distributions

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

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

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

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

TABLE 8
LOCATIONS OF STATIONS RELATIVE TO INDIAN POINT

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

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

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

3.2.3.3 Diurnal Wind Distributions

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

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

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

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

3.2.3.4 Resultant and Concurrent Hourly Winds

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

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

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

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

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

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

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

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

* Less than 10 valid data points

TABLE 11 (Cont'd)

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

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

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

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

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

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

3.2.3.5 Summary-Trajectory II Study

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

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

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

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

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

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

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

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

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

- Nocturnal flow patterns generally north to south with minimal with speeds

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

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

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

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

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

TABLE 14
SUMMARY OF TRAJECTORY END-POINT COUNTS

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

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

TABLE 15
SUMMARY OF TRAJECTORY END-POINTS (Percent)

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

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

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

Peekskill Bay
Haverstraw Bay
Tappan Zee

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

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

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

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

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

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

3.2.4 January 1, 1979 Through December 31, 1980

3.2.4.1 Data Analyses

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

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

3.2.4.2 Wind Frequency Distributions

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

frequency extremes are noted on a nine year basis for La Guardia and a 21 year basis for Bridgeport.

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

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

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

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

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

TABLE 16A
HISTORICAL COMPARISONS OF WIND FREQUENCY DISTRIBUTIONS
MARCH

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

* 9 Year High (1972-1980)
** 9 Year Low (1972-1980)
21 Year Low (1960-1980)
21 Year High (1960-1980)

TABLE 16B
HISTORICAL COMPARISONS OF
WIND FREQUENCY DISTRIBUTIONS
JULY

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

* 9 Year High (1972-1980)
** 9 Year Low (1972-1980)
# 21 Year High (1960-1980)
21 Year Low (1960-1980)

TABLE 16C
HISTORICAL COMPARISONS OF
WIND FREQUENCY DISTRIBUTIONS
DECEMBER

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

TABLE 17
COMPARISON OF PERCENT WIND FREQUENCY DISTRIBUTIONS – SUMMER

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

TABLE 18
COMPARISON OF PERCENT WIND FREQUENCY DISTRIBUTIONS - WINTER

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

3.2.4.3 Diurnal Wind Direction Distributions

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

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

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

3.2.4.4 Wind Speed Distributions

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

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

TABLE 19
COMPARISON OF DIURNAL RESULTANT WIND DIRECTIONS

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

Job Number: 01-4122-00

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

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

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

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

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

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

Job Number: 01-4122-00

TABLE 21

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

DIURNAL ANALYSIS

MAY 1, 79,80 – OCT. 31, 79,80

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

CUMULATIVE PROBABILITY DISTRIBUTION

MAY 1, 79, 80 - OCT 31, 79, 80

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

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

Job Number: 01-4122-00

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

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

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

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

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

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

Job Number: 01-4122-00

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

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

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

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

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

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

TABLE 24
MAXIMUM DIURNAL WIND SPEEDS (MPH)

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

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

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

3.2.4.5 Wind Velocities and Atmospheric Stability

3.2.4.5.1 Joint Frequency Distribution of Wind Direction and Stability

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

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

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

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

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

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

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

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

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

3.2.4.5.2 Frequency of Occurrence of Stability Categories

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

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

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

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

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

TABLE 28
HISTORICAL COMPARISONS OF PERCENT OCCURRENCE OF STABILITY

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

* Concurrent basis

[1] Composite year with temperature correction - concurrent basis (FSAR 3)

1* Composite year concurrent basis (FSAR 3)

** March-December only concurrent basis

NA Not Available

NOTE: Gradient for IP3 Tower: 30-2 M

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

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

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

* Single observations
** Less than ten observations

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

The average wind speeds as a function of stability category are shown in Table 33 for the 1973-1974 data collection period and the 1979-1980 data period. As noted, there

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* Single observations
** Less than 15 observations

are some apparent anomalies with respect to the 122 meter level wind speeds. These are probably induced by the few data points available in 1973-1974 rather than a reality. Where the total number of occurrences were in a more reliable range, at the 10 meter and the 122 meter levels, there was good agreement between average wind speeds.

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

4.0 SUMMARY

4.1 METERORology

4.1.1 General

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

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

4.1.2 122 Meter Meteorological Tower System

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

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

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

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

Appropriate meteorological data are transmitted from the Meteorological Trailer to:

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

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

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

4.1.3 Local Meteorological Characteristics

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

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

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

Peekskill Bay
Haverstraw Bay
Tappan Zee

4.1.4 Conclusion

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

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NOTE: This information is classified as Historical Information

APPENDIX 2B

INDIAN POINT FSAR
UPDATE

REVISED

Prepared for:

Consolidated Edison Company, Inc.
of New York
4 Irving Place
New York, NY 10003

Submitted by:

WOODWARD-CYCLE CONSULTANTS
201 Willowbrook Boulevard
Wayne, New Jersey 07470

16 April 1982

TABLE OF CONTENTS

INDIAN POINT FSAR UPDATE

INTRODUCTION

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

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

Physiography

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

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

General Geology and Tectonics

The eastern third of the North American continent has been the site of episodic tectonism since Precambrian time (see Table1). Paleozoic aged tectonism has molded a broad geologically varying zone known as the Appalachian orogen. The core of the orogen is marked by intrusive rocks modified by ductile and brittle deformation and regional metamorphism. The Indian Point site lies within the Manhattan Prong of the Appalachian Mountains.

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

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

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

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

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

Geology of the Hudson Highlands

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

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

Geology of the Manhattan Prong

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

The stratigraphy of the Manhattan Prong has long been the subject of controversy and frequent revision, however, most recent workers recognize five formations within the prong. In order of decreasing age they are the Fordham Gneiss, the Yonkers-Pound

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

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

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

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

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

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

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

Fordham E - Silliceous biotite-quartz plagioclase gneiss.

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

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

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

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

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

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

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

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

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

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

The Geology of the Newark Basin

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

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

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

History of Brittle Deformation

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

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

The earliest documented movement along the Ramapo fault system is Grenville age (Ratcliffe 1976). Textural evidence seen in the Canopus Pluton indicates that movement along the Canopus Hollow fault (north of Peekskill) was synchronous with emplacement of the pluton. Flow structures and mylonitization displayed within the pluton indicate crystallization during shearing. Drag folds and the overall shape of the pluton suggest right lateral strike-slip motion. The Canopus Pluton, a diorite-monzontie,

has been radiometrically dated at 1150 m.y. (Rb/Sr), thus providing a minimum age for movement along the fault.

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

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

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

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

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

Reactivation and development of new faults occurred during Mesozoic time (Ratcliffe, 1980). Deep seated zones of weakness in crystalline basement were utilized in the

development of the Triassic basin, particularly the Ramapo-Cheesecote and Mott Farm Road faults (Ratcliffe, 1976). Structural evidence indicates that normal faulting was dominant during mesozoic time, with the latest activity along the Mott Farm Road branch of the Ramapo, dated at 163 m.y. by K-Ar methods (Ratcliffe, 1976). Late north-south vertical strike-slip faults with both left-lateral and right-lateral movements are present in the vicinity of Indian Point. The relationship of these faults to the Ramapo is uncertain, although the faults are probably Mesozoic in age (Ratcliffe, 1976; Dames & Moore, 1977).

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

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

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

Recent Seismicity

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

regional overview does suggest a higher level of activity in the northern New Jersey, southeastern New York area than that of the surrounding areas.

A number of hypotheses have been proposed to explain the observed pattern of seismicity. Current seismicity in the northeastern United States has been attributed to: a proposed stress system in which the maximum compressive stress trends to the northwest (Yang and Aggarwal, 1981); proximity to relatively young igneous bodies (McKeown, 1978); the response of the crust to glacial unloading (Stein and others, 1979); and, in the immediate region about the site, proposed reactivation of pre-existing fault zones based on a spatial correlation with surface traces of faults exposed within the region (Aggarwal and Sykes, 1978).

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

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

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

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

regime has been proposed that adequately accommodates the observed pattern of low level seismicity, regional stress measurement data and those stresses inferred from recently reported focal mechanism solutions.

CONCLUSION

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

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GLOSSARY

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

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

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

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

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

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

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

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

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

chlorite - A green iron-magnesium rich platy mineral.

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

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

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

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

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

euhedral - A crystal bounded by well developed crystal faces.

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

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

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

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

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

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

isocinal folds - A fold in which the limbs are parallel.

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

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

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

metasediment - A metamorphosed sedimentary rock.

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

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

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

orogen - A region that has been subjected to orogeny.

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

pelite - A sedimentary or metamorphic rock rich in aluminum.

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

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

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

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

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

tectonic - Pertaining to the forces that cause crustal deformation.

throw – The vertical component of fault motion.

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

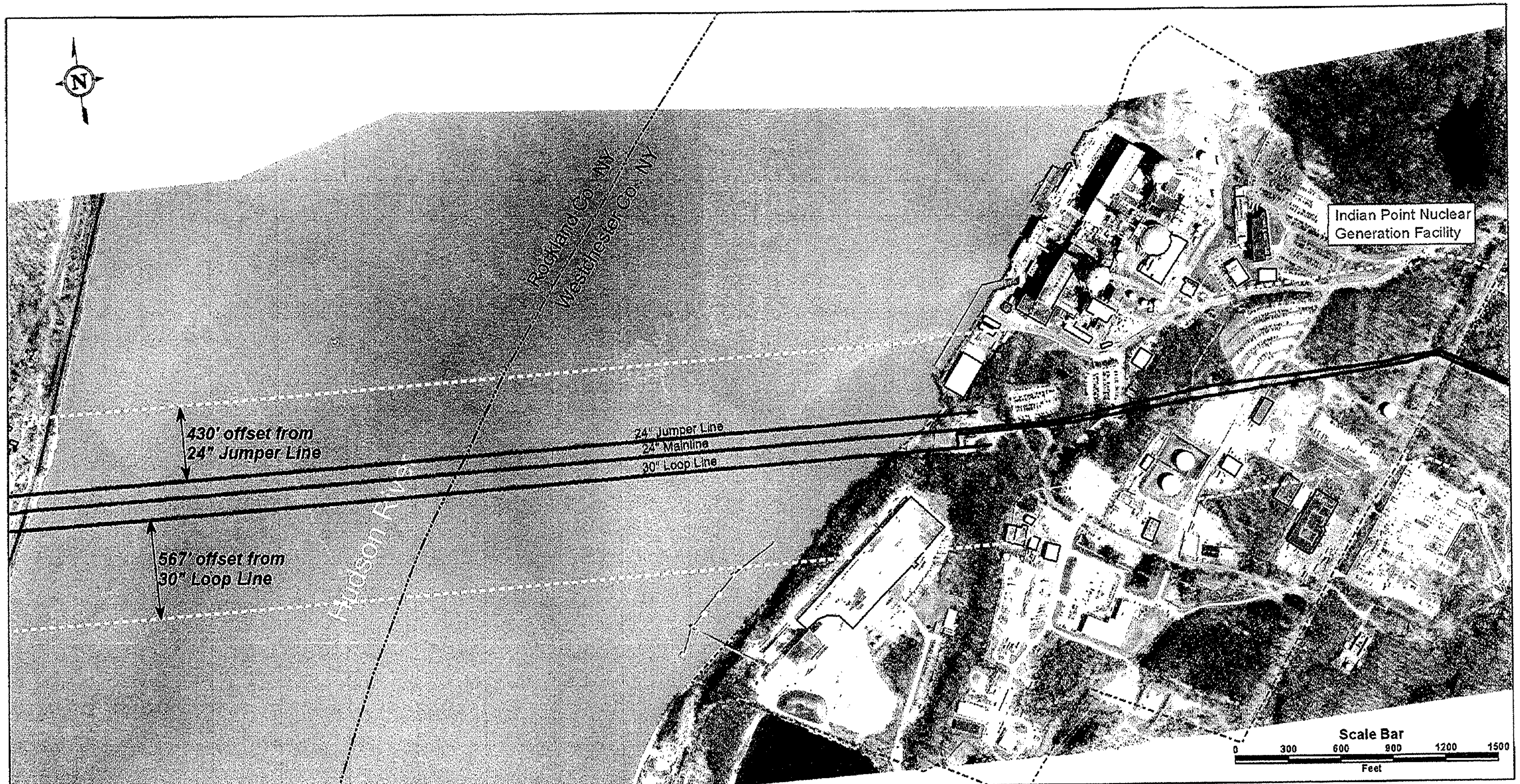
TABLE 1
GEOLOGICAL TIME SCALE

TABLE 2
PROPOSED CORRELATION OF THE STRATIGRAPHIC SUBDIVISIONS IN THE MANHATTAN PRONG WITH ROCKS IN ADJACENT AREAS

TABLE 3
GEOLOGIC HISTORY IN THE CROTON FALLS AREA
(after Brock & Mose, 1979)

FIGURES

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 2.2-1 AERIAL PHOTO OF INDIAN POINT SITE AND SURROUNDING AREA	
MIC. No. 1999MC3572	REV. No. 17A



—·—	Nuclear Plant Property
	Potential Impact Radius Extents
——	Algonquin Gas Trans. Pipeline

Hudson River Crossing & Indian Point Nuclear Generation Facility

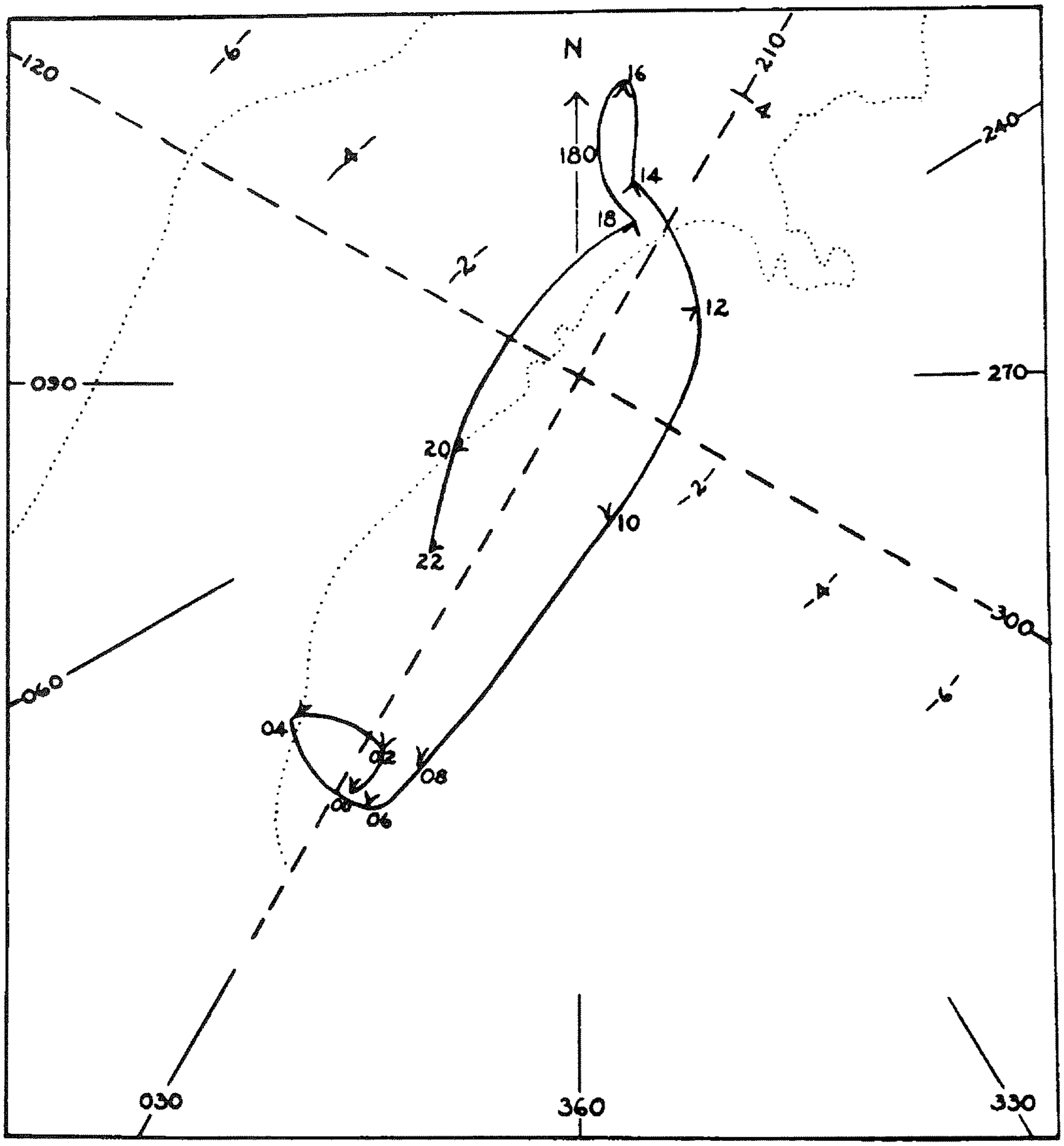
Rockland & Westchester Counties, New York

Algonquin Gas Transmission Co., Segment STON-SEAS

(Imagery flown in 2006)

Spectra Energy

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 2.2-3 HUDSON RIVER CROSSING & INDIAN POINT NUCLEAR GENERATING FACILITY	
UFSAR FIGURE 2.2-3	REV. No. 21



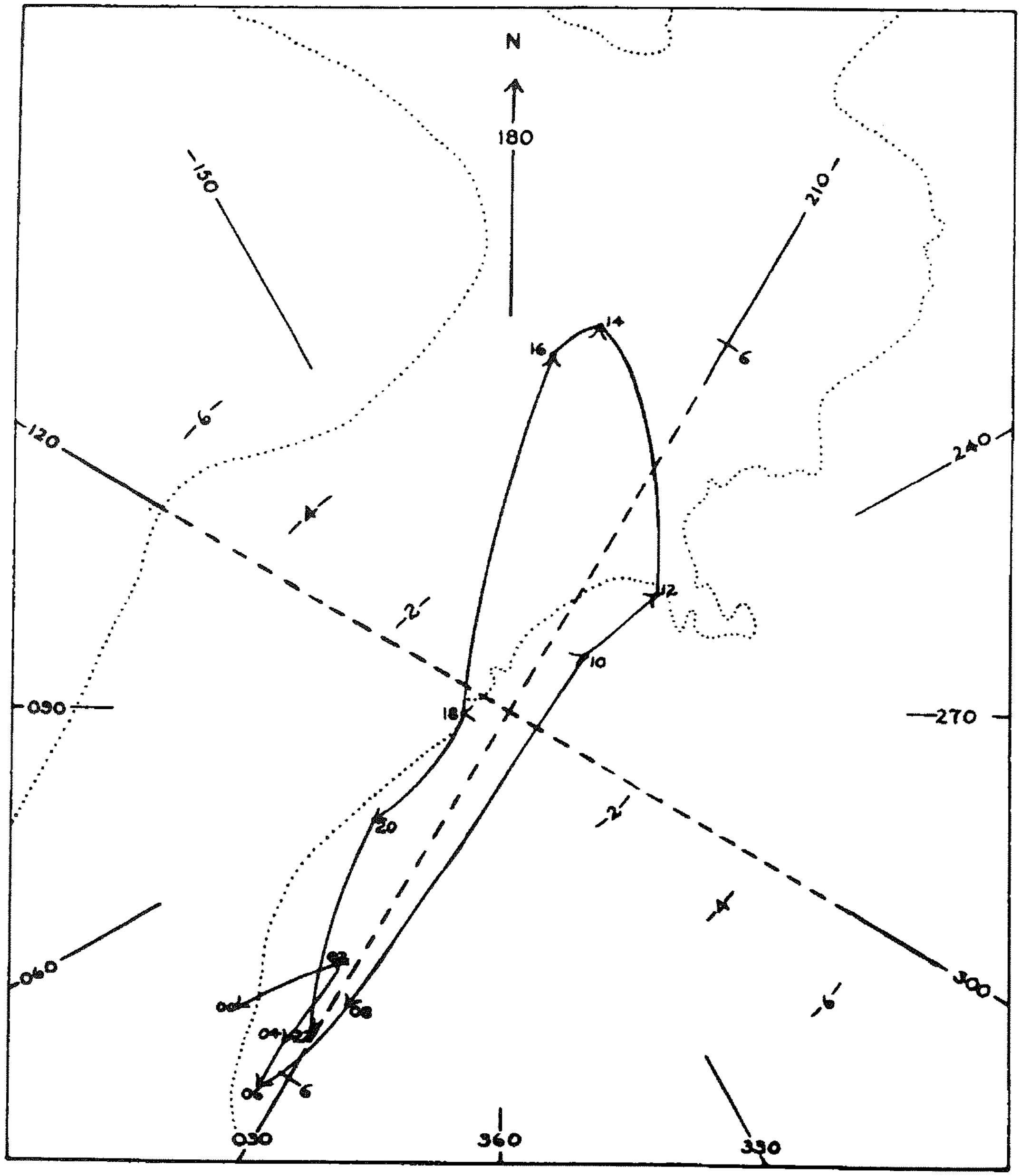
Geostrophic Winds Less Than 16 MPH
September - October, 1955
70 Ft Above River

INDIAN POINT UNIT No. 2

UFSAR FIGURE 2.6-2

DIURNAL VARIATION OF MEAN VECTOR WIND FOR 24 HOUR PERIODS OF WEAK PRESSURE GRADIENT CONDITIONS

MIC. No. 2001MB1193 | REV. No. 17A



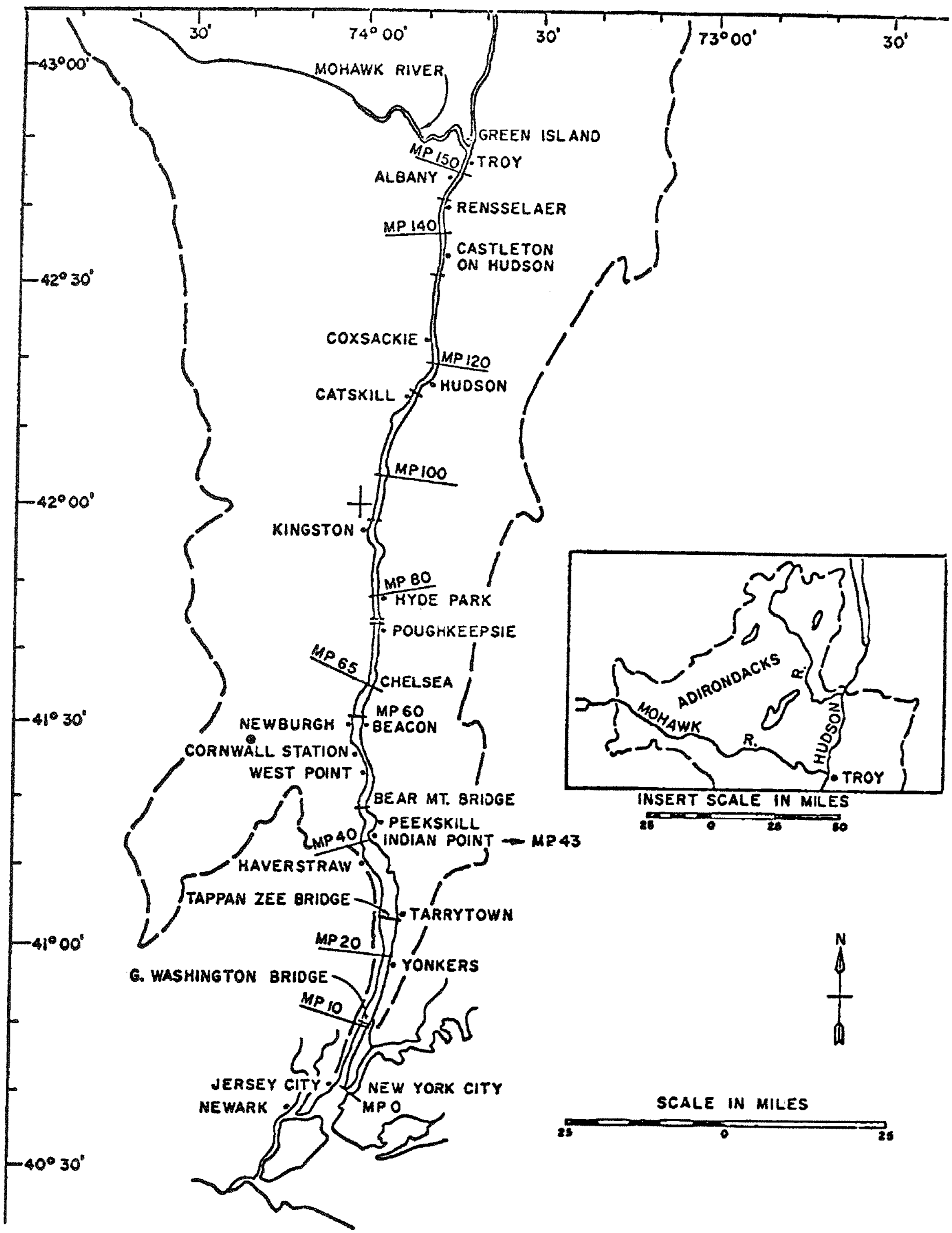
September - October, 1955
70 Ft Above River

INDIAN POINT UNIT No. 2

UFSAR FIGURE 2.6-1

DIURNAL VARIATION OF MEAN VECTOR WIND FOR VIRTUALLY ZERO PRESSURE GRADIENT CONDITIONS

MIC. No. 2001MB1191	REV. No. 17A

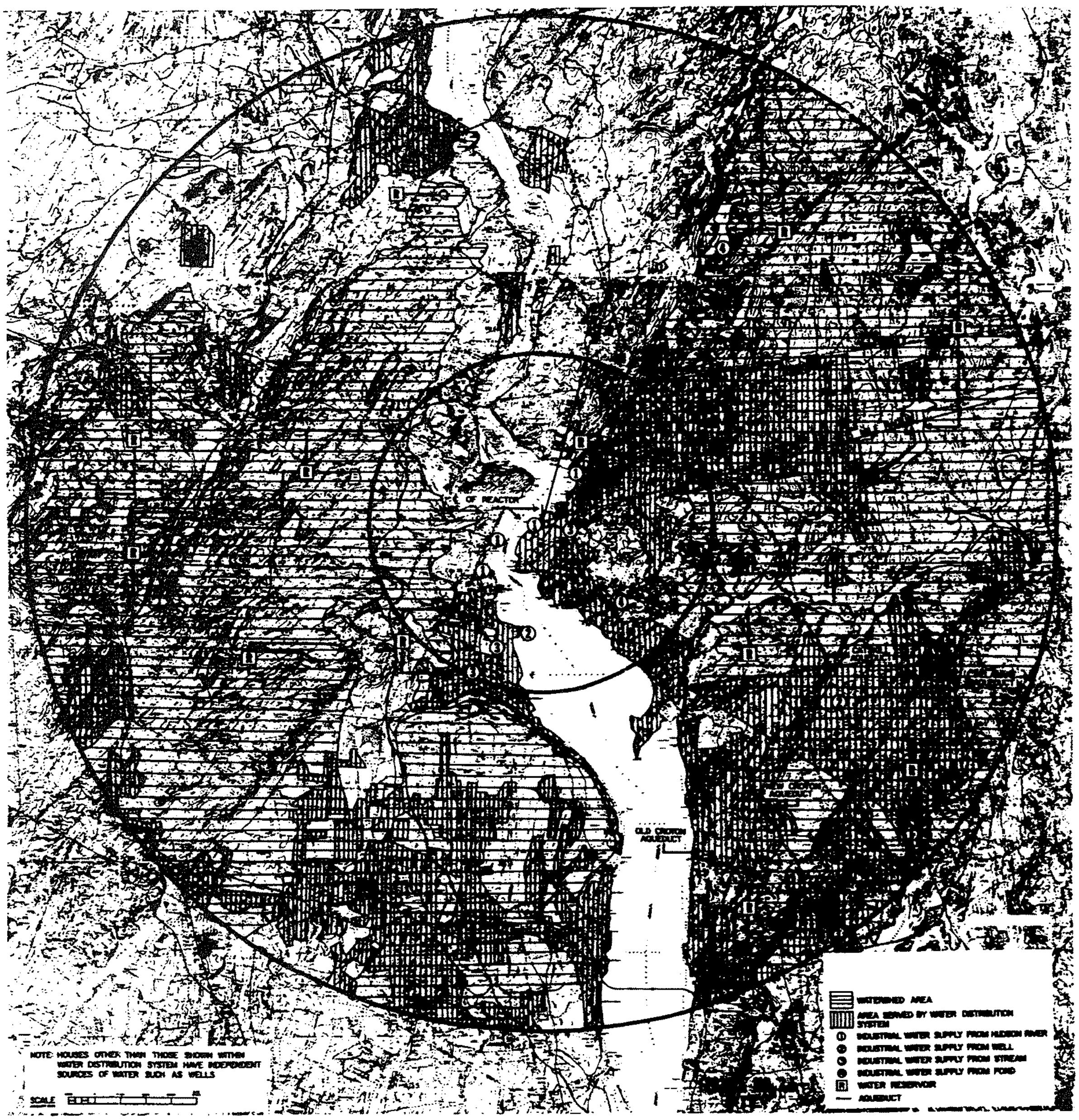


INDIAN POINT UNIT No. 2

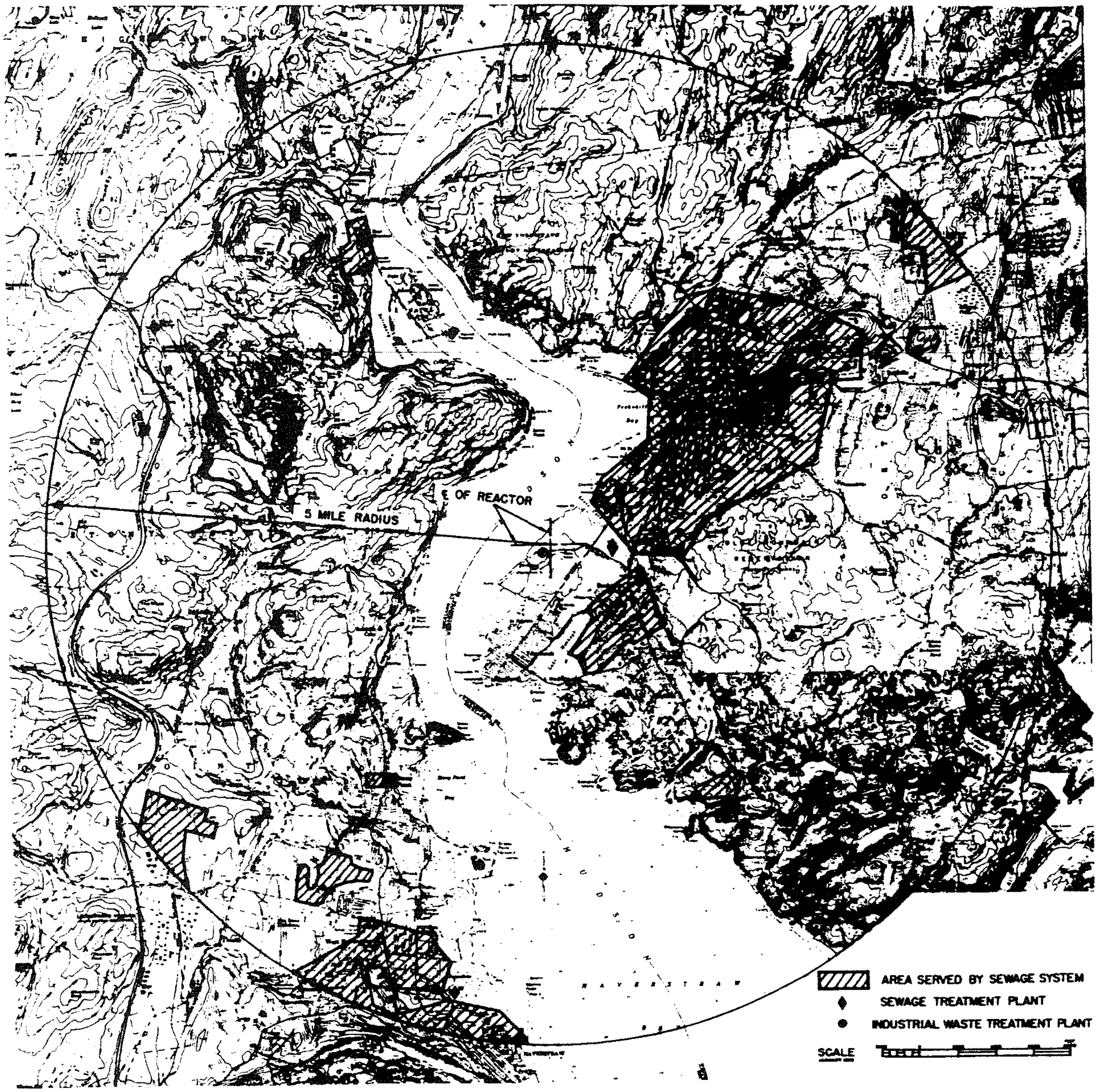
UFSAR FIGURE 2.5-2

HUDSON RIVER DRAINAGE BASIN

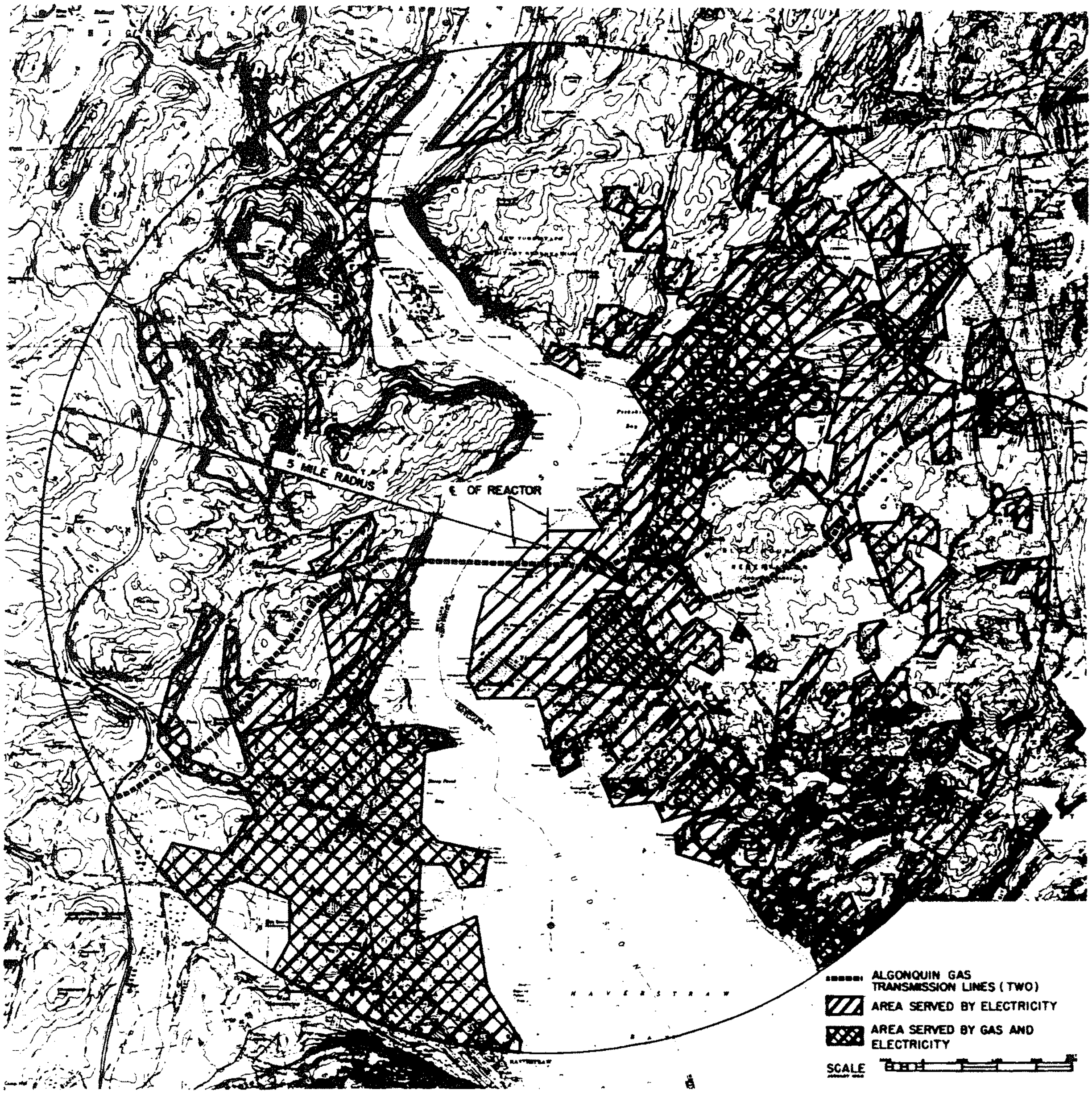
MIC. No. 2001MB1190	REV. No. 17A



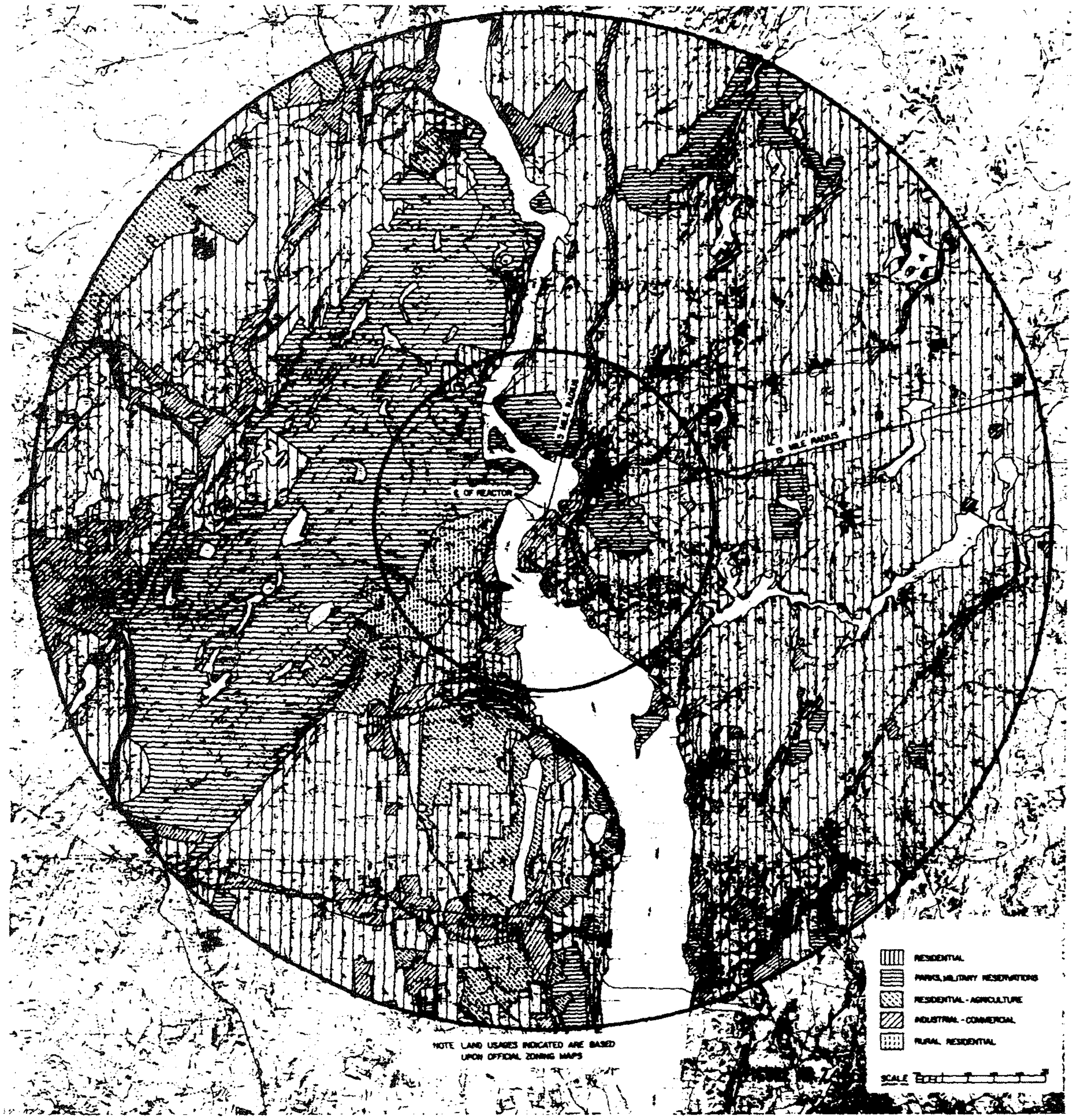
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 2.5–1 MAP & DESCRIPTION SHOWING LOCATION OF SOURCES OF POTABLE & INDUSTRIAL WATER SUPPLIES & WATERSHED AREAS	
MIC. No. 2001MB1189	REV. No. 17A



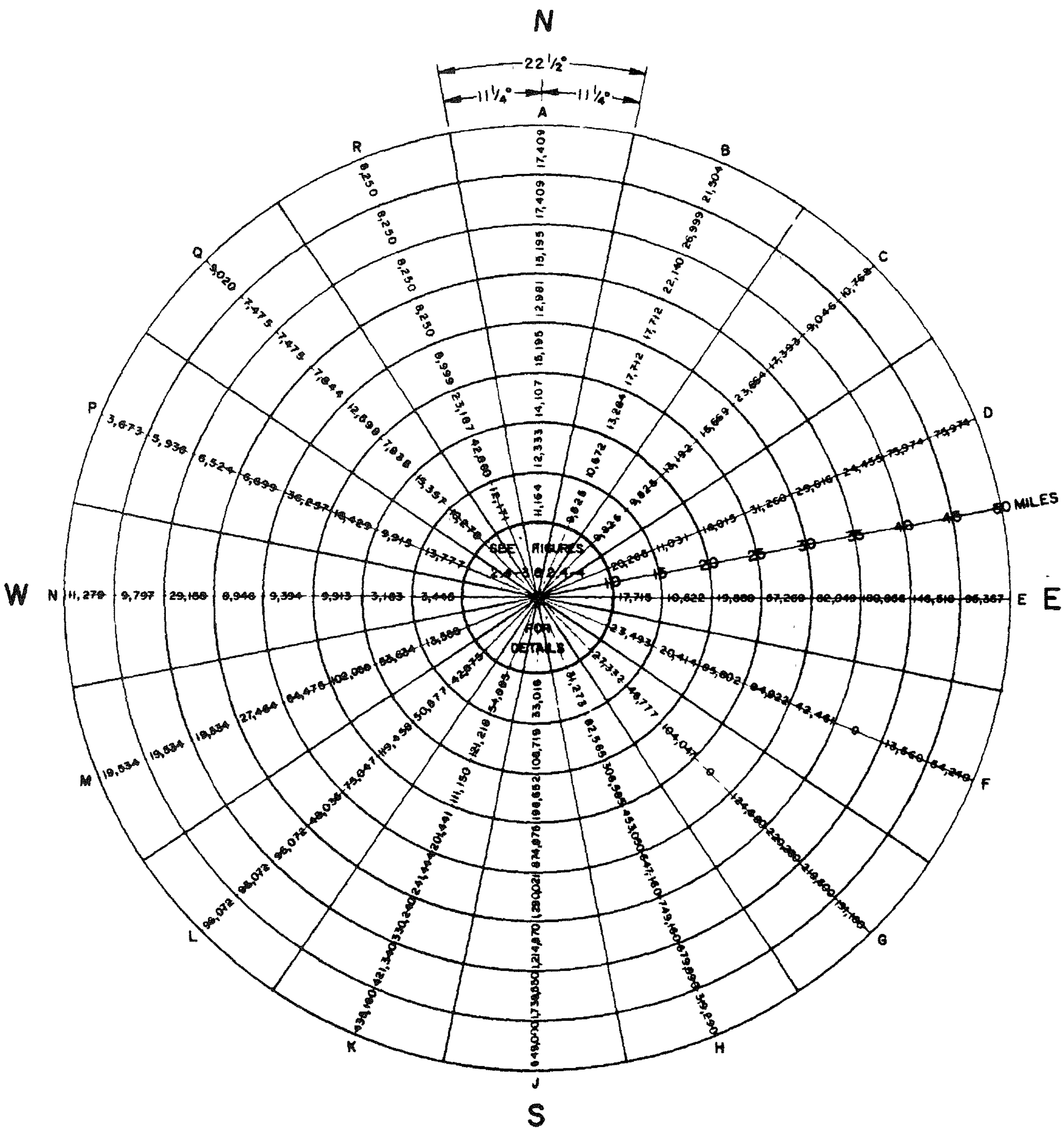
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 2.4-8 MAP AND DESCRIPTION OF THE AREA SHOWING SEWAGE SYSTEMS	
MIC. No. 1999MC3582	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 2.4-7	
MAP AND DESCRIPTION OF THE AREA SHOWING PUBLIC UTILITIES	
MIC. No. 1999MC3581	REV. No. 17A

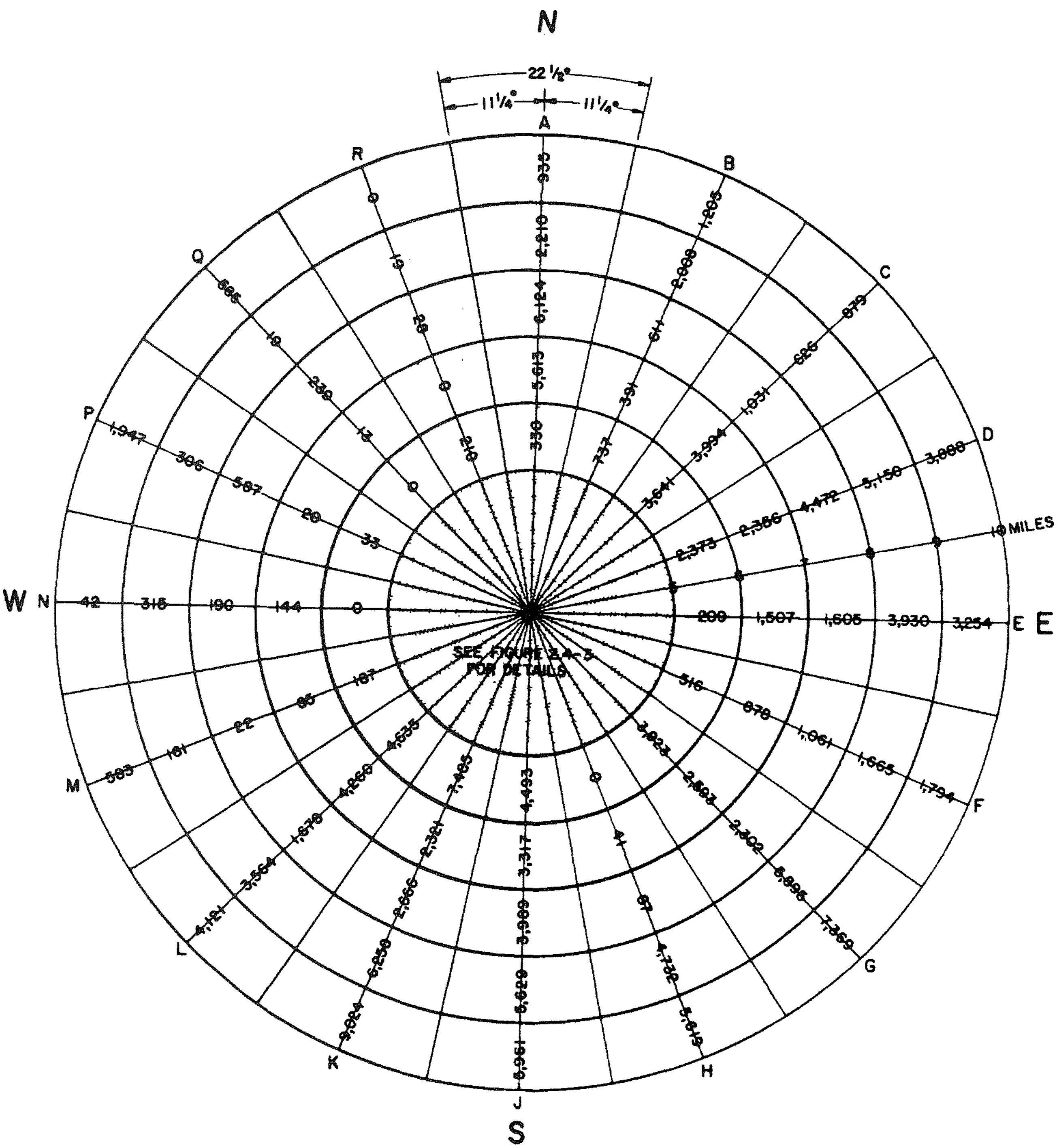


INDIAN POINT UNIT No. 2	
UFSAR FIGURE 2.4–6 MAP AND DESCRIPTION SHOWING LAND USAGE	
MIC. No. 1999MC3580	REV. No. 17A



1980 PROJECTED POPULATIONS

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 2.4-5 FIFTY MILE SECTOR/ZONE DIAGRAM	
MIC. No. 1999MC3579	REV. No. 17A



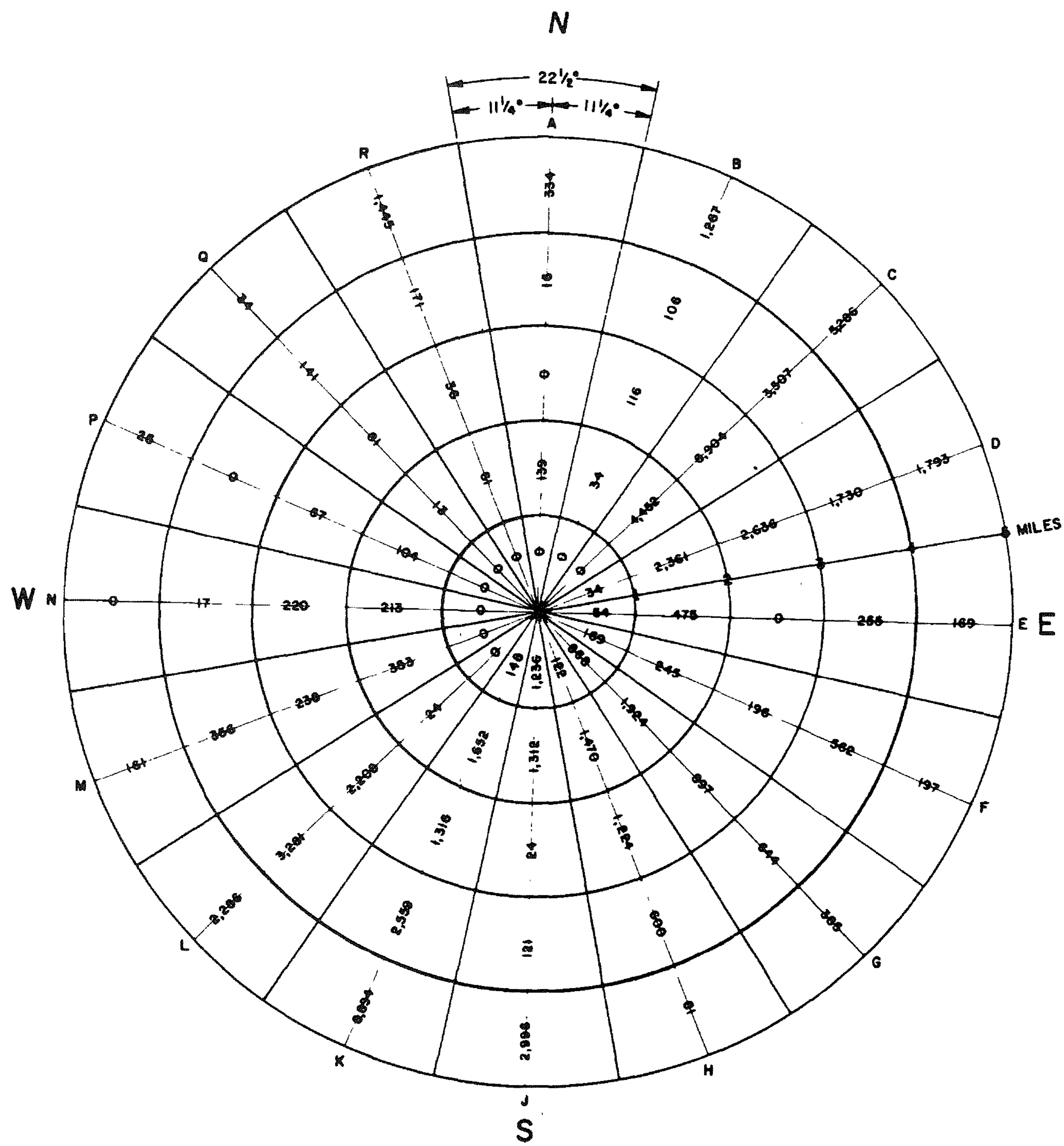
1980 PROJECTED POPULATIONS

INDIAN POINT UNIT No. 2

UFSAR FIGURE 2.4-4

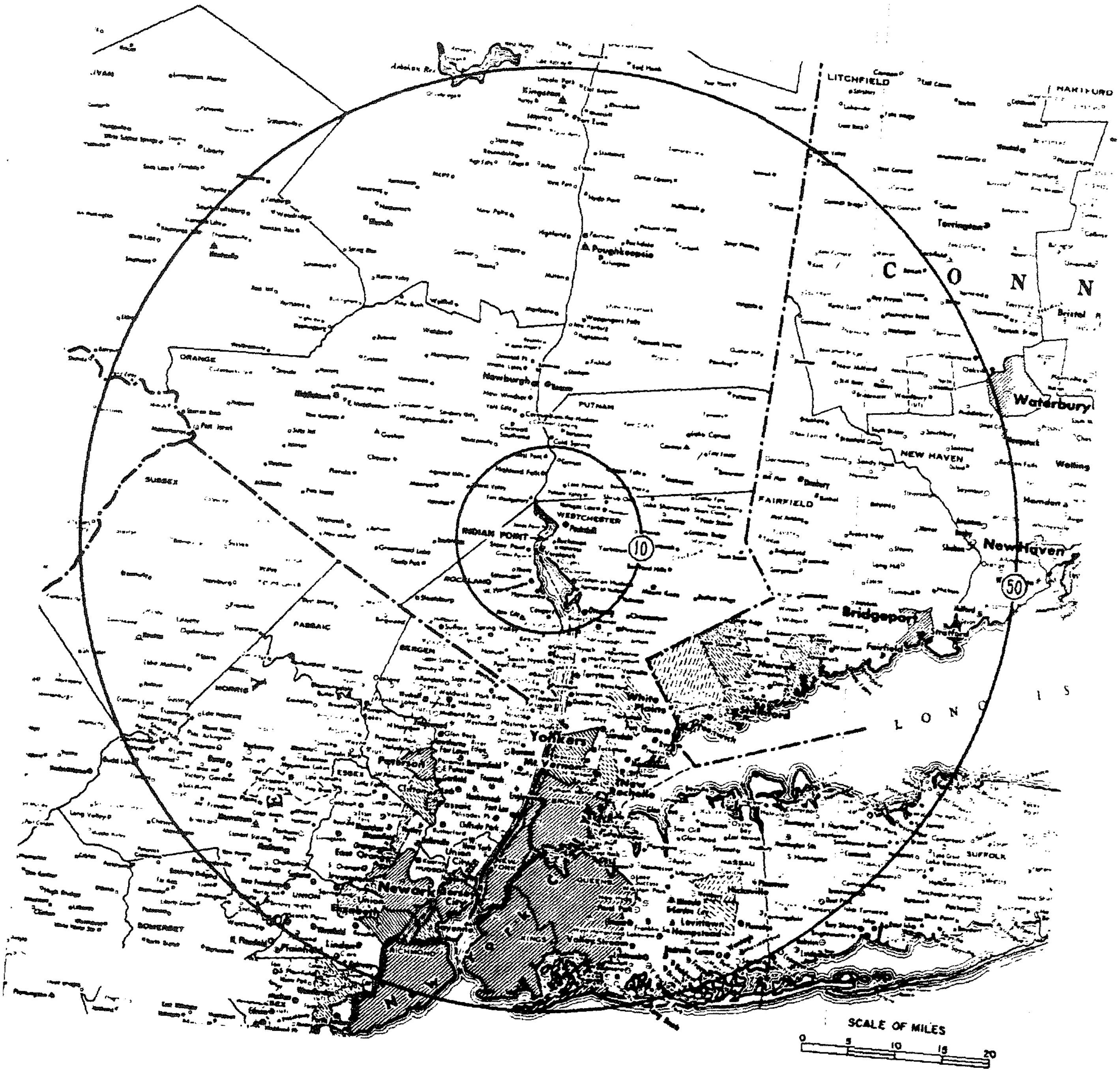
TEN MILE SECTOR/ZONE DIAGRAM

MIC. No. 1999MC3578	REV. No. 17A



1980 PROJECTED POPULATIONS

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 2.4-3 FIVE MILE SECTOR/ZONE DIAGRAM	
MIC. No. 1999MC3577	REV. No. 17A

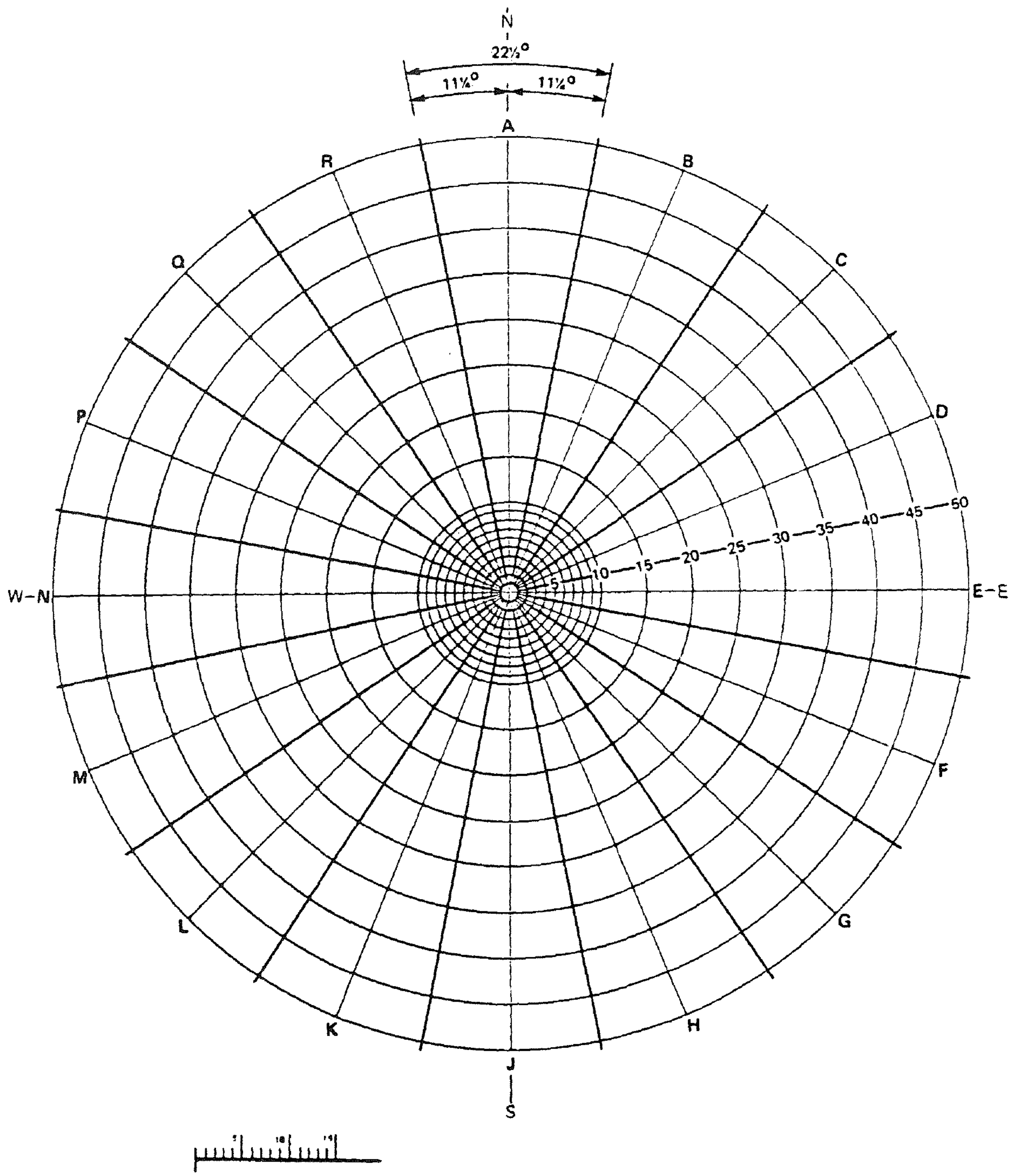


INDIAN POINT UNIT No. 2

UFSAR FIGURE 2.4-2

INDIAN POINT STATION
TEN AND FIFTY MILE RADIUS MAP

MIC. No. 1999MC3576	REV. No. 17A

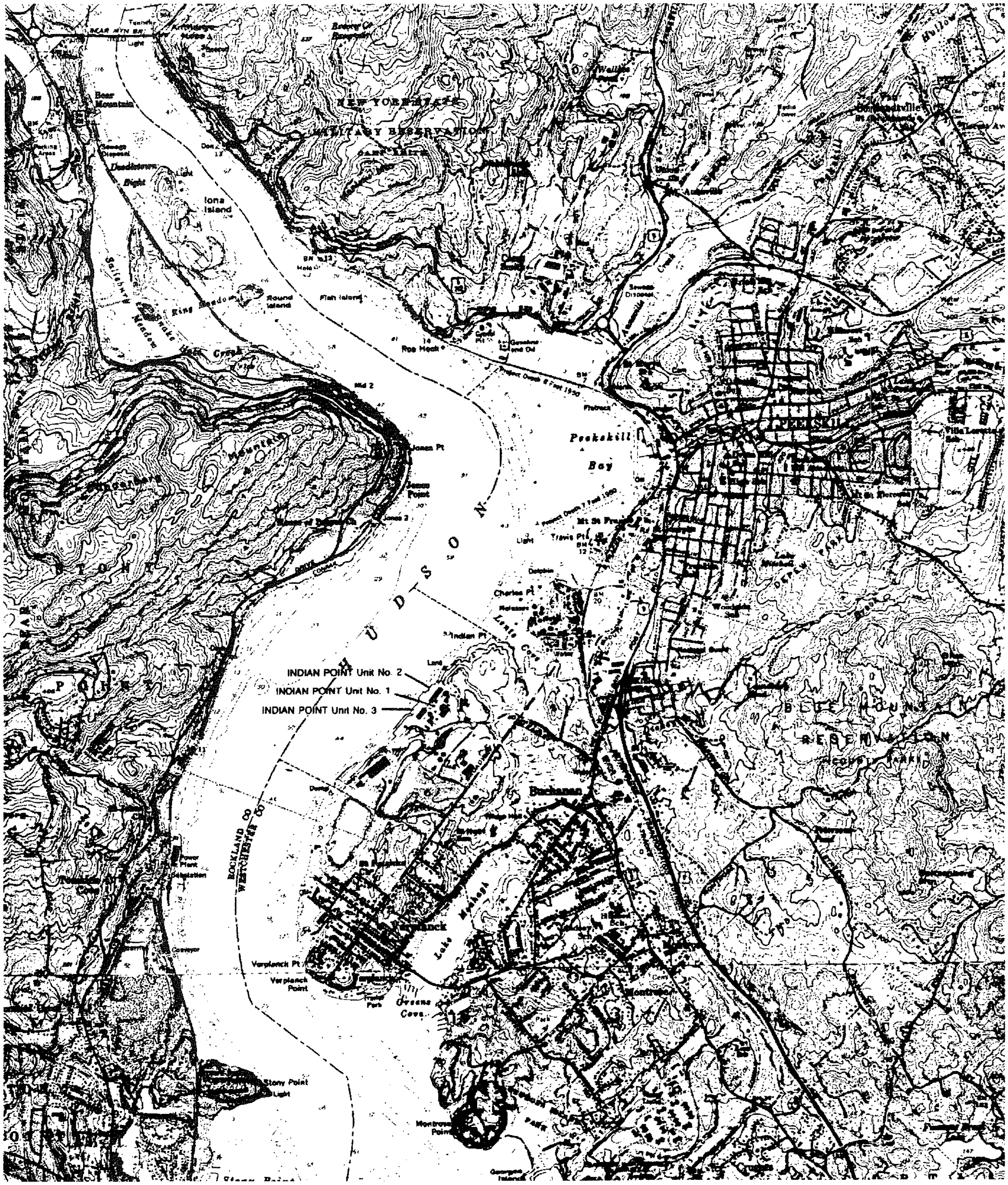


INDIAN POINT UNIT No. 2

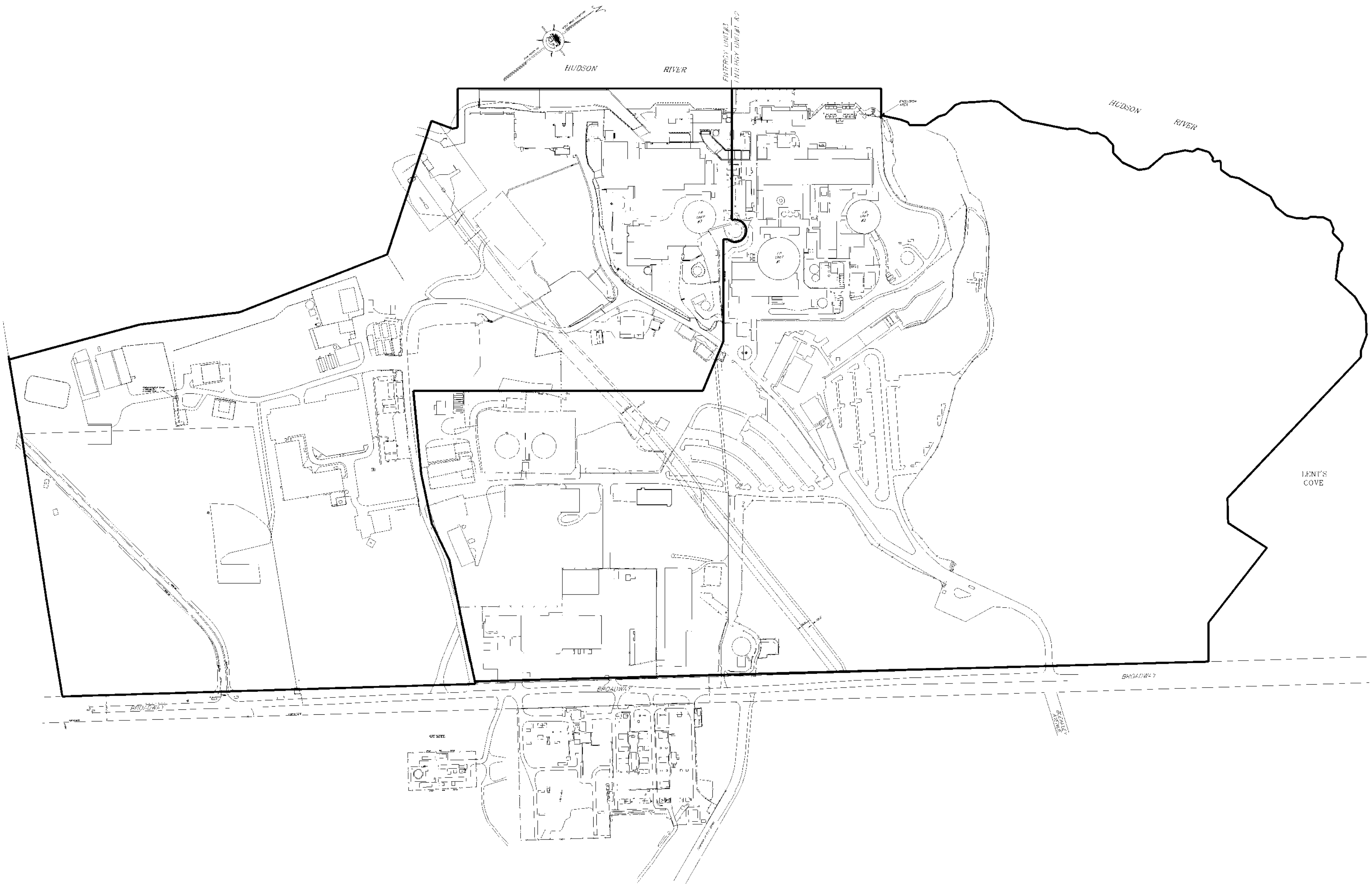
UFSAR FIGURE 2.4-1

SCHEMATIC SECTOR/ZONE DIAGRAM

MIC. No. 1999MC3575	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 2.3-1	
TOPOGRAPHICAL MAP OF INDIAN POINT AND SURROUNDING AREA	
MIC. No. 1999MC3574	REV. No. 17A



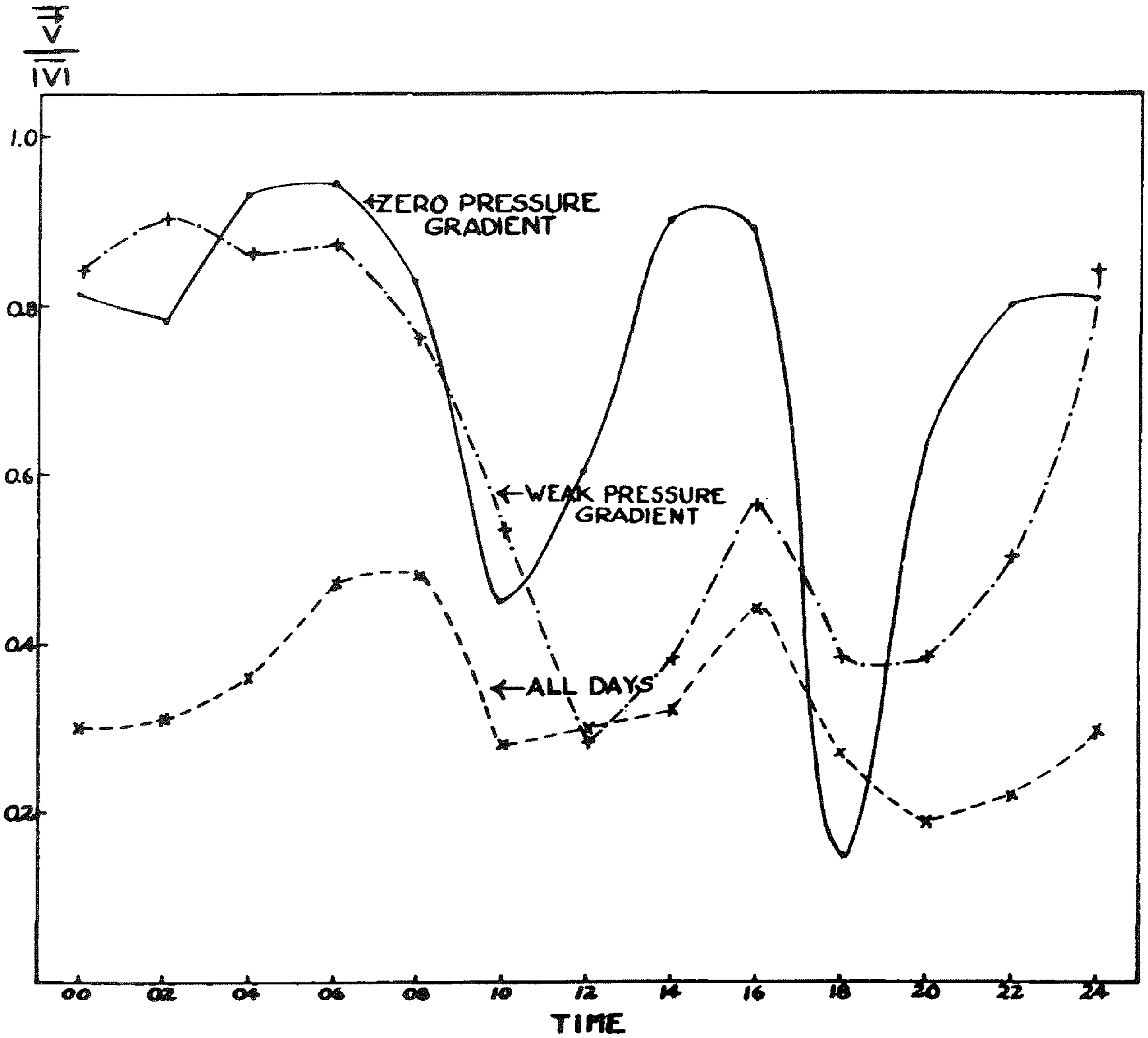
NOT TO SCALE

INDIAN POINT UNIT No. 2

UFSAR FIGURE 2.2-2

INDIAN POINT BUILDING IDENTIFICATION

MIC. No. 1999MC3573 | REV. No. 17B



INDIAN POINT UNIT No. 2

UFSAR FIGURE 2.6-3

STEADINESS OF WIND AS A FUNCTION OF TIME OF DAY FOR INDICATED PRESSURE GRADIENT CONDITIONS

MIC. No. 2001MB1194	REV. No. 17A

GEOLOGIC HISTORY IN THE CROTON FALLS AREA
(AFTER BROCK & MOSE, 1979)

ACADIAN	(F_6 = tight to isoclinal fold; deformation of Goldens Bridge granite) (Goldens Bridge granite) Joints with muscovite and tourmaline ? open folds F_5 = folded isograd. S_5: retrograde muscovite schistosity; associated retrograde kyanite and staurolite; locally associated shearing and faulting. (Croton Falls granites intruded 387 $\pm$ 37 m.y. ago) (Croton Falls ultramafic complex; includes xenoliths of refolded schists; could b synchronous with F_3 folding). ? Open folds
TACONIC	Quartz $\pm$ garnet $\pm$ sillimanite, and quartz plus two feldspar + garnet + sillimanite segregations. F_4 = locally developed reclined close folds and chevron folds. Faint S_4: realigned biotite and sillimanite in F_3 segregations and host rocks. Sillimanite K-feldspar facies. F_3 = very common near-isoclinal folds; refolds S_1 schistosity and F_2 flasers. Variable S_3: slight re-alignment of sillimanite and biotite. Quartz-K-feldspar - sillimanite segregations. Sillimanite-K-feldspar facies. F_2 = isoclinal to near-isoclinal intrafolial folds. Strong cataclastic S_2: flasers of quartz-feldspar segregations and plagioclase, smeared garnets, partial realignment of biotite and sillimanite. Quartz plagioclase segregations. F_1 = a few isoclinal folds. Strong S_1: alignment of biotite, sillimanite, and quartz $\pm$ plagioclase segregations. Porphyroblastic garnets $\pm$ plagioclase.
	Manhattan Schist deposition Inwood Marble deposition Lowerre Quartzite deposition unconformity (Intrusion of Yonkers and Pound Ridge granite gneisses at 575-600 m.y.)
GRENVILLE	Fordham Gneiss (Hudson Highlands gneiss and related rocks metamorphosed at 0.9-1.2 b.y. and still contain granulite facies relicts.)

Note: All features or events except those in parenthesis are demonstrable in one roadcut about 1 km south of the Croton Falls Complex on Route 684.

TABLE 3

YORK SERVICES CORPORATION JOB NUMBER: 01-4122-00
ONE RESEARCH DRIVE, STAMFORD, CONNECTICUT 06906

CLIENT: CON EDISON CO. OF N.Y.
LOCATION: INDIAN POINT (10M)

TABLE 30

DIURNAL VARIATION OF STABILITY CLASS AND WIND SPEED
(CONCURRENT DATA)

MAY 1, 79,80 - OCT 31,79,80 SUMMER

PASQUILL STABILITY CATEGORY

MEAN SPEED (MPH)/NUMBER OF OBSERVATIONS

HOUR	A	B	C	D	E	F	G
100	5.0 / 2	0.0 / 0	0.0 / 0	4.1 / 49	2.7 / 197	2.2 / 94	2.4 / 15
200	0.0 / 0	0.0 / 0	0.0 / 0	4.0 / 62	2.7 / 186	2.2 / 98	2.2 / 11
300	0.0 / 0	0.0 / 0	0.0 / 0	4.0 / 73	2.7 / 175	2.1 / 97	2.5 / 12
400	0.0 / 0	0.0 / 0	0.0 / 0	4.1 / 62	2.6 / 187	2.5 / 97	2.1 / 12
500	0.0 / 0	0.0 / 0	0.0 / 0	3.7 / 62	2.6 / 195	2.3 / 89	3.2 / 8
600	0.0 / 0	0.0 / 0	4.0 / 2	3.4 / 123	2.5 / 180	2.2 / 43	3.4 / 5
700	2.7 / 13	3.2 / 28	3.3 / 29	3.2 / 157	2.5 / 107	2.1 / 18	1.0 / 1
800	3.5 / 68	3.2 / 43	3.2 / 44	3.1 / 152	2.2 / 46	4.7 / 3	0.0 / 0
900	3.6 / 142	3.5 / 42	3.2 / 41	2.9 / 105	3.4 / 27	0.0 / 0	0.0 / 0
1000	3.8 / 197	3.4 / 35	2.9 / 32	3.1 / 75	3.1 / 17	0.0 / 0	0.0 / 0
1100	4.0 / 233	3.2 / 17	3.7 / 28	3.5 / 64	3.0 / 18	0.0 / 0	0.0 / 0
1200	4.1 / 238	3.5 / 23	3.6 / 24	3.7 / 57	3.8 / 16	0.0 / 0	0.0 / 0
1300	4.3 / 226	3.7 / 29	3.8 / 23	3.8 / 62	4.8 / 16	3.5 / 1	0.0 / 0
1400	4.3 / 204	4.3 / 24	3.9 / 23	3.8 / 87	3.7 / 19	2.5 / 1	0.0 / 0
1500	4.1 / 159	4.4 / 32	3.8 / 35	3.9 / 100	3.7 / 28	4.8 / 2	4.0 / 1
1600	4.0 / 111	3.7 / 29	4.4 / 34	3.8 / 142	3.2 / 36	3.0 / 2	0.0 / 0
1700	3.7 / 55	3.6 / 26	3.6 / 26	3.9 / 154	3.0 / 88	2.5 / 8	0.0 / 0
1800	2.8 / 5	3.6 / 7	3.2 / 19	3.7 / 146	2.7 / 158	1.2 / 22	0.0 / 0
1900	0.0 / 0	0.0 / 0	0.0 / 0	3.8 / 75	2.5 / 227	1.7 / 55	1.0 / 1
2000	1.5 / 1	0.0 / 0	0.0 / 0	4.3 / 45	2.6 / 226	1.7 / 79	1.5 / 6
2100	0.0 / 0	0.0 / 0	0.0 / 0	4.3 / 42	2.7 / 216	2.1 / 92	1.9 / 7
2200	6.5 / 1	2.0 / 1	0.0 / 0	4.4 / 47	2.7 / 208	1.8 / 84	2.1 / 18
2300	0.0 / 0	0.0 / 0	0.0 / 0	4.0 / 55	2.9 / 182	1.9 / 106	1.7 / 14
2400	0.0 / 0	0.0 / 0	2.5 / 1	3.9 / 55	2.7 / 188	2.1 / 98	2.4 / 16
TOTAL	4.0 /1655	3.6 / 336	3.5 / 361	3.7 /2050	2.7 /2939	2.1 /1089	2.2 / 127

NOV 1, 79,80 - APR 30,79,80 WINTER

PASQUILL STABILITY CATEGORY

MEAN SPEED (MPH)/NUMBER OF OBSERVATIONS

HOUR	A	B	C	D	E	F	G
100	10.0 / 1	0.0 / 0	21.0 / 1	6.1 / 112	4.1 / 187	2.1 / 47	2.8 / 12
200	11.5 / 1	0.0 / 0	0.0 / 0	6.5 / 111	3.9 / 191	2.0 / 44	3.0 / 14
300	0.0 / 0	0.0 / 0	0.0 / 0	6.6 / 109	4.0 / 191	2.0 / 48	2.8 / 13
400	2.0 / 1	0.0 / 0	7.5 / 2	6.6 / 107	3.9 / 194	1.9 / 40	2.7 / 16
500	3.0 / 2	0.0 / 0	8.0 / 1	6.4 / 108	3.7 / 194	2.4 / 44	2.4 / 11
600	2.8 / 2	0.0 / 0	0.0 / 0	6.2 / 115	3.6 / 198	2.3 / 35	3.0 / 11
700	0.0 / 0	0.0 / 0	6.0 / 5	6.2 / 134	3.6 / 183	2.5 / 33	2.6 / 7
800	6.3 / 5	6.2 / 10	4.5 / 14	5.5 / 158	3.7 / 152	2.0 / 18	2.7 / 5
900	5.5 / 21	5.7 / 25	6.2 / 38	5.3 / 181	3.4 / 89	2.8 / 6	2.0 / 2
1000	5.8 / 62	7.0 / 35	5.4 / 56	5.1 / 141	3.6 / 61	4.2 / 3	2.0 / 2
1100	5.8 / 74	6.3 / 47	5.2 / 38	5.3 / 128	3.7 / 48	3.3 / 3	4.0 / 2
1200	6.0 / 115	6.2 / 37	6.1 / 47	5.2 / 116	4.1 / 39	5.1 / 6	5.0 / 1
1300	6.1 / 97	7.0 / 56	6.0 / 46	5.5 / 108	4.0 / 48	3.3 / 4	5.0 / 2
1400	5.9 / 82	6.6 / 44	6.2 / 57	5.7 / 114	4.2 / 59	4.0 / 3	6.5 / 2
1500	5.6 / 43	5.8 / 31	6.6 / 42	5.9 / 176	4.1 / 62	2.9 / 5	6.5 / 1
1600	5.2 / 20	5.6 / 12	5.3 / 15	5.8 / 222	4.3 / 81	2.8 / 6	3.5 / 3
1700	5.1 / 8	3.7 / 2	6.9 / 10	5.9 / 173	4.2 / 150	2.2 / 14	2.3 / 2
1800	0.0 / 0	0.0 / 0	11.0 / 1	6.1 / 135	4.0 / 198	2.0 / 18	2.1 / 7
1900	0.0 / 0	0.0 / 0	0.0 / 0	6.3 / 106	4.3 / 210	2.1 / 41	3.6 / 4
2000	4.5 / 2	0.0 / 0	0.0 / 0	6.3 / 105	4.2 / 207	1.9 / 40	1.9 / 6
2100	0.0 / 0	4.5 / 1	0.0 / 0	6.0 / 111	4.2 / 197	2.1 / 43	2.5 / 8
2200	0.0 / 0	0.0 / 0	6.0 / 1	6.0 / 102	4.3 / 197	2.1 / 45	2.6 / 14
2300	0.0 / 0	0.0 / 0	8.0 / 1	6.1 / 109	4.0 / 191	2.0 / 46	2.5 / 12
2400	12.0 / 1	0.0 / 0	10.3 / 3	6.0 / 114	4.0 / 180	1.8 / 53	2.8 / 9
TOTAL	5.9 / 557	6.4 / 300	6.0 / 378	5.9 /3095	4.0 /3507	2.2 / 645	2.8 / 166

IONA SERVICES CORPORATION — JOB NUMBER: 01-4122-00
ONE RESEARCH DRIVE, STAMFORD, CONNECTICUT 06906

CLIENT: CON EDISON CO. OF N.Y.
LOCATION: INDIAN POINT (122M)

TABLE 31

DIURNAL VARIATION OF STABILITY CLASS AND WIND SPEED (CONCURRENT DATA)

MAY 1, 79,80 - OCT 31,79,80 — SUMMER

PASQUILL STABILITY CATEGORY

MEAN SPEED (MPH)/NUMBER OF OBSERVATIONS

HOUR	A	B	C	D	E	F	G
100	0.0 / 0	0.0 / 0	0.0 / 0	11.8 / 86	6.8 / 190	4.4 / 48	5.3 / 2
200	0.0 / 0	0.0 / 0	0.0 / 0	10.9 / 93	6.8 / 176	4.2 / 56	0.0 / 0
300	0.0 / 0	0.0 / 0	0.0 / 0	10.8 / 96	6.5 / 177	4.5 / 51	3.0 / 1
400	0.0 / 0	0.0 / 0	0.0 / 0	10.9 / 93	6.3 / 182	4.7 / 49	3.5 / 2
500	0.0 / 0	0.0 / 0	0.0 / 0	11.0 / 92	6.0 / 188	4.0 / 40	4.0 / 3
600	0.0 / 0	0.0 / 0	0.0 / 0	10.4 / 123	5.6 / 174	4.5 / 26	4.0 / 1
700	0.0 / 0	9.3 / 4	8.9 / 5	8.7 / 194	5.2 / 114	3.7 / 8	0.0 / 0
800	11.3 / 6	10.6 / 9	8.0 / 23	7.2 / 235	5.3 / 53	2.0 / 1	0.0 / 0
900	11.7 / 13	9.4 / 27	7.6 / 47	7.2 / 211	7.0 / 31	0.0 / 0	0.0 / 0
1000	10.6 / 32	9.0 / 63	9.6 / 52	6.7 / 167	7.6 / 18	0.0 / 0	0.0 / 0
1100	11.0 / 59	9.9 / 58	9.0 / 49	7.7 / 149	6.8 / 17	0.0 / 0	0.0 / 0
1200	10.7 / 78	9.7 / 51	9.4 / 60	9.1 / 128	10.8 / 11	12.0 / 1	0.0 / 0
1300	11.1 / 59	10.2 / 65	10.4 / 50	9.9 / 140	11.6 / 14	0.0 / 0	0.0 / 0
1400	11.6 / 55	11.1 / 68	10.7 / 46	9.7 / 147	12.0 / 14	0.0 / 0	0.0 / 0
1500	11.4 / 36	12.5 / 47	10.8 / 51	10.5 / 179	10.7 / 14	10.0 / 1	0.0 / 0
1600	13.3 / 16	11.9 / 28	11.1 / 45	11.0 / 210	10.7 / 26	0.0 / 0	0.0 / 0
1700	10.3 / 2	12.3 / 10	11.7 / 23	11.3 / 245	11.6 / 42	14.0 / 2	0.0 / 0
1800	0.0 / 0	15.0 / 1	10.7 / 3	11.3 / 198	10.1 / 122	6.0 / 1	0.0 / 0
1900	0.0 / 0	0.0 / 0	0.0 / 0	11.7 / 118	9.7 / 199	4.9 / 7	0.0 / 0
2000	0.0 / 0	0.0 / 0	0.0 / 0	12.3 / 86	9.5 / 217	5.5 / 21	0.0 / 0
2100	0.0 / 0	0.0 / 0	0.0 / 0	12.5 / 88	8.8 / 205	4.8 / 34	0.0 / 0
2200	0.0 / 0	0.0 / 0	0.0 / 0	12.2 / 100	7.8 / 183	5.4 / 43	3.5 / 1
2300	0.0 / 0	0.0 / 0	8.0 / 1	11.6 / 89	7.7 / 187	5.5 / 49	0.0 / 0
2400	0.0 / 0	0.0 / 0	0.0 / 0	11.4 / 91	7.2 / 185	5.0 / 50	7.0 / 1
TOTAL	11.2 / 356	10.4 / 431	9.8 / 455	10.0 /3358	7.6 /2739	4.8 / 488	4.3 / 11

NOV 1, 79,80 - APR 30,79,80 — WINTER

PASQUILL STABILITY CATEGORY

MEAN SPEED (MPH)/NUMBER OF OBSERVATIONS

HOUR	A	B	C	D	E	F	G
100	0.0 / 0	0.0 / 0	0.0 / 0	13.9 / 172	8.6 / 155	4.1 / 25	10.4 / 4
200	0.0 / 0	0.0 / 0	0.0 / 0	13.5 / 174	8.5 / 148	5.3 / 32	8.0 / 3
300	0.0 / 0	0.0 / 0	0.0 / 0	13.7 / 164	8.6 / 160	4.9 / 30	8.3 / 3
400	0.0 / 0	0.0 / 0	0.0 / 0	13.2 / 167	8.4 / 155	4.4 / 33	10.8 / 2
500	0.0 / 0	0.0 / 0	0.0 / 0	13.4 / 163	7.8 / 163	4.1 / 28	10.0 / 2
600	0.0 / 0	0.0 / 0	0.0 / 0	13.4 / 164	7.0 / 164	4.9 / 24	6.9 / 4
700	0.0 / 0	0.0 / 0	0.0 / 0	13.5 / 167	6.9 / 166	5.1 / 21	4.0 / 1
800	0.0 / 0	0.0 / 0	14.5 / 4	12.7 / 195	7.3 / 141	3.8 / 15	0.0 / 0
900	0.0 / 0	10.7 / 5	13.0 / 12	12.0 / 241	6.5 / 93	7.9 / 6	0.0 / 0
1000	14.0 / 1	14.3 / 7	11.2 / 25	11.3 / 260	7.1 / 62	6.8 / 3	0.0 / 0
1100	11.7 / 8	10.8 / 19	13.4 / 32	11.6 / 243	7.8 / 57	8.5 / 2	0.0 / 0
1200	14.2 / 15	13.4 / 24	13.7 / 45	11.8 / 220	9.3 / 54	10.0 / 3	0.0 / 0
1300	15.5 / 12	12.6 / 24	14.1 / 44	12.2 / 227	10.3 / 52	8.0 / 1	7.0 / 1
1400	15.6 / 9	12.9 / 14	15.7 / 41	12.4 / 241	10.0 / 55	21.0 / 1	12.0 / 1
1500	13.0 / 2	10.6 / 5	13.6 / 26	13.2 / 270	10.3 / 56	13.0 / 2	0.0 / 0
1600	12.8 / 3	11.3 / 2	15.6 / 8	13.1 / 278	10.4 / 65	10.3 / 4	0.0 / 0
1700	0.0 / 0	0.0 / 0	20.5 / 3	13.2 / 239	10.5 / 112	8.4 / 5	0.0 / 0
1800	0.0 / 0	0.0 / 0	0.0 / 0	13.7 / 201	9.5 / 148	10.5 / 10	17.0 / 1
1900	0.0 / 0	0.0 / 0	0.0 / 0	13.6 / 172	9.8 / 178	10.4 / 10	0.0 / 0
2000	7.5 / 1	0.0 / 0	0.0 / 0	13.6 / 175	9.6 / 167	6.7 / 16	0.0 / 0
2100	0.0 / 0	0.0 / 0	0.0 / 0	13.2 / 174	9.5 / 162	6.0 / 22	0.0 / 0
2200	0.0 / 0	13.0 / 1	0.0 / 0	12.9 / 168	9.1 / 161	5.6 / 25	4.5 / 1
2300	8.0 / 1	16.0 / 1	0.0 / 0	13.1 / 166	8.5 / 160	5.3 / 25	2.0 / 1
2400	0.0 / 0	0.0 / 0	0.0 / 0	13.1 / 176	8.5 / 152	4.8 / 26	8.8 / 3
TOTAL	13.9 / 51	12.4 / 102	13.9 / 240	12.9 /4819	8.6 /2986	5.6 / 369	8.6 / 27

CLIENT: CON EDISON CO. OF N.Y.
LOCATION: INDIAN POINT (DELTA-T 122M-60M)

TABLE 32

DIURNAL VARIATION OF STABILITY CLASS AND WIND SPEED (CONCURRENT DATA)

MAY 1, 79,80 - OCT 31,79,80

SUMMER

PASQUILL STABILITY CATEGORY

MEAN SPEED (MPH)/NUMBER OF OBSERVATIONS

HOUR	A	B	C	D	E	F	G
100	0.0 / 0	0.0 / 0	18.0 / 1	9.8 / 160	5.8 / 138	5.6 / 19	1.6 / 2
200	0.0 / 0	0.0 / 0	15.3 / 2	9.9 / 161	5.2 / 137	4.2 / 23	2.0 / 2
300	0.0 / 0	0.0 / 0	17.5 / 1	9.5 / 171	5.4 / 126	4.3 / 24	2.3 / 3
400	0.0 / 0	0.0 / 0	18.0 / 1	9.2 / 169	5.6 / 135	3.6 / 19	4.3 / 2
500	0.0 / 0	0.0 / 0	0.0 / 0	9.3 / 167	5.1 / 132	3.3 / 19	4.9 / 5
600	0.0 / 0	0.0 / 0	14.3 / 3	9.5 / 166	5.1 / 122	4.0 / 28	4.6 / 5
700	6.5 / 1	2.0 / 1	10.0 / 3	9.0 / 202	4.6 / 105	2.8 / 12	3.0 / 1
800	13.3 / 3	0.0 / 0	12.8 / 5	7.7 / 260	4.0 / 52	2.7 / 7	0.0 / 0
900	7.0 / 1	15.3 / 2	9.0 / 6	8.0 / 267	5.3 / 52	3.0 / 1	0.0 / 0
1000	0.0 / 0	0.0 / 0	11.7 / 12	8.5 / 277	4.0 / 43	0.0 / 0	0.0 / 0
1100	0.0 / 0	14.5 / 2	11.7 / 3	9.5 / 277	5.2 / 50	0.0 / 0	0.0 / 0
1200	0.0 / 0	14.0 / 2	11.7 / 9	10.0 / 281	6.4 / 35	4.5 / 1	12.0 / 1
1300	13.0 / 1	11.5 / 3	9.6 / 4	10.7 / 282	7.7 / 38	0.0 / 0	0.0 / 0
1400	0.0 / 0	13.3 / 6	11.9 / 20	10.9 / 265	7.3 / 39	0.0 / 0	0.0 / 0
1500	5.5 / 2	13.0 / 9	11.4 / 25	11.1 / 254	9.1 / 38	0.0 / 0	0.0 / 0
1600	0.0 / 0	9.9 / 4	13.3 / 12	11.3 / 271	9.6 / 38	0.0 / 0	0.0 / 0
1700	9.0 / 1	10.0 / 1	12.8 / 5	11.5 / 282	10.3 / 33	10.5 / 2	0.0 / 0
1800	5.0 / 1	15.0 / 1	0.0 / 0	11.2 / 259	9.6 / 64	0.0 / 0	0.0 / 0
1900	12.0 / 1	0.0 / 0	0.0 / 0	10.8 / 228	9.2 / 94	2.0 / 1	0.0 / 0
2000	0.0 / 0	0.0 / 0	0.0 / 0	10.9 / 196	8.6 / 122	6.9 / 6	0.0 / 0
2100	4.0 / 1	26.0 / 1	0.0 / 0	10.9 / 194	7.3 / 120	4.1 / 11	0.0 / 0
2200	13.0 / 1	0.0 / 0	23.0 / 1	10.1 / 193	7.0 / 117	5.7 / 13	3.5 / 2
2300	0.0 / 0	4.5 / 1	13.0 / 2	10.1 / 173	6.6 / 136	4.4 / 14	0.0 / 0
2400	0.0 / 0	0.0 / 0	15.5 / 2	9.8 / 176	6.2 / 128	4.3 / 21	0.0 / 0
TOTAL	9.3 / 13	12.6 / 33	12.1 / 117	10.0 /5337	6.4 /2094	4.3 / 221	4.0 / 23

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

WINTER

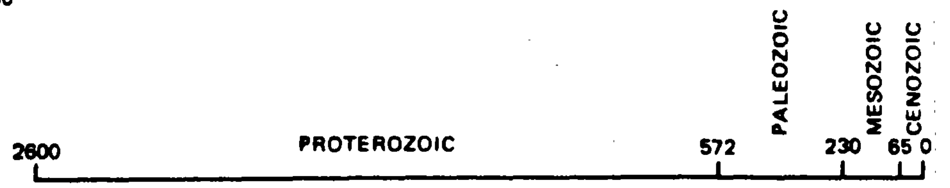
PASQUILL STABILITY CATEGORY

MEAN SPEED (MPH)/NUMBER OF OBSERVATIONS

HOUR	A	B	C	D	E	F	G
100	5.0 / 1	0.0 / 0	0.0 / 0	12.4 / 244	7.3 / 97	10.0 / 13	4.0 / 1
200	0.0 / 0	0.0 / 0	9.0 / 1	12.3 / 235	7.4 / 108	8.8 / 11	4.0 / 2
300	0.0 / 0	0.0 / 0	0.0 / 0	12.7 / 230	7.0 / 114	6.8 / 11	2.5 / 2
400	0.0 / 0	0.0 / 0	0.0 / 0	12.4 / 228	6.8 / 116	4.5 / 12	1.5 / 1
500	7.0 / 1	0.0 / 0	0.0 / 0	12.2 / 236	5.9 / 105	5.4 / 14	0.0 / 0
600	0.0 / 0	8.0 / 1	11.0 / 1	11.7 / 235	6.2 / 105	5.5 / 12	4.0 / 2
700	0.0 / 0	0.0 / 0	0.0 / 0	12.1 / 225	6.2 / 117	4.6 / 13	0.0 / 0
800	10.0 / 1	0.0 / 0	19.5 / 2	12.5 / 224	6.3 / 118	3.7 / 10	0.0 / 0
900	8.5 / 1	5.0 / 1	33.0 / 1	12.2 / 258	6.0 / 91	4.7 / 5	0.0 / 0
1000	0.0 / 0	30.0 / 1	11.5 / 2	11.6 / 283	6.7 / 69	2.5 / 3	0.0 / 0
1100	12.0 / 1	11.5 / 1	31.5 / 2	11.8 / 292	7.4 / 61	3.6 / 4	0.0 / 0
1200	10.5 / 2	0.0 / 0	27.7 / 3	12.3 / 296	8.6 / 59	15.0 / 1	0.0 / 0
1300	14.6 / 4	30.0 / 1	23.8 / 5	12.5 / 299	9.7 / 51	8.0 / 1	0.0 / 0
1400	11.0 / 2	30.6 / 2	19.0 / 1	12.8 / 298	10.4 / 55	10.2 / 3	0.0 / 0
1500	9.3 / 3	26.5 / 2	21.1 / 7	13.0 / 290	10.0 / 56	9.5 / 2	18.0 / 1
1600	4.0 / 1	0.0 / 0	35.0 / 1	13.1 / 294	10.4 / 61	8.8 / 3	0.0 / 0
1700	4.5 / 1	0.0 / 0	22.0 / 1	12.7 / 285	10.8 / 72	0.0 / 0	0.0 / 0
1800	0.0 / 0	0.0 / 0	6.0 / 1	12.5 / 269	10.0 / 84	11.5 / 6	0.0 / 0
1900	0.0 / 0	0.0 / 0	0.0 / 0	12.3 / 262	9.7 / 97	31.0 / 1	0.0 / 0
2000	0.0 / 0	26.0 / 1	10.0 / 1	12.3 / 253	8.7 / 99	17.6 / 5	0.0 / 0
2100	13.0 / 1	0.0 / 0	12.5 / 1	12.3 / 240	8.6 / 110	9.5 / 6	0.0 / 0
2200	8.5 / 2	11.0 / 1	8.0 / 1	11.9 / 239	8.1 / 108	7.5 / 5	0.0 / 0
2300	12.0 / 2	0.0 / 0	13.0 / 1	12.1 / 234	7.0 / 115	9.9 / 4	0.0 / 0
2400	4.0 / 1	16.0 / 1	0.0 / 0	12.2 / 229	7.6 / 111	5.5 / 13	12.3 / 2
TOTAL	9.9 / 24	21.0 / 12	20.4 / 32	12.3 /6178	7.8 /2179	7.2 / 158	6.3 / 11

GEOLOGIC TIME SCALE

ABSOLUTE AGE (In millions of years Before Present)	ERA		PERIOD		EPOCH	SIGNIFICANT EVENTS
0.01		CENOZOIC	QUATERNARY		HOLOCENE (RECENT)	
1.8		CENOZOIC	QUATERNARY		PLEISTOCENE	WISCONSINAN GLACIATION
		CENOZOIC	TERTIARY	NEOGENE	PLIOCENE	
22.5		CENOZOIC	TERTIARY	NEOGENE	MIOCENE	
		CENOZOIC	TERTIARY	PALEOGENE	OLIGOCENE	
		CENOZOIC	TERTIARY	PALEOGENE	EOCENE	
65		CENOZOIC	TERTIARY	PALEOGENE	PALEOCENE	
141		MESOZOIC	CRETACEOUS			
195		MESOZOIC	JURASSIC			OPENING OF ATLANTIC
230		MESOZOIC	TRIASSIC			BREAK UP OF PANGEA
280		PALEOZOIC	PERMIAN			ALLEGHENIAN OROGENY
		PALEOZOIC	Carboniferous	PENNSYLVANIAN		
345		PALEOZOIC	Carboniferous	MISSISSIPPIAN		
395		PALEOZOIC	DEVONIAN			ACADIAN OROGENY
435		PALEOZOIC	SILURIAN			
502		PALEOZOIC	ORDOVICIAN			TACONIC OROGENY
572		PALEOZOIC	CAMBRIAN			
1000	PRECAMBRIAN	PROTEROZOIC	HADRYNIAN			OPENING OF IAPETUS
1800	PRECAMBRIAN	PROTEROZOIC	HELIKIAN			GRENVILLIAN OROGENY
2600	PRECAMBRIAN	PROTEROZOIC	APHEBIAN			
4500	PRECAMBRIAN		ARCHEAN			



BAR SCALE
(In millions of years Before Present)

(After: F.W.B. van Eysinga, 1978)

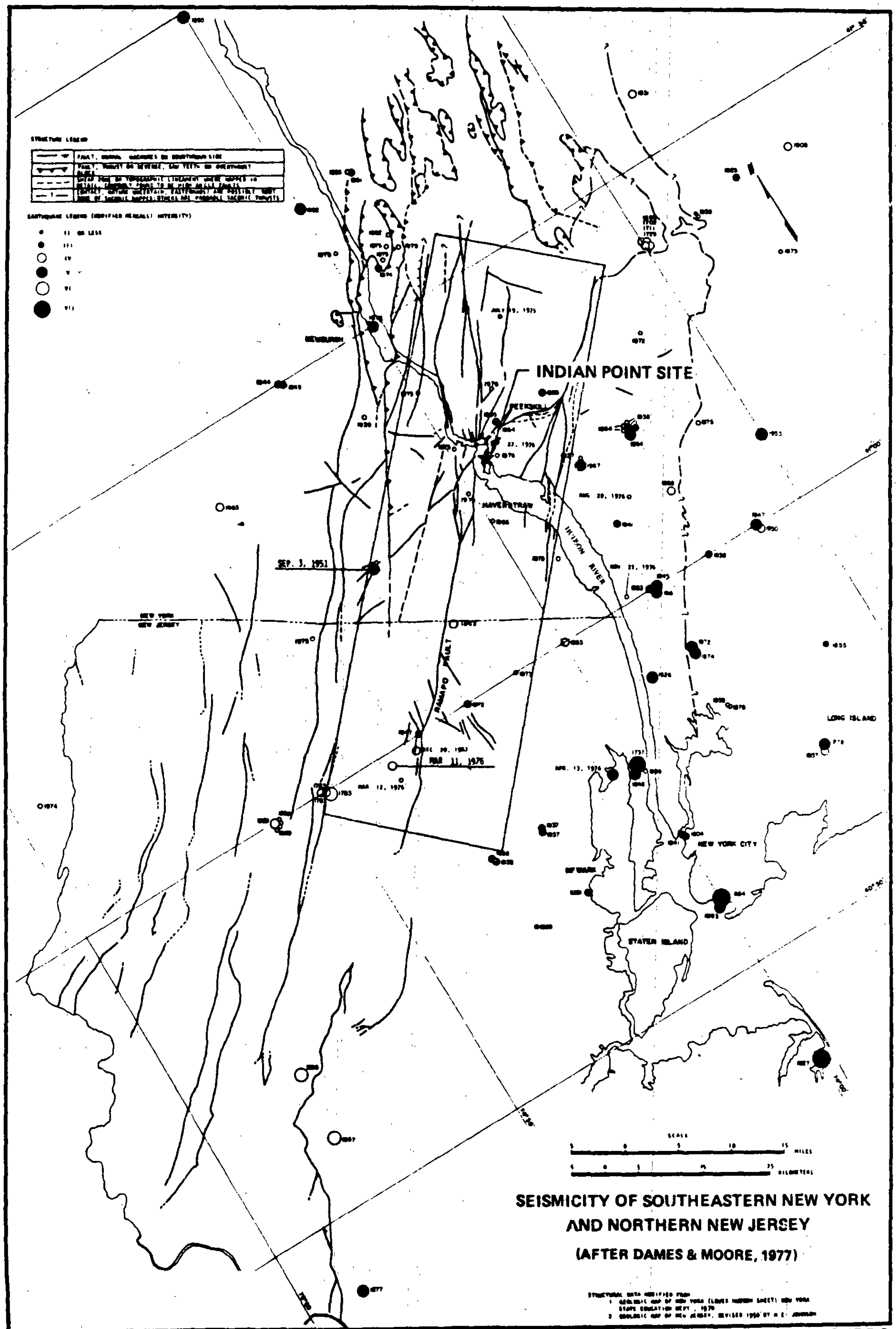
TABLE 1

AGE \ LOCATION	MANHATTAN PRONG THIS REPORT		DUTCHESS COUNTY, NEW YORK FISHER, (1962a, 1962b) KNOPF, (1962)		WESTERN MASSACHUSETTS AND CONNECTICUT ZEN AND HARTSHORN, (1966) ZEN, (1967)		WESTERN VERMONT DOLL AND OTHERS, (1961) ZEN, (1961)
MIDDLE ORDOVICIAN	MANHATTAN SCHIST	C-MEMBER B-MEMBER A-MEMBER INTERBEDDED MARBLE AND SCHIST	WALLOOMSAC FORMATION BALMVILLE LIMESTONE		WALLOOMSAC FORMATION INTERBEDDED LIMESTONE AND SCHIST		IRA FORMATION WHIPPLE MARBLE MEMBER
EARLY ORDOVICIAN	INWOOD MARBLE	E-MEMBER	STOCKBRIDGE GROUP = WAPPINGERS GROUP	COPAKE LIMESTONE ROCHDALE LIMESTONE	STOCKBRIDGE FORMATION	UNIT-G	CHIPMAN FORMATION BASCOM FORMATION
		D-MEMBER		HALCYON LAKE		UNIT-F UNIT-E UNIT-D	CUTTING DOLOMITE SHELBURNE FORMATION
CAMBRIAN		C-MEMBER		BRIARCLIFF DOLOMITE		UNIT-C	CLARENDON SPRINGS DOLOMITE DANBY FORMATION
		B-MEMBER		PINE PLAINS FORMATION		UNIT-B	WINOOSKI DOLOMITE MONKTON QUARTZITE
		A-MEMBER		STISSING DOLOMITE		UNIT-A	DUNHAM DOLOMITE
	LOWERRE QUARTZITE		POUGHQUAG QUARTZITE		CHESHIRE QUARTZITE DALTON FORMATION		CHESHIRE QUARTZITE DALTON FORMATION
PRECAMBRIAN	FORDHAM GNEISS YONKERS GNEISS		GNEISS		BERKSHIRE MASSIF		MOUNT HOLLY COMPLEX

PROPOSED CORRELATION OF THE STRATIGRAPHIC SUBDIVISIONS IN THE MANHATTAN PRONG WITH ROCKS IN ADJACENT AREAS

(AFTER HALL, 1968)

TABLE 2



SEISMICITY OF SOUTHEASTERN NEW YORK AND NORTHERN NEW JERSEY

(AFTER DAMES & MOORE, 1977)

FIGURE 2

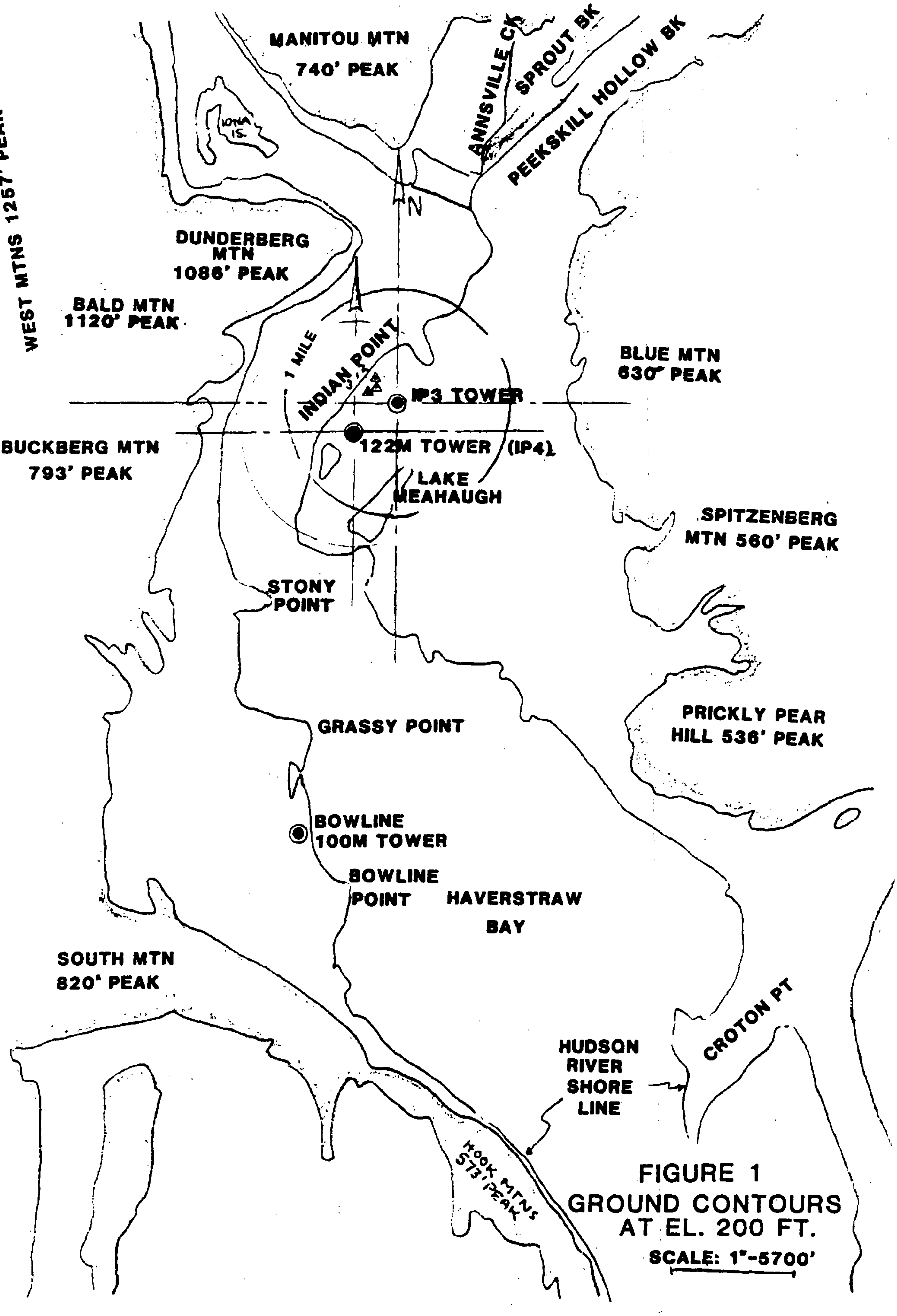


FIGURE 1
GROUND CONTOURS
AT EL. 200 FT.

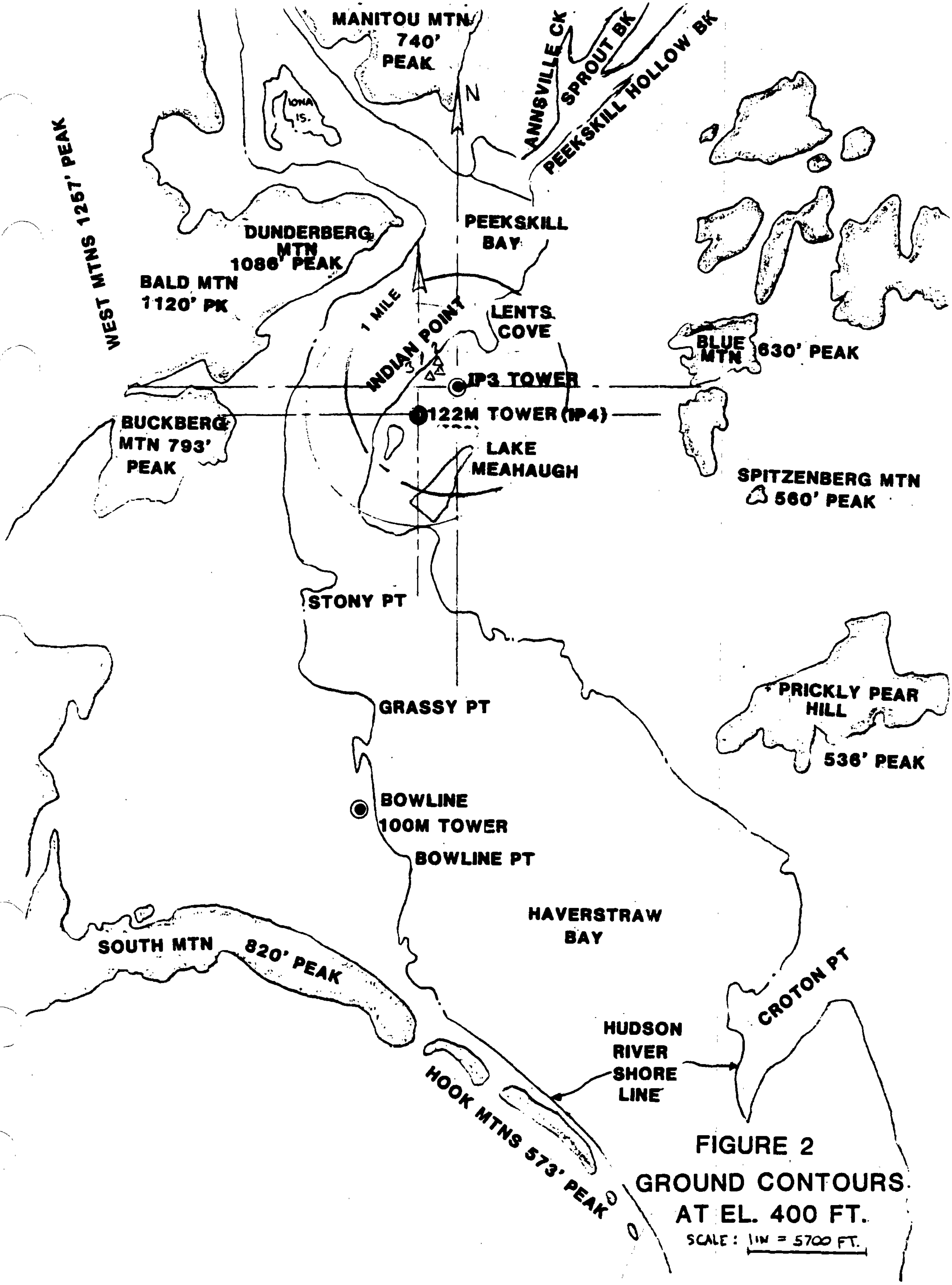


FIGURE 2
GROUND CONTOURS
AT EL. 400 FT.
SCALE: 1 IN = 5700 FT.

FIGURE 3

ELEVATIONS IN THE INDIAN POINT REGION

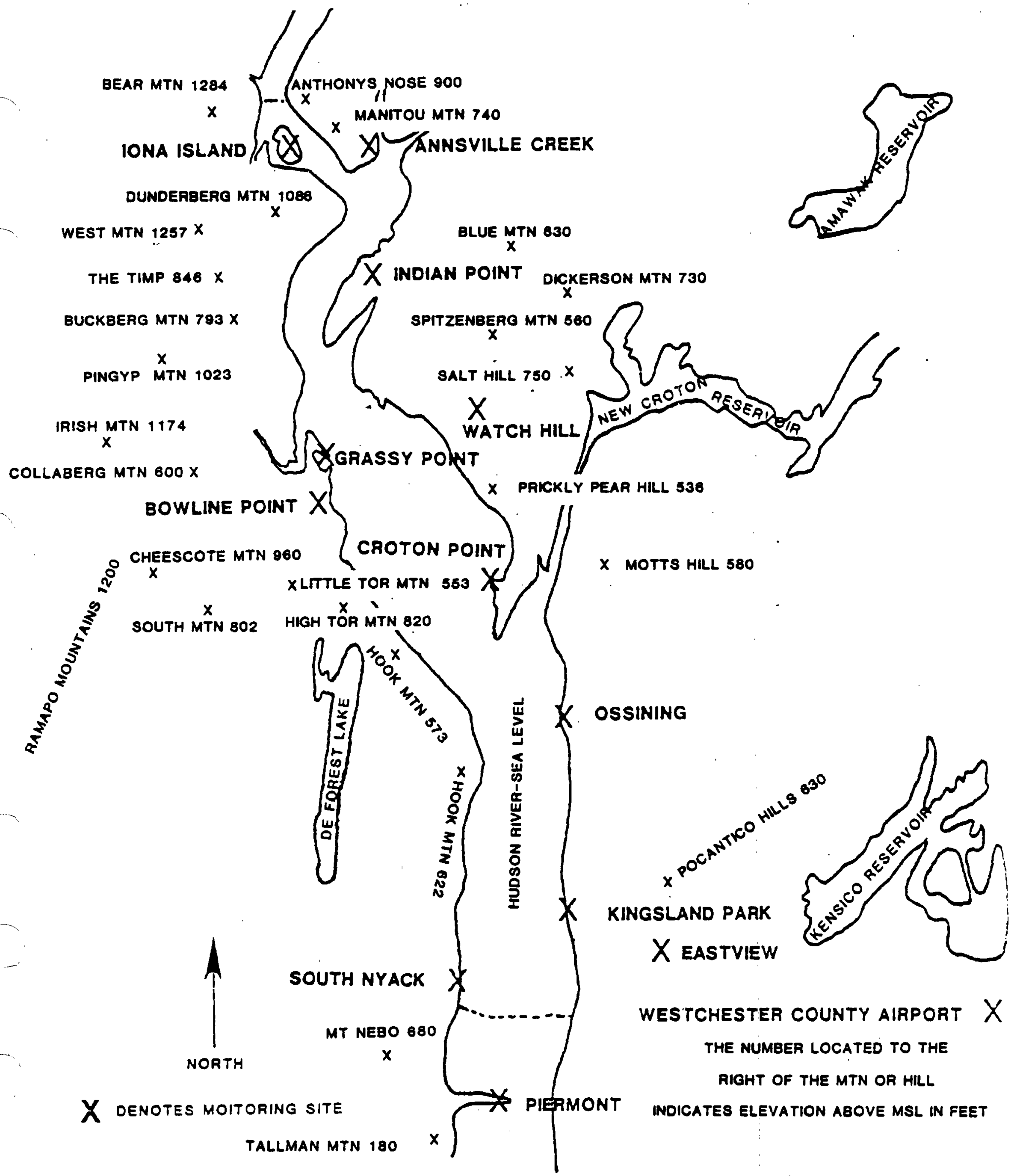


FIGURE 4

WATER COURSES IN THE INDIAN POINT REGION

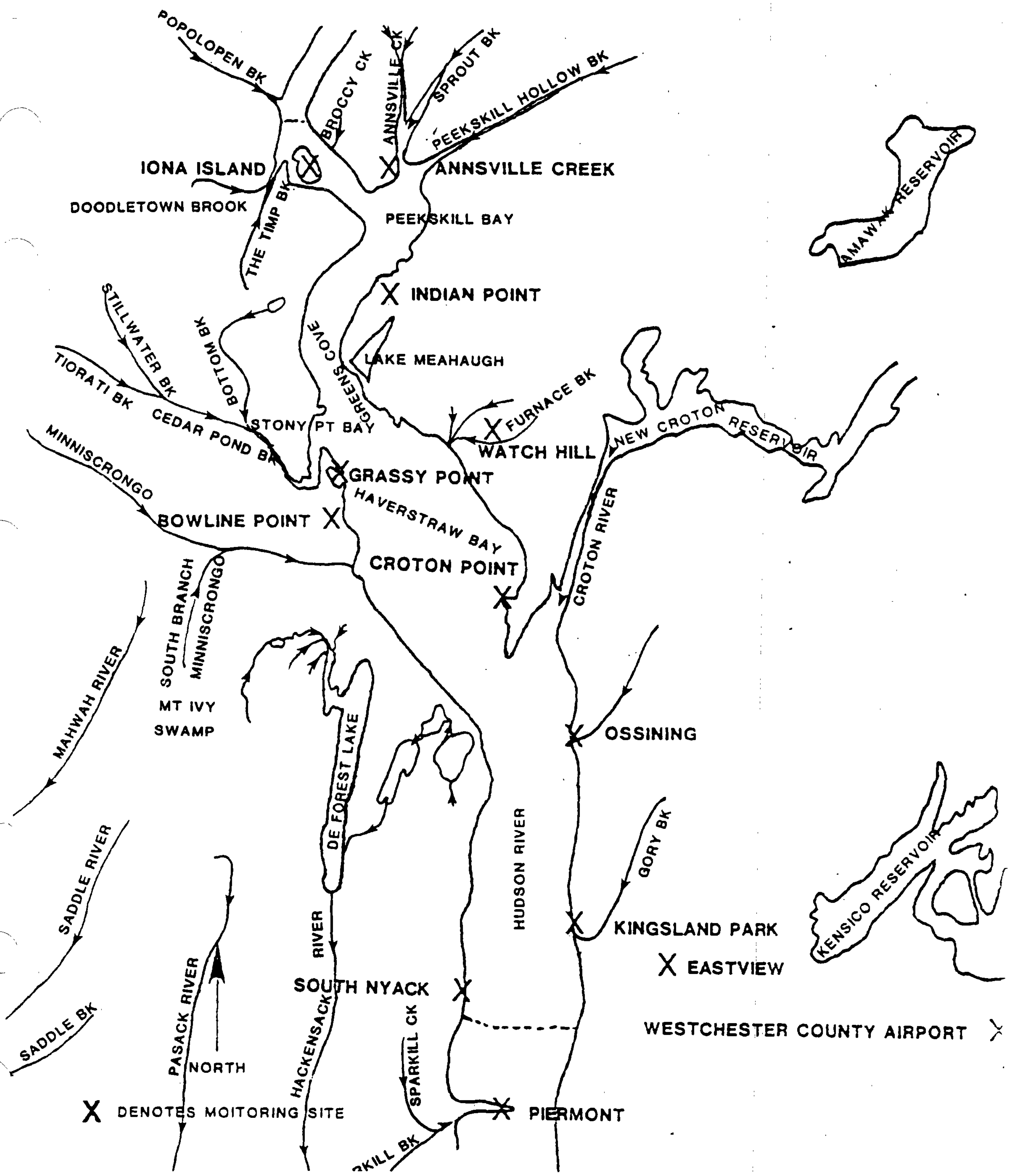


FIGURE 5

EXISTING AND HISTORICAL METEOROLOGICAL TOWERS AT INDIAN POINT

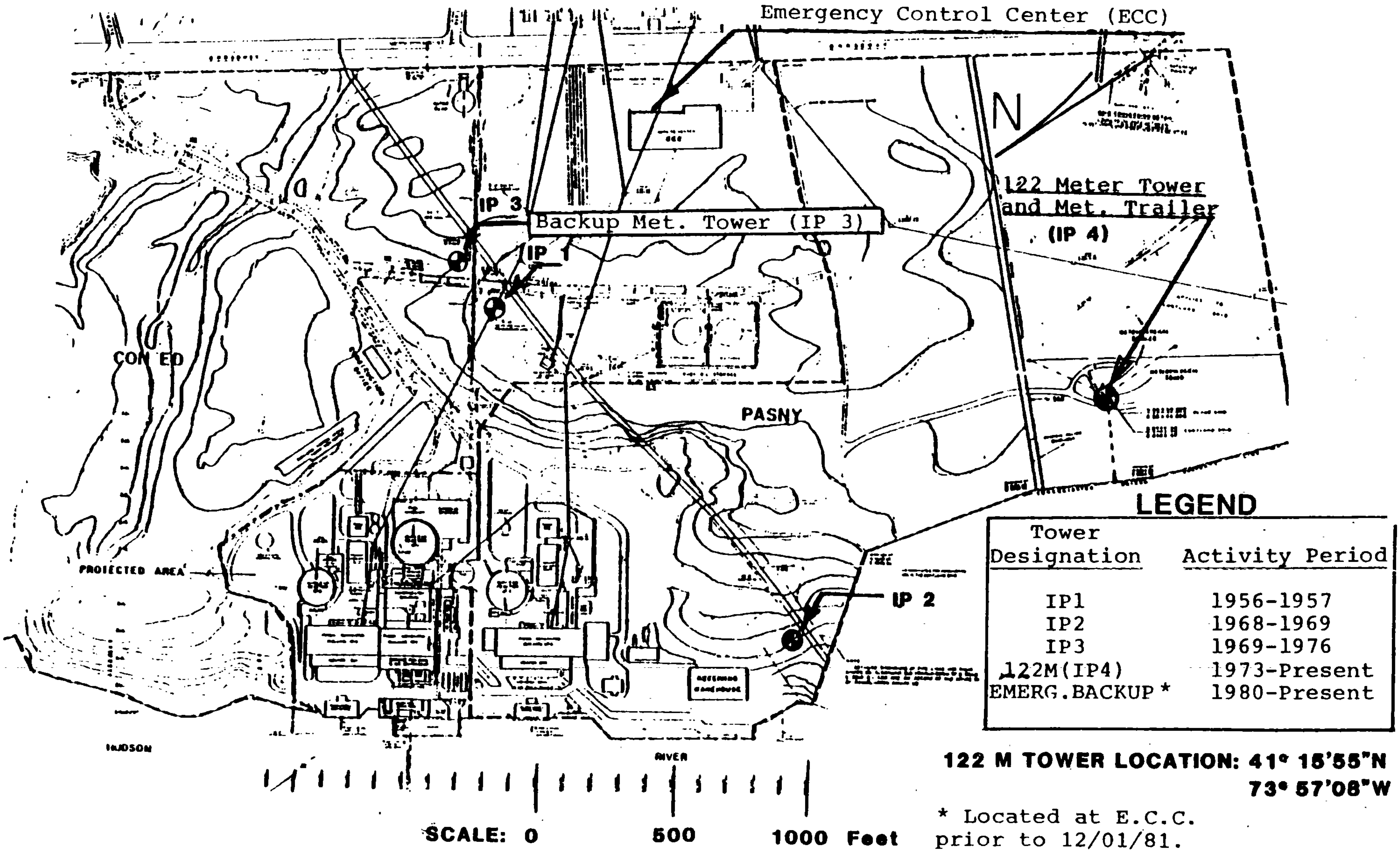


FIGURE 6

INDIAN POINT METEOROLOGICAL SITE

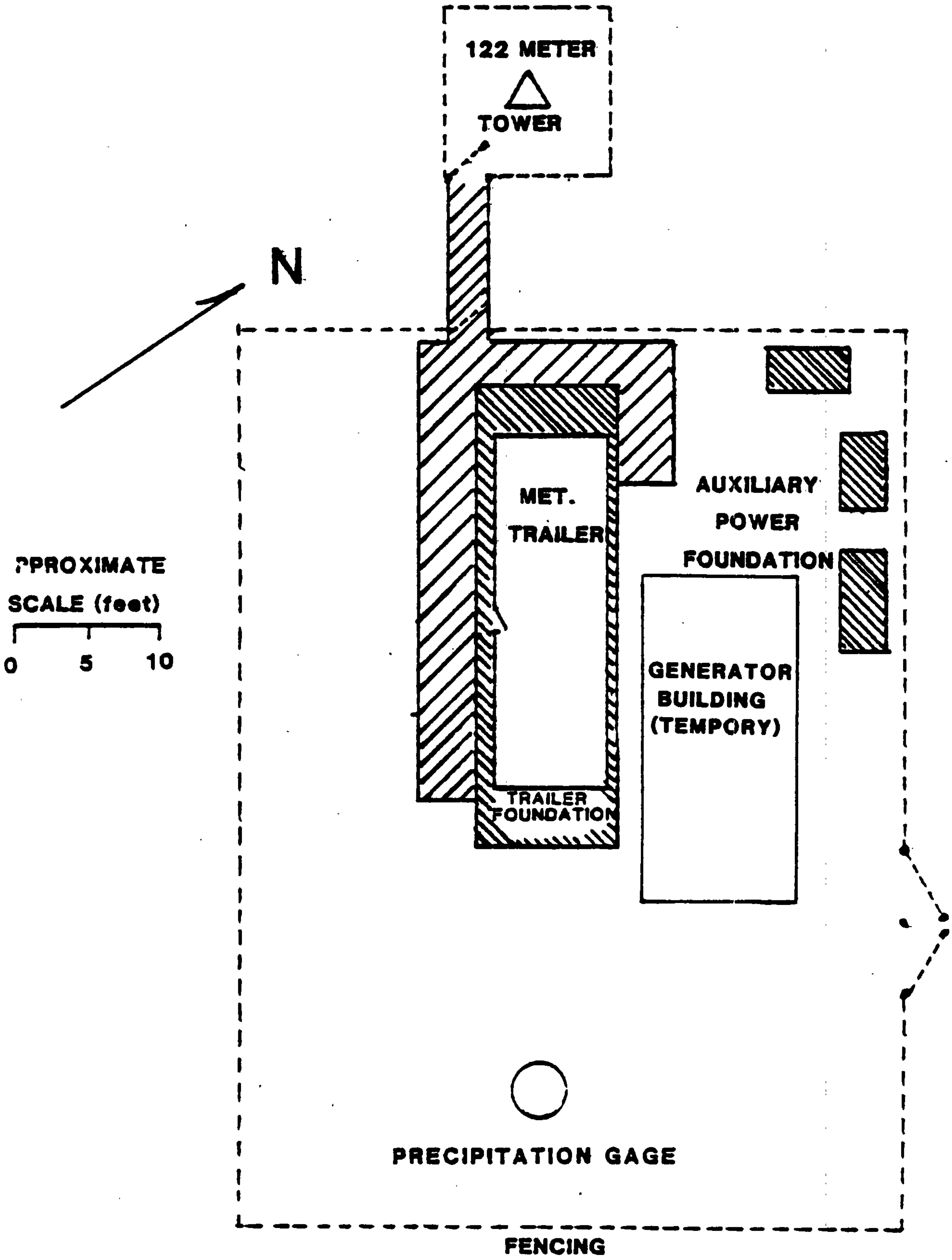
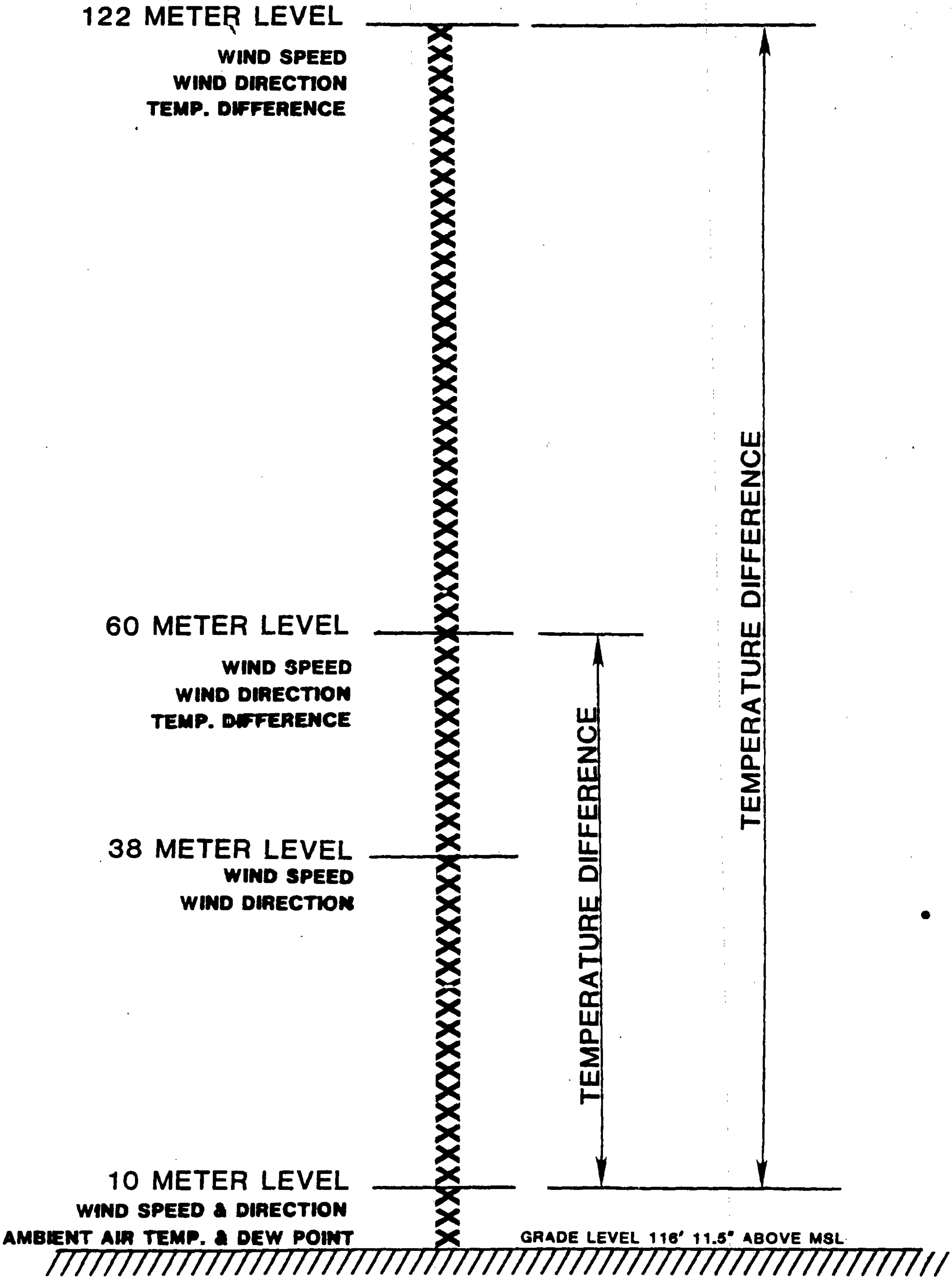


FIGURE 7

TOWER CONFIGURATION



NOTE: ALL TOWER LEVELS ARE ABOVE GRADE ELEVATION

FIGURE 8
STATION CONFIGURATION 12/01/81

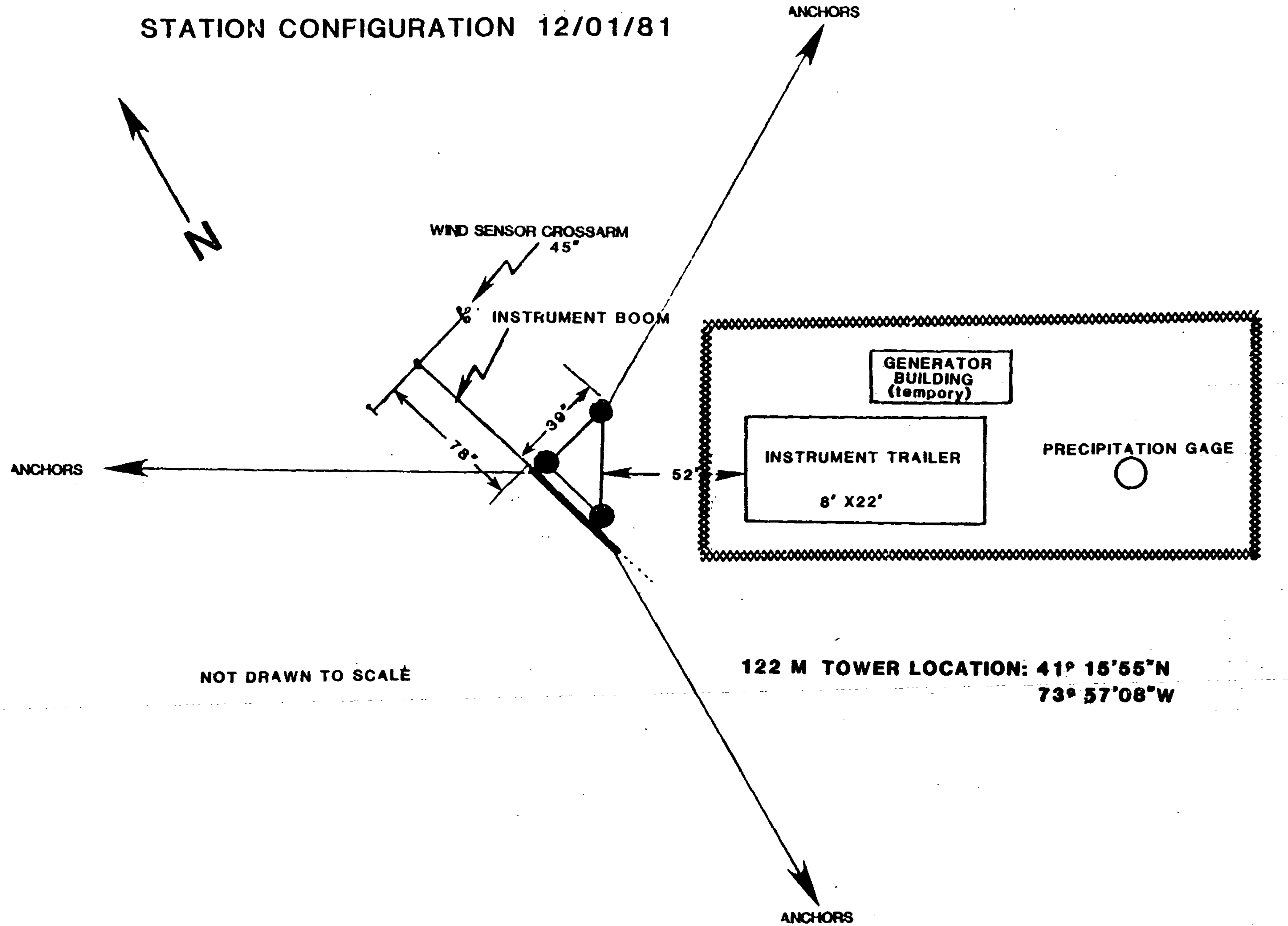


FIGURE 9

INDIAN POINT-METEOROLOGICAL SUPPORT SYSTEMS

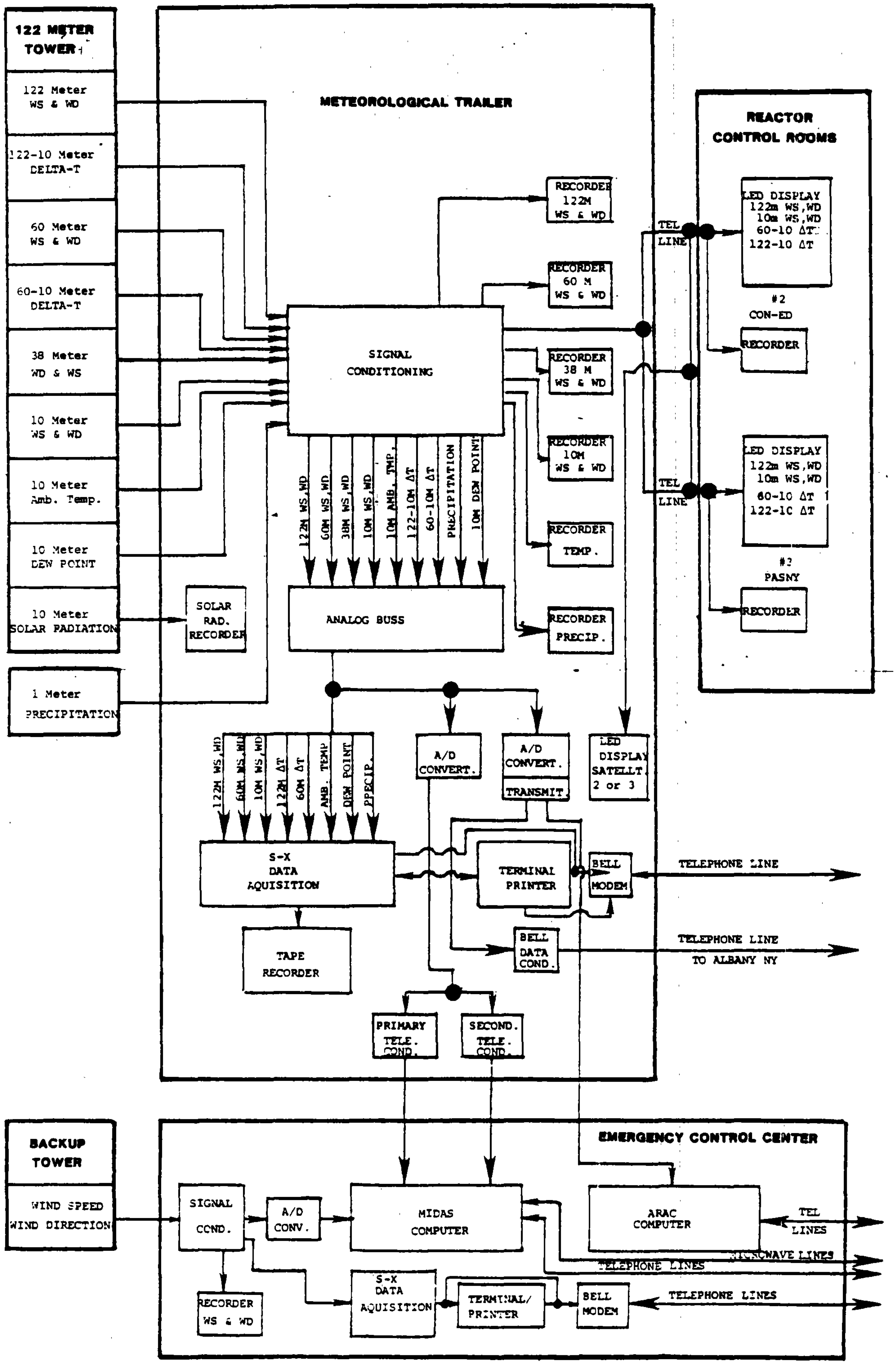


FIGURE 10A

TWO STATION WIND CORRELATION

DATA PERIOD

OCTOBER 1973

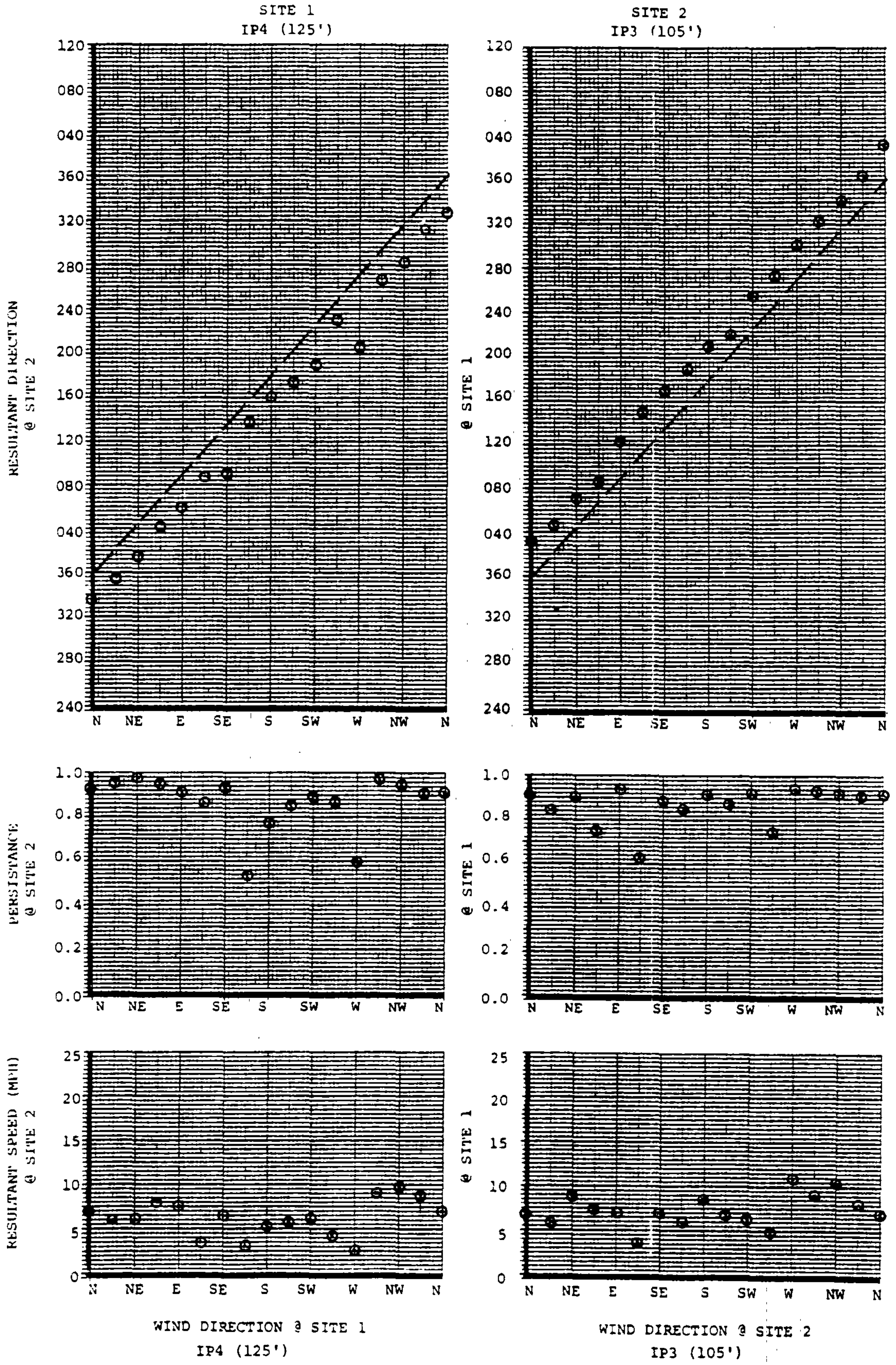


FIGURE 10B

TWO STATION WIND CORRELATION

DATA PERIOD

DECEMBER 1973

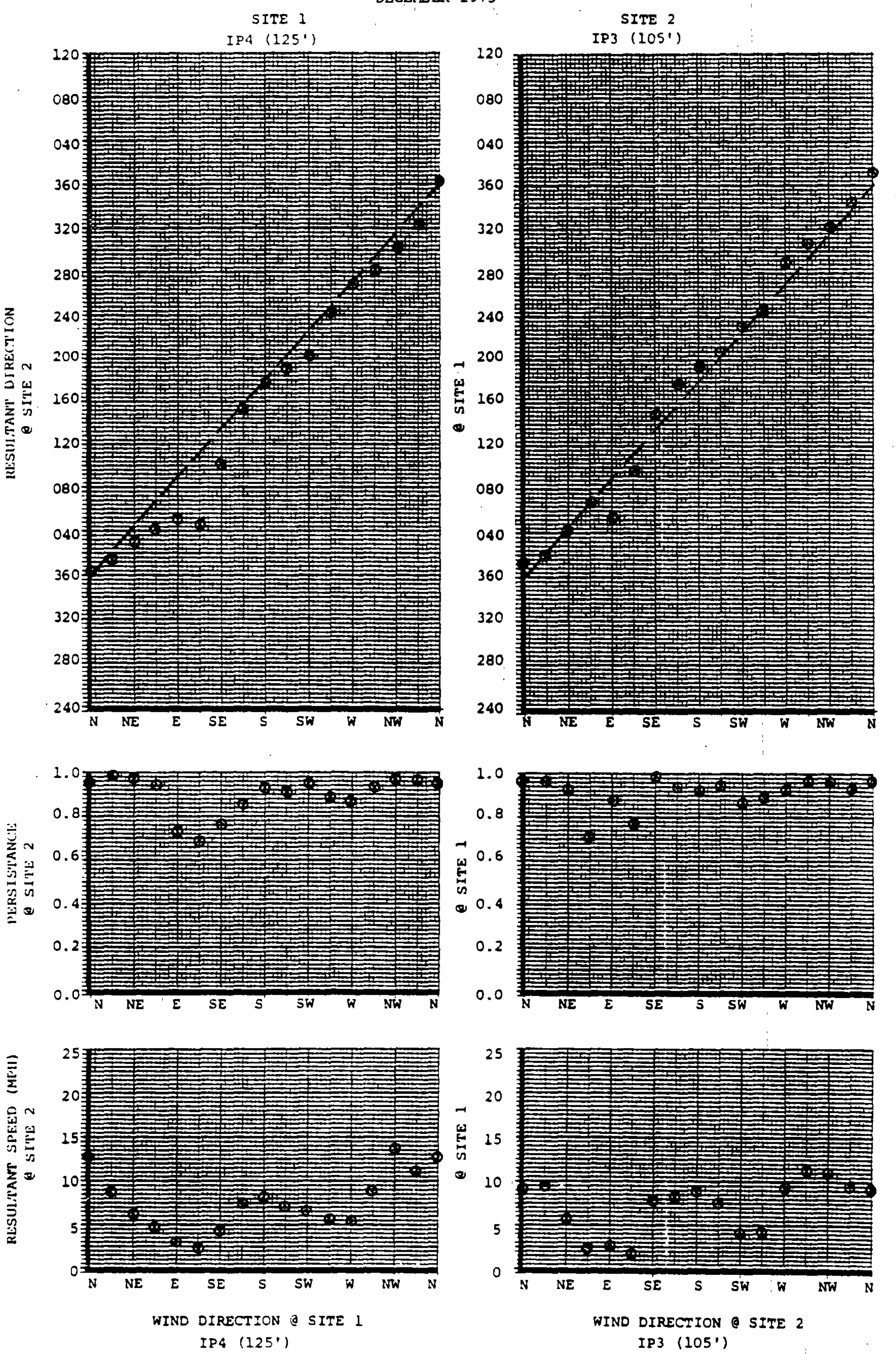


FIGURE 11

POSITION OF ONE MILE GRID

IN RELATION TO

TOPOGRAPHIC FEATURES

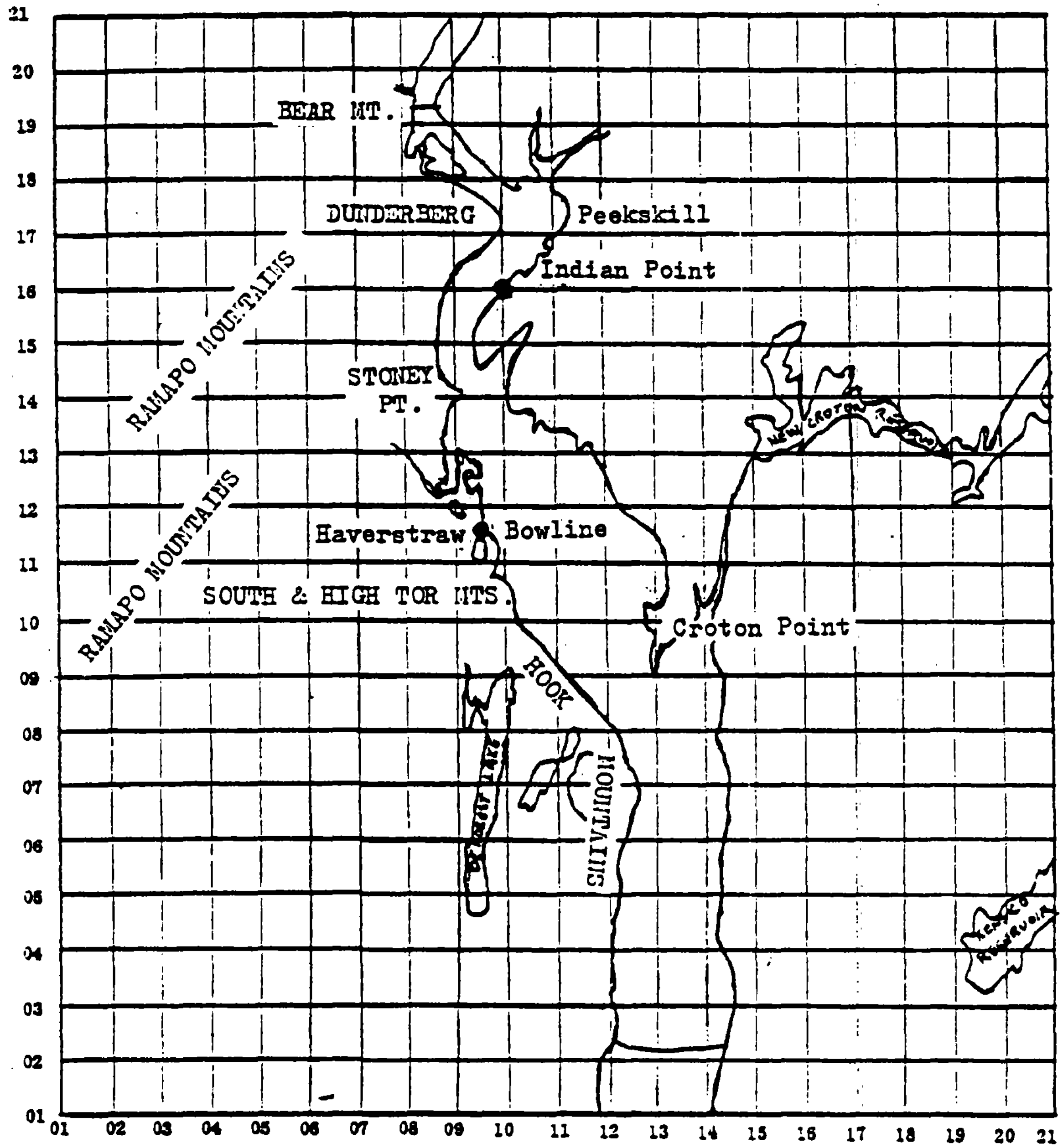


FIGURE 12

POSITION OF WIND FILES ON GRID

1: INDIAN POINT   2: BOWLINE POINT

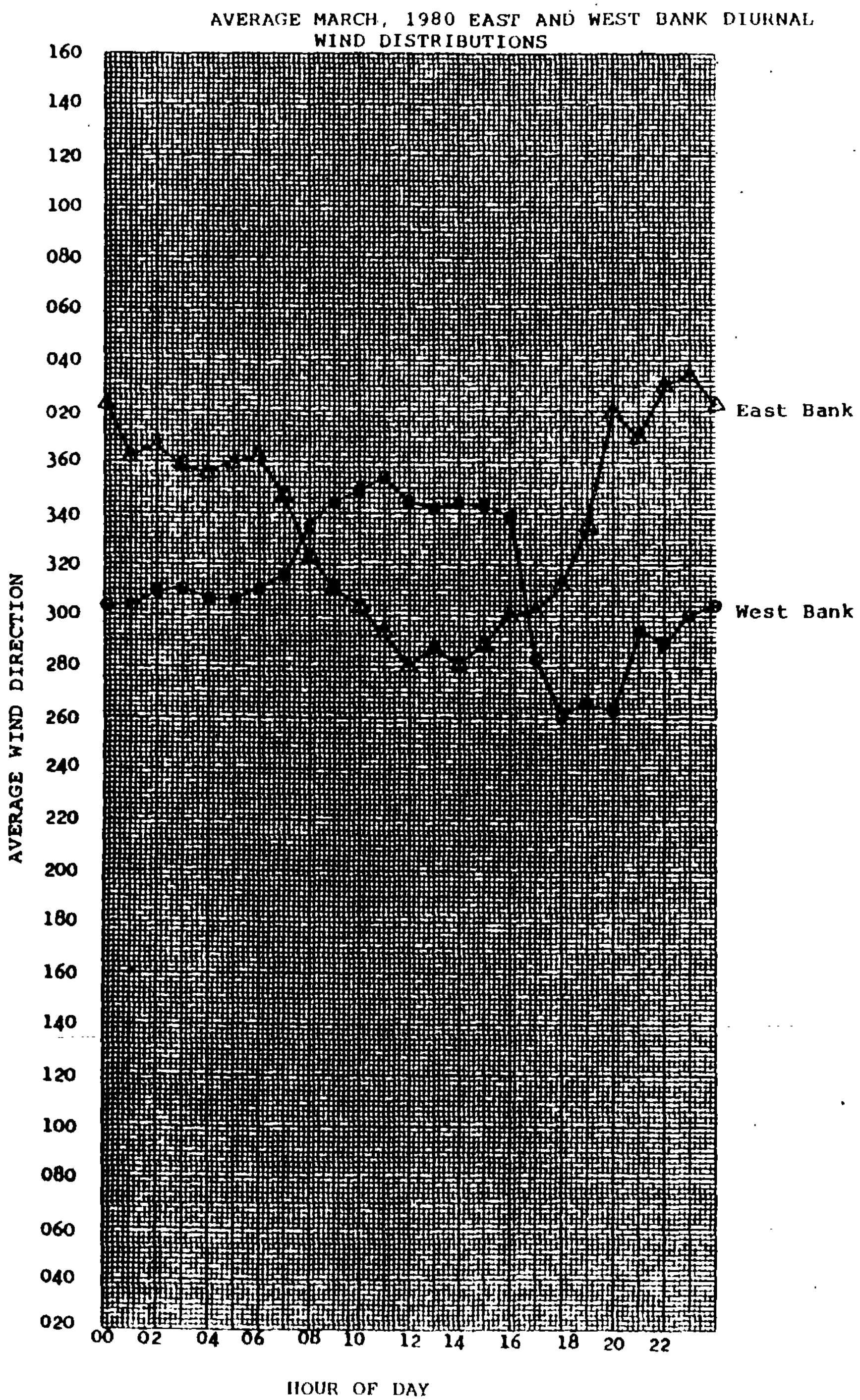
PATTERN 1

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21
21	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
20	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
19	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
18	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
15	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
14	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
13	2	2	2	2	2	2	2	2	2	2	2	1	1	1	1	1	1	1	1	1	1
12	2	2	2	2	2	2	2	2	2	2	2	2	1	1	1	1	1	1	1	1	1
11	2	2	2	2	2	2	2	2	2	2	2	2	2	2	1	1	1	1	1	1	1
10	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2
9	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2
8	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2
7	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2
6	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2
5	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2
4	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2
3	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2
2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2
1	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2

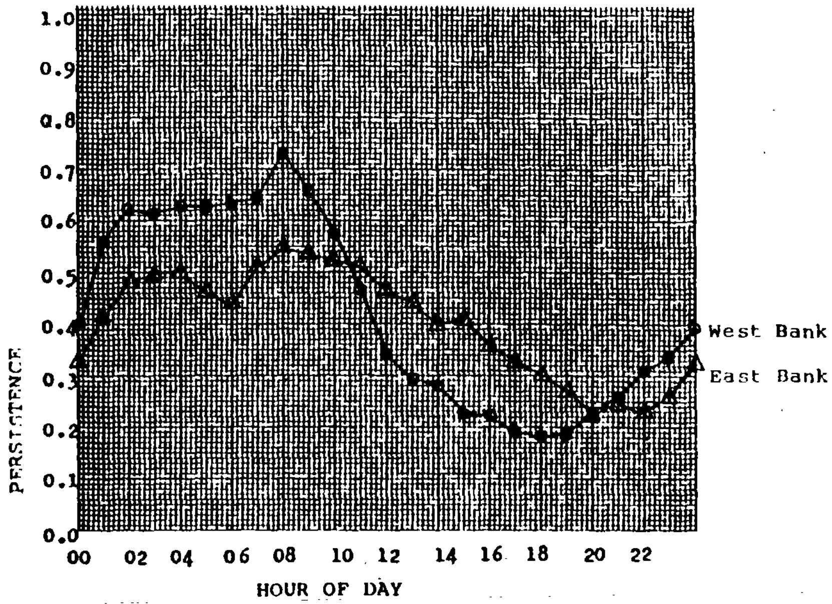
PATTERN 2

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21
21	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
20	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
19	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
18	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
15	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
14	2	2	2	2	2	2	2	2	2	2	1	1	1	1	1	1	1	1	1	1	1
13	2	2	2	2	2	2	2	2	2	2	2	1	1	1	1	1	1	1	1	1	1
12	2	2	2	2	2	2	2	2	2	2	2	2	1	1	1	1	1	1	1	1	1
11	2	2	2	2	2	2	2	2	2	2	2	2	2	2	1	1	1	1	1	1	1
10	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2
9	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2
8	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2
7	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2
6	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2
5	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2
4	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2
3	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2
2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2
1	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2	2

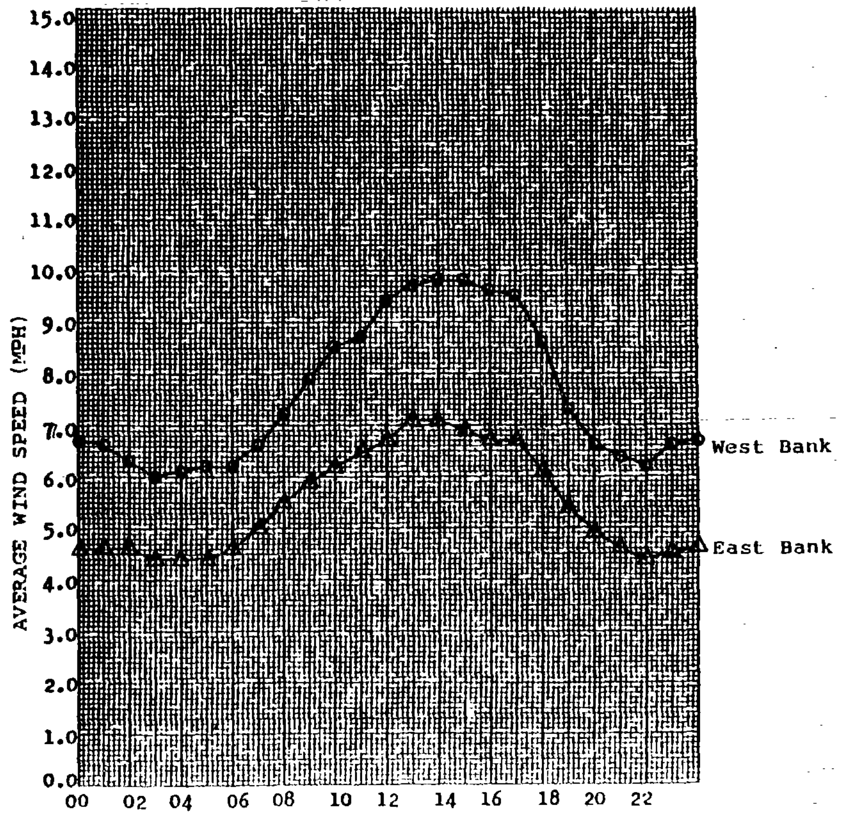
FIGURE 13



AVERAGE MARCH, 1980 EAST AND WEST BANK DIURNAL
WIND DISTRIBUTIONS
AVERAGE WIND DIRECTION
HOUR OF DAY
East Bank
West Bank

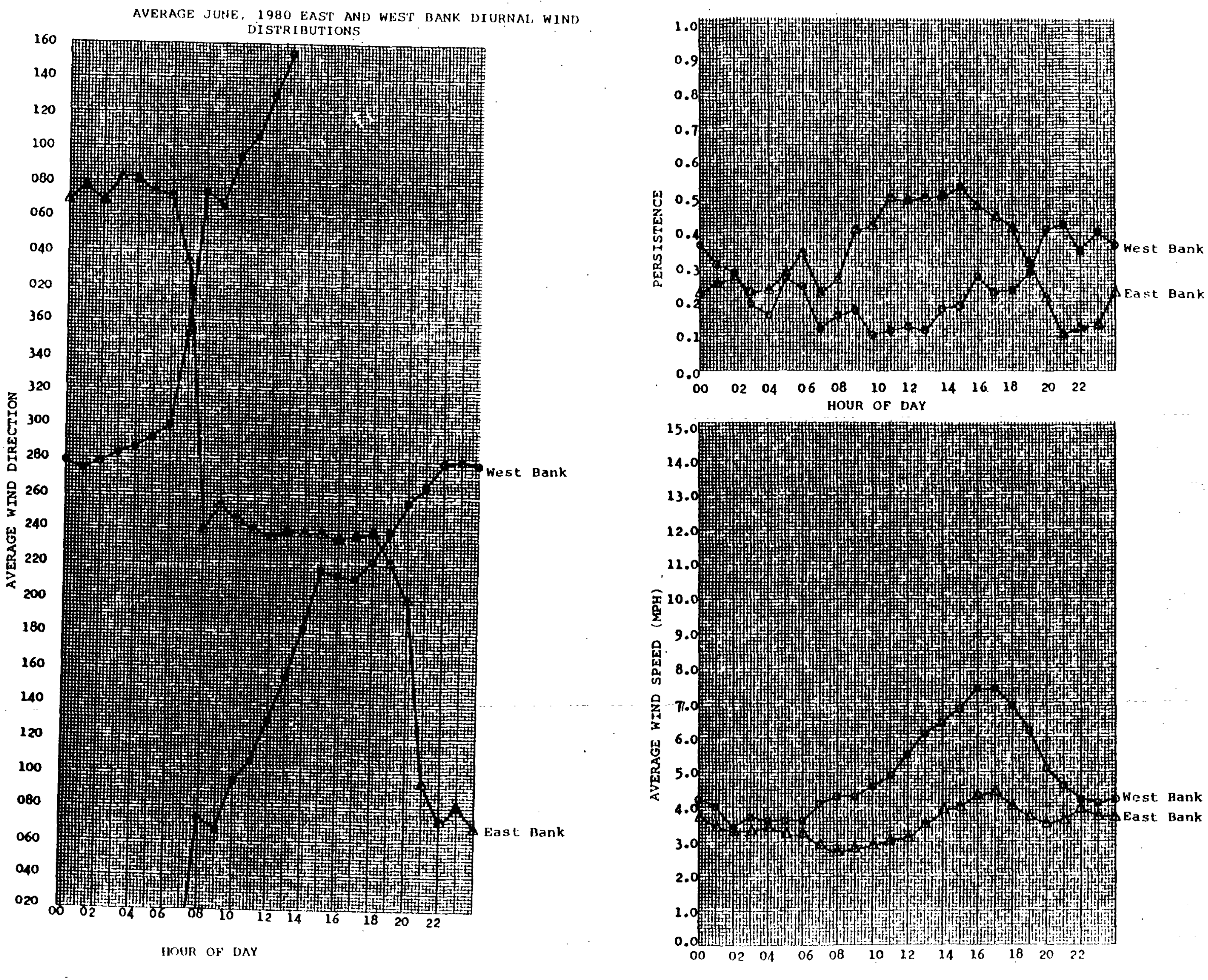


PERSISTENCE
HOUR OF DAY
West Bank
East Bank



AVERAGE WIND SPEED (MPH)
West Bank
East Bank

FIGURE 14



AVERAGE JUNE, 1980 EAST AND WEST BANK DIURNAL WIND DISTRIBUTIONS
AVERAGE WIND DIRECTION
HOUR OF DAY
West Bank
East Bank
PERSISTENCE
HOUR OF DAY
West Bank
East Bank
AVERAGE WIND SPEED (MPH)
West Bank
East Bank

FIGURE 15

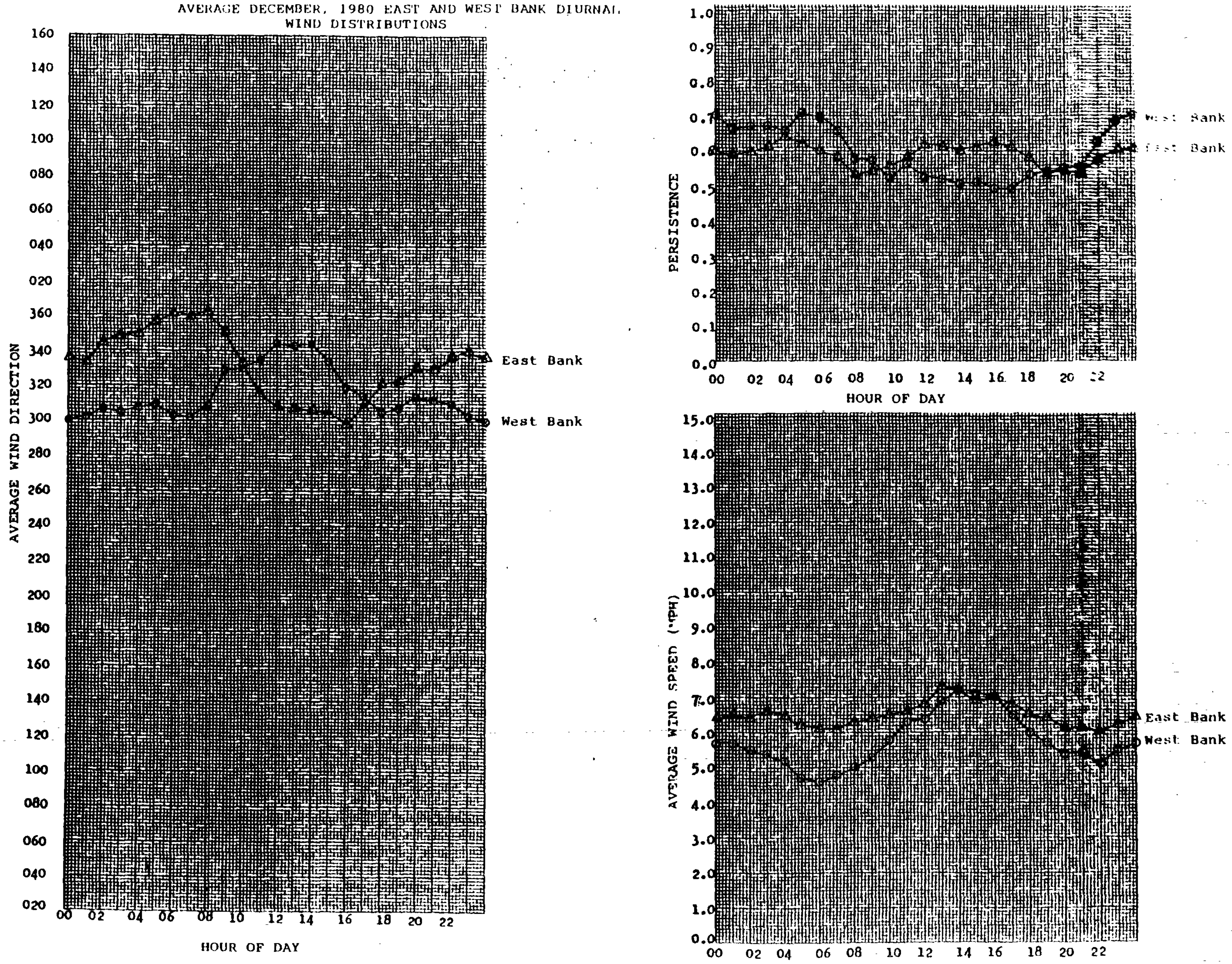
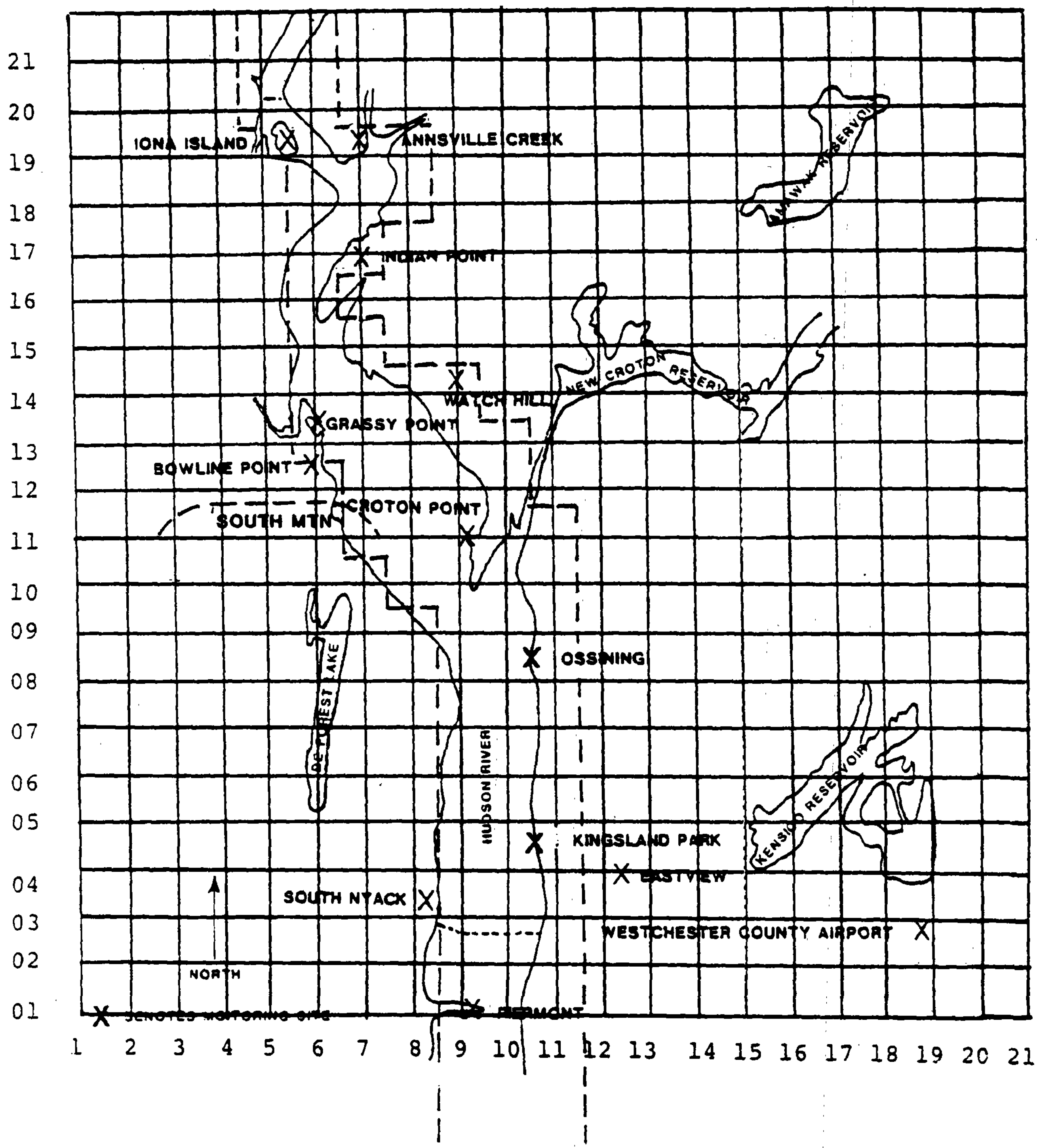

AVERAGE DECEMBER, 1980 EAST AND WEST BANK DIURNAL WIND DISTRIBUTIONS
AVERAGE WIND DIRECTION
HOUR OF DAY
East Bank
West Bank
PERSISTENCE
HOUR OF DAY
West Bank
East Bank
AVERAGE WIND SPEED (MPH)
East Bank
West Bank

FIGURE 16

LOCATIONS OF MONITORING SITES IN RELATION TO ONE MILE GRID



Idealized
River Valley

FIGURE 17

COMPARISON OF 10M LEVEL DIURNAL WIND DISTRIBUTIONS

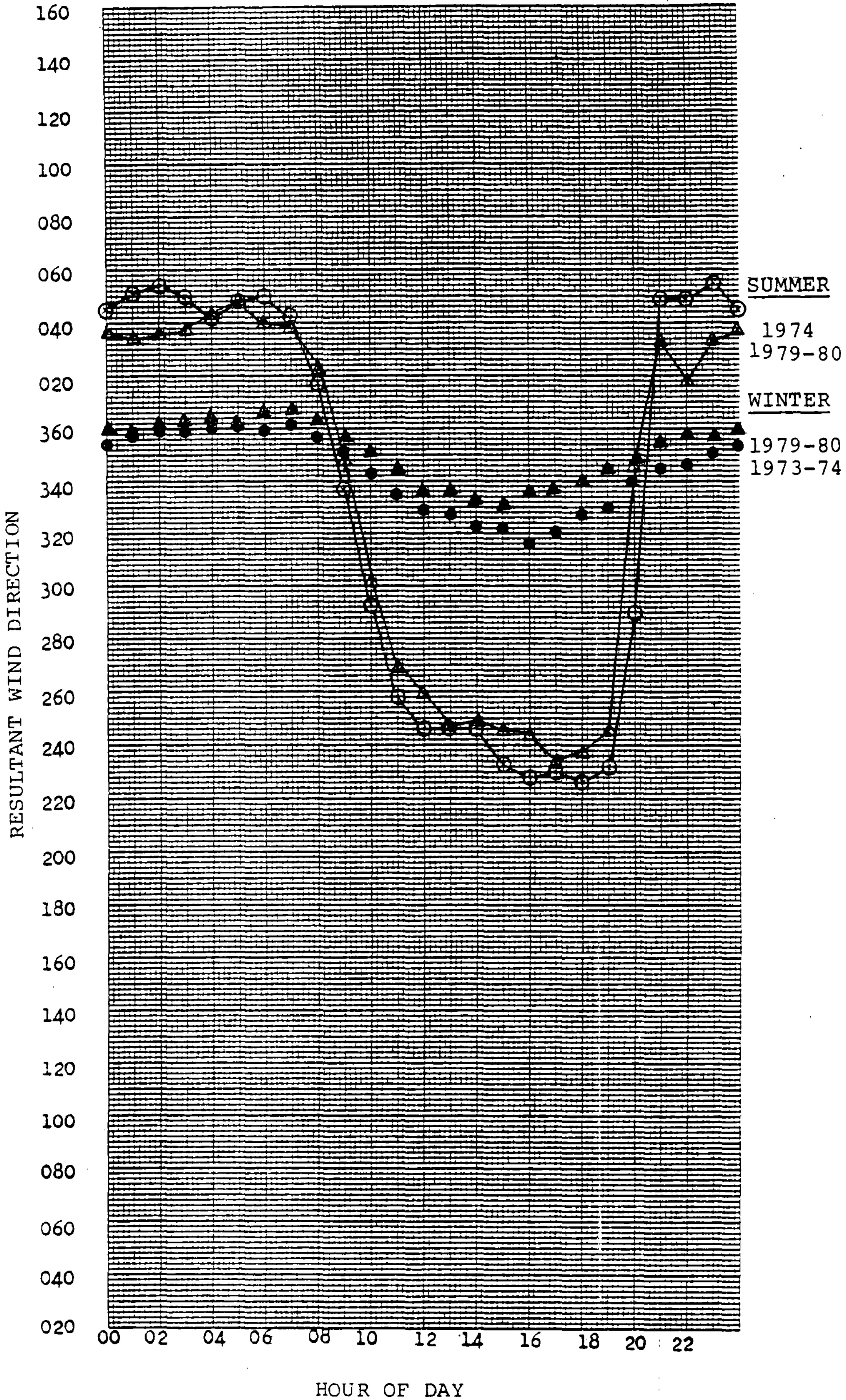


FIGURE 18
COMPARISON OF 122M LEVEL DIURNAL AND WIND DISTRIBUTION

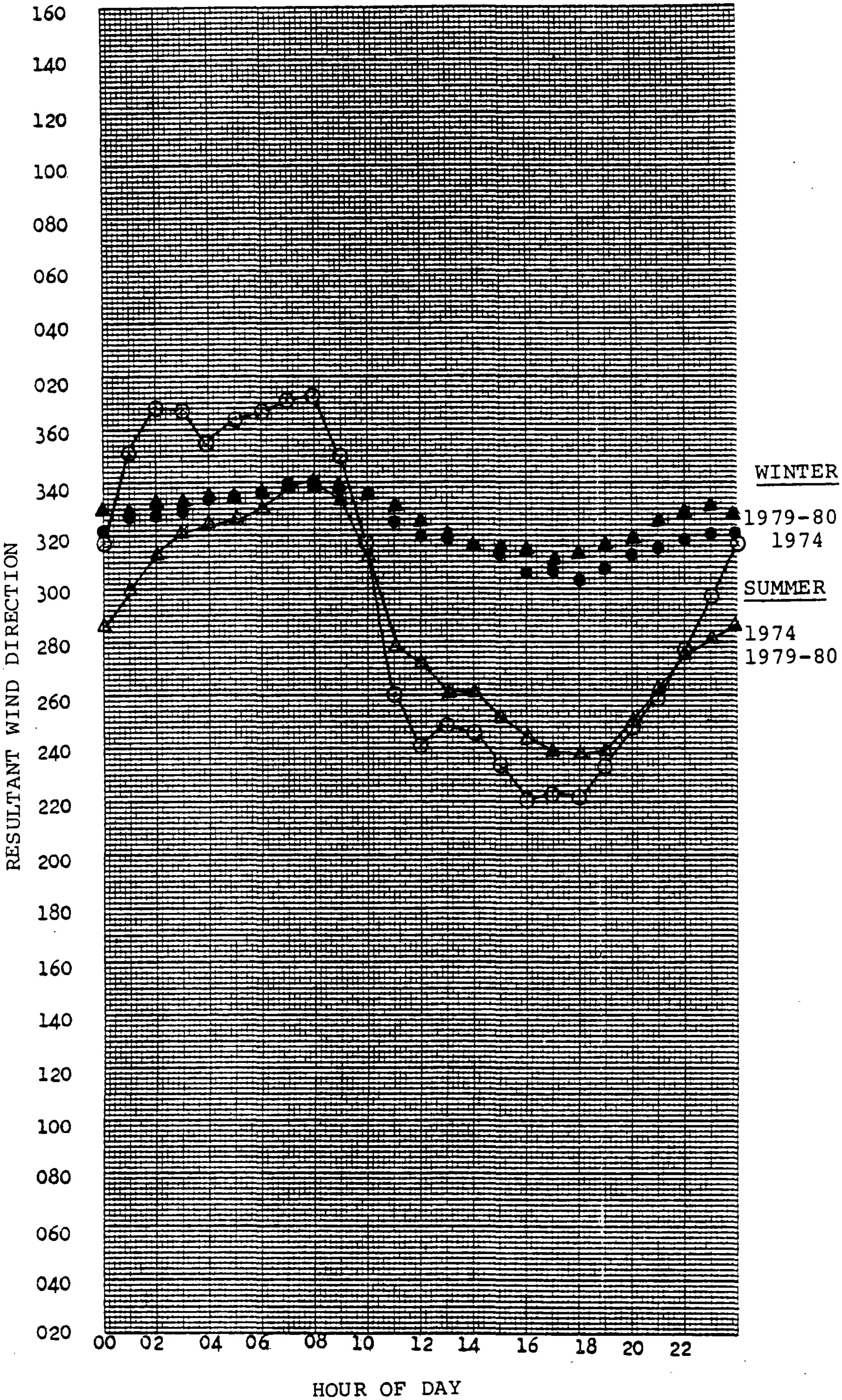


FIGURE 19

DIURNAL DISTRIBUTION OF WIND SPEEDS

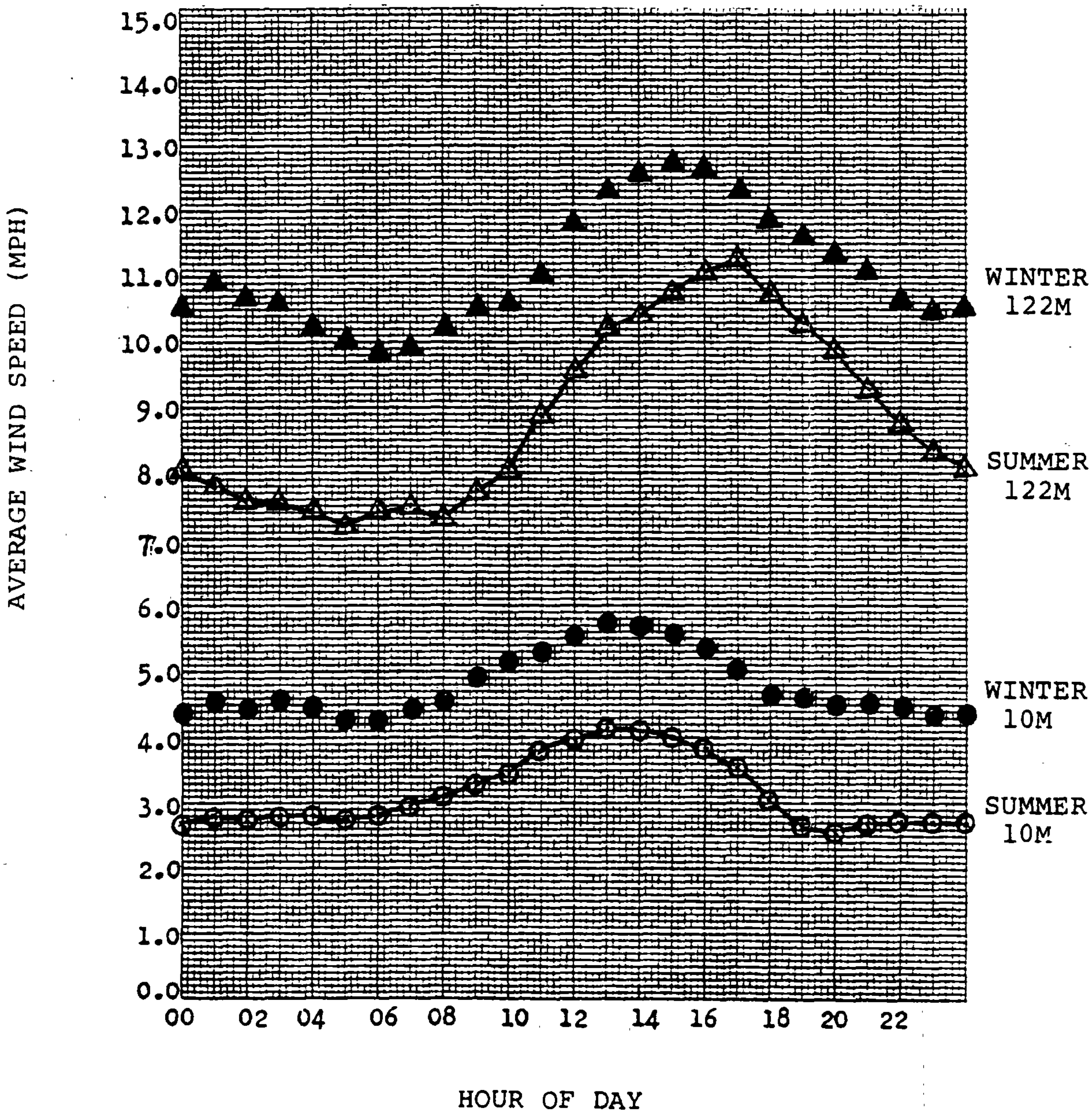
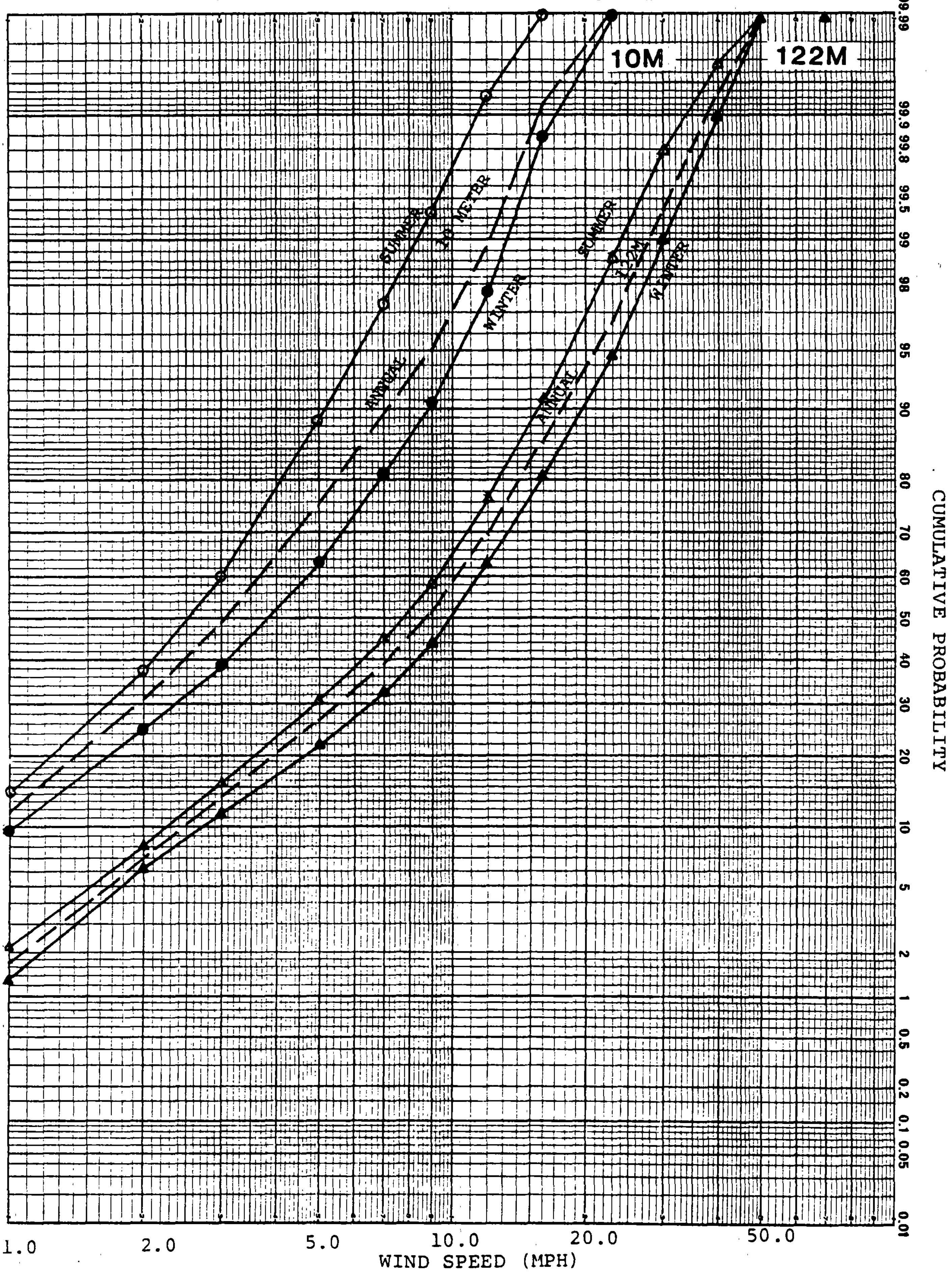
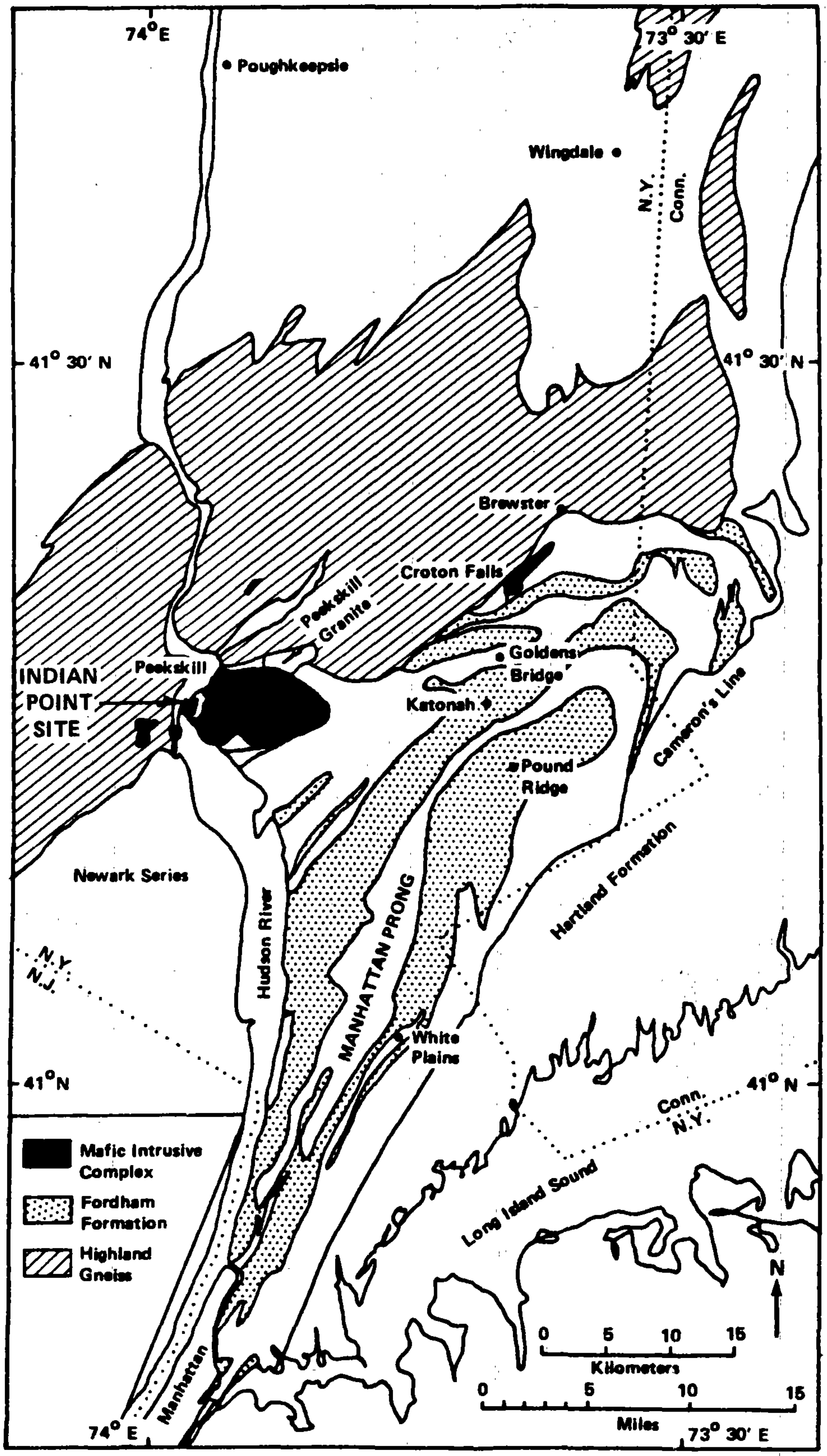


FIGURE 20

PERCENT PROBABILITY DISTRIBUTION OF WIND SPEEDS





GEOLOGIC MAP, SOUTHEASTERN NEW YORK

(AFTER BROCK & MOSE, 1979)

FIGURE 1

CHAPTER 3
REACTOR

3.0 DESCRIPTION

The reactor core has a number of fuel regions with fuel assemblies arranged in a zone and/or checkered core pattern. The fuel rods are cold worked Zircaloy or ZIRLO™ tubes containing slightly enriched uranium-dioxide fuel.

The fuel assembly is a canless 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 are loaded as feed fuel. For Cycle 17, 18, 19 and 20 15x15 Upgraded fuel design assemblies have been loaded as feed fuel.

Rod cluster control assemblies and wet annular burnable absorber rods are inserted into the guide thimbles of the fuel assemblies. The absorber sections of the control rods are fabricated of silver-indium-cadmium alloy sealed in stainless steel tubes.

The control rod drive mechanisms for the full-length rod cluster control assemblies are of the magnetic latch type. The latches are controlled by three magnetic coils. They are so designed that upon a loss of power to the coils, the rod cluster control assembly is released and falls by gravity to shut down the reactor.

3.1 DESIGN BASES

3.1.1 Performance Objectives

The reactor thermal power analyzed is 3216 MWt (3230 MWt total NSSS power), which is the licensed power rating. Calculations and operating experience indicate that hot channel factors will be considerably less than those used for design purposes in this application.

The initial reactor core fuel loading and programming were designed to yield the first cycle average burnup of 16,100 MWd/metric ton uranium (MTU). Cycles 2 through 4 reload designs yielded an average cycle burnup of 10,000 MWd/metric ton uranium, and Cycles 5 through 10 yielded an average cycle burnup of 13,500 MWd/metric ton uranium. Cycles 11, 12, and 13 achieved burnups of 18,094 MWd/MTU, 20,674 MWd/MTU, and 21,650 MWd/MTU, all of which included coastdown operation. Cycle 14 achieved a cycle burnup of 18,970 MWd/MTU. Cycle 15 achieved a cycle burnup of 22,110 MWd/MTU. Cycle 16 achieved a cycle burnup of 23,950 MWd/MTU with a coastdown. Cycle 17 achieved a cycle burnup of 18,708 MWd/MTU. Cycle 18 achieved a cycle burnup of 24,456 MWd/MTU. Cycle 20 is designed to achieve a cycle burnup of 25,193 MWd/MTU. The fuel rod cladding is designed to maintain its integrity for the anticipated core life. The effects of gas release, fuel dimensional changes, and corrosion-induced or irradiation-induced changes in the mechanical properties of cladding are considered in the design of the fuel assemblies.

Rod cluster control assemblies are employed to provide sufficient reactivity control to terminate any credible power transient prior to reaching the minimum departure from nucleate boiling ratio (DNBR). This is accomplished by ensuring sufficient cluster control worth to shut the reactor down by at least 1.3-percent throughout core life in the hot condition with the most reactive cluster control stuck in the fully withdrawn position. Redundant equipment is provided to add soluble poison to the reactor coolant in the form of boric acid to maintain shutdown margin when the reactor is cooled to ambient temperatures.

Experimental measurements from critical experiments or operating reactors, or both, are used to validate the methods employed in the design. During design, nuclear parameters have been calculated for every phase of operation of the first core and reload cycles and, where applicable, are compared with design limits to show that an adequate margin of safety exists.

In the thermal hydraulic design of the core, the maximum fuel and clad temperatures during normal reactor operation and at overpower conditions have been conservatively evaluated and found to be consistent with safe operating limitations.

3.1.2 Principal Design Criteria

3.1.2.1 Reactor Core Design

Criterion: The reactor core with its related controls and protection systems shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits, which have been stipulated and justified. The core and related auxiliary system designs shall provide this integrity under all expected conditions of normal operation with appropriate margins for uncertainties and for specified transient situations, which can be anticipated. (GDC 6)

The reactor core, with its related control and protection system, is designed to function throughout its design lifetime without exceeding acceptable fuel damage limits. The core design, together with reliable process and decay heat removal systems, provides for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and anticipated transient situations, including the effects of the loss of reactor coolant flow (Section 14.1.6), trip of the turbine generator (Section 14.1.8), loss of normal feedwater (Section 14.1.9), and loss of all offsite power (Section 14.1.12). The reactor protection system is designed to actuate a reactor trip for any anticipated combination of plant conditions, when necessary, to ensure a minimum departure from nucleate boiling (DNB) ratio equal to or greater than the applicable safety analysis limit DNBR.

The integrity of fuel cladding is ensured by preventing excessive fuel swelling, excessive fuel densification, excessive clad heating, excessive cladding stress and strain. This is achieved by designing the fuel rods so that the following conservative limits are not exceeded during normal operation or any anticipated transient condition (Condition II events):

1. Minimum DNB ratio equal to or greater than the safety analysis limit DNBR.
2. Fuel center temperature below 4700°F.
3. Internal gas pressure (limited to value below that which could cause the diameter-to-gap to increase due to outward clad creep during steady-state operation, and which could cause excessive DNB propagation to occur).
4. Clad stresses less than the Zircaloy or ZIRLO™ yield strength.
5. Clad strain less than 1-percent.

The ability of fuel designed and operated to these criteria to withstand postulated normal and abnormal service conditions is shown by analyses described in Chapter 14 to satisfy the demands of plant operation well within applicable regulatory limits.

The reactor coolant pumps provided for the plant are supplied with sufficient rotational inertia to maintain an adequate flow coastdown and prevent core damage in the event of a simultaneous loss of power to all pumps as discussed in section 14.1.6.1.

In the unlikely event of a turbine trip from full power without an immediate reactor trip, the subsequent reactor coolant temperature increase and volume insurge to the pressurizer result in a high pressurizer pressure trip and thereby prevent fuel damage for this transient. A loss of external electrical load of 50-percent of full power or less is normally controlled by rod cluster insertion together with a controlled steam dump to the condenser to prevent a large temperature and pressure increase in the reactor coolant system and thus prevent a reactor trip. In this case, the overpower-overtemperature protection would guard against any combination of pressure, temperature, and power, which could result in a DNB ratio less than the applicable safety analysis limit DNBR during the transient.

In neither the turbine trip nor the loss-of-flow events do the changes in coolant conditions provoke a nuclear power excursion because of the large system thermal inertia and relatively small void fraction as discussed in UFSAR sections 14.1.8 and 14.1.6, respectively. Protection circuits actuated directly by the coolant conditions identified with core limits are therefore effective in preventing core damage.

3.1.2.2 Suppression of Power Oscillations

Criterion: The design of the reactor core with its related controls and protection systems shall ensure that power oscillations, the magnitude of which could cause damage in excess of acceptable fuel damage limits, are not possible or can be readily suppressed. (GDC 7)

The potential for possible spatial oscillations of power distribution for this core has been reviewed. It is concluded that low frequency xenon oscillations may occur in the axial dimension, and the control rods are provided to suppress these oscillations. The core is expected to be stable to xenon oscillations in the X-Y dimension. Ex-core instrumentation is provided to obtain necessary information concerning power distribution. This instrumentation is adequate to enable the operator to monitor and control xenon induced oscillations. (Incore instrumentation is used periodically to calibrate and verify the information provided by the ex-core instrumentation. The analysis, detection, and control of these oscillations is discussed in Reference 1.)

3.1.2.3 Redundancy of Reactivity Control

Criterion: Two independent reactivity control systems, preferably of different principles, shall be provided. (GDC 27)

Two independent reactivity control systems are provided, one involving rod cluster control assemblies and the other involving chemical shim.

3.1.2.4 Reactivity Hot Shutdown Capability

Criterion: The reactivity control systems provided shall be capable of making and holding the core subcritical from any hot standby or hot operating condition. (GDC 28)

The reactivity control systems provided are capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes. The maximum excess reactivity expected for the core occurs for the cold, clean condition at the beginning-of-life of the initial core.

The rod cluster control assemblies are divided into control banks and shutdown banks. The control banks used in combination with chemical shim control provide control of the reactivity changes of the core throughout the life of the core during power operation. These banks of rod cluster control assemblies are used to compensate for short term reactivity changes at power that might be produced due to variations in reactor power level or in coolant temperature. The chemical shim control is used to compensate for the more slowly occurring changes in reactivity throughout core life such as those due to fuel depletion and fission product buildup.

3.1.2.5 Reactivity Shutdown Capability

Criterion: One of the reactivity control systems provided shall be capable of making the core subcritical under any anticipated operating condition including anticipated operational transients sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margin should assure subcriticality with the most reactive control rod fully withdrawn. (GDC 29)

The reactor core, together with the reactor control and protection system, is designed so that the minimum allowable DNBR is equal to or greater than the applicable safety analysis limit DNBR and there is no fuel melting during normal operation including anticipated transients.

The shutdown banks are provided to supplement the control banks of rod cluster control assemblies to make the reactor at least 1.3-percent subcritical (k_{eff} = 0.987) following trip from any credible operating condition assuming the most reactive rod cluster control assembly is in the fully withdrawn position.

Sufficient shutdown capability is also provided so that the minimum DNBR is equal to or greater than the applicable safety analysis limit DNBR, assuming the most reactive rod to be in the fully withdrawn position for the most severe anticipated cooldown transient associated with a single active failure, e.g., accidental opening of a steam bypass, or relief valve, or safety valve stuck open. This is achieved by the combination of control rods and automatic boric acid injection via the emergency core cooling system. The minimum design margin is 1.3-percent throughout core life, as discussed in UFSAR Section 14.2.5.2, assuming the maximum worth control rod is in the fully withdrawn position allowing 10-percent uncertainty in the control rod worth calculations.

Technical Specification 3.1.1 specifies the actual minimum required shutdown margin for core design.

Manually controlled boric acid addition is used to maintain the shutdown margin for the long-term conditions of xenon decay and plant cooldown. Redundant equipment is provided to guarantee the capability of adding boric acid to the reactor coolant system.

3.1.2.6 Reactivity Holddown Capability

Criterion: The reactivity control systems provided shall be capable of making the core subcritical under credible accident conditions with appropriate margins for contingencies and limiting any subsequent return to power such that there will be no undue risk to the health and safety of the public. (GDC 30)

Normal reactivity shutdown capability is provided by control rods following a trip signal, with boric acid injection used to compensate for the long-term xenon decay transient and for plant cooldown. As discussed in response to the previous criteria, the shutdown capability maintains the minimum DNBR above the limiting value and prevents exceeding core safety limits as a result of the cooldown associated with a safety valve stuck fully open.

Any time that the reactor is at power, the quantity of boric acid retained in the boric acid tanks and ready for injection always exceeds that quantity required for the normal cold shutdown. This quantity always exceeds the quantity of boric acid required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay. Boric acid is pumped from the boric acid tanks by one of two boric acid transfer pumps to the suction of one of three charging pumps, which injects boric acid into the reactor coolant. Any charging pump and either boric acid transfer pump can be operated from diesel-generator power on loss of station power.

On the basis of the above, the injection of boric acid is shown to afford backup reactivity shutdown capability, diverse from rod cluster controls, which normally serve this function in the short-term situation. Shutdown for long-term and reduced temperature conditions can be accomplished with boric acid injection using redundant components, thus achieving the measure of reliability implied by the criterion.

Alternately, boric acid solution at lower concentration can be supplied from the refueling water storage tank. This solution can be transferred directly by the charging pumps or alternately by the safety injection pumps. The reduced boric acid concentration lengthens the time required to achieve equivalent shutdown.

3.1.2.7 Reactivity Control Systems Malfunction

Criterion: The reactor protection systems shall be capable of protecting against any single malfunction of the reactivity control system, such as unplanned continuous withdrawal (not ejection or dropout of a control rod, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits. (GDC 31)

The reactor protection systems are capable of protecting against any single credible malfunction of the reactivity control system, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits.

Reactor shutdown with rods is completely independent of the normal rod control functions since the trip breakers completely interrupt the power to the rod mechanisms regardless of existing control signals.

Details of the effects of continuous withdrawal of a control rod and continuous deboration are described in Section 14.1 and Section 9.2, respectively.

3.1.2.8 Maximum Reactivity Worth of Control Rods

Criterion: Limits, which include reasonable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to lose capability of cooling the core. (GDC 32)

Limits, which include considerable margin, are placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals so as to lose capability to cool the core.

The reactor control system employs control rod clusters. A portion of these are designated shutdown rods and are fully withdrawn during power operation. The remaining rod clusters comprise the control groups, which are used to control load and reactor coolant temperature. The rod cluster drive mechanisms are wired into preselected groups, and are therefore prevented from being withdrawn in other than their respective groups. The rod drive mechanism is of the magnetic latch type and the coil actuation is sequenced to provide variable speed rod travel. The maximum reactivity insertion rate is analyzed in the detailed plant analysis assuming two of the highest worth groups to be accidentally withdrawn at maximum speed. This yields a reactivity insertion rate of the order of 70 pcm/sec, which is well within the capability of the overpower-overtemperature protection circuits to prevent core damage.

No single credible mechanical or electrical control system malfunction can cause a rod cluster to be withdrawn at a speed greater than 72 steps per min (approximately 45-in. per min) when the Rod Control System is operating in the MANUAL mode. Single credible electrical system failures can result in withdrawal speeds greater than 72 steps per min during automatic operation. The automatic rod withdrawal feature however, has been permanently defeated at IP2.

3.1.3 Safety Limits

The reactor is capable of meeting the performance objectives throughout core life under both operating and malfunction conditions without violating the integrity of the fuel elements. Thus the release of unacceptable amounts of fission products to the coolant is prevented.

The limiting conditions for operation established in the Technical Specifications specify the functional capacity of performance levels permitted to assure safe operation of the facility.

Design parameters, which are pertinent to safety limits are specified below for the nuclear, reactivity control, thermal and hydraulic, and mechanical aspects of the design.

3.1.3.1 Nuclear Limits

The nuclear axial peaking factor Fz, and the nuclear enthalpy rise hot channel factor $F^N \Delta_H$ are limited in their combined relationship so as not to exceed the F_Q or DNBR limits.

Potential axial xenon oscillations are controlled with control rods to preclude adverse core conditions. The protection system ensures that the nuclear core limits are not exceeded.

For Cycles 13 and beyond, a cycle specific Core Operating Limits Report (COLR) is generated. Provided in the COLR is the cycle specific F_q and $F\Delta_H$ limits as well as the limiting K(z) curve (normalized Fq* Power versus core height axial envelope). Also, cycle specific rod insertion limits and axial flux difference band width (delta-I) limits are provided in the COLR. Utilization of the COLR provides maximized operational and/or design flexibility, while adherence to the limits of the COLR and plant Technical Specifications assures that DNB and overpower design limits are met. The expected values for the nuclear heat flux and nuclear enthalpy rise hot channel factors for the current cycle are provided in Table 3.2-1A.

3.1.3.2 Reactivity Control Limits

The control system and the operational procedures provide adequate control of the core reactivity and power distribution. The following control limits are met:

1. A minimum hot shutdown margin of 1.3% K_{eff} is available throughout core life assuming a 10-percent uncertainty in the control rod calculation.
2. This shutdown margin is maintained with the most reactive rod cluster control assembly in the fully withdrawn position.
3. The shutdown margin is maintained at ambient temperature by the use of soluble poison.

Technical Specification 3.1.1 specifies the actual minimum required shutdown margin for core design.

3.1.3.3 Thermal and Hydraulic Limits

The reactor core is designed to meet the following limiting thermal and hydraulic criteria:

1. The minimum allowable DNBR during normal operation, including anticipated transients, is equal to the applicable safety analysis limit DNBR.
2. Fuel temperature not to exceed 4700^{o}F during any anticipated operating condition or anticipated malfunction.

To maintain fuel rod integrity and prevent fission product release, it is necessary to prevent clad overheating under all operating conditions. This is accomplished by preventing a departure from nucleate boiling (DNB), which would cause a large decrease in the heat transfer coefficient between the fuel rods and the reactor coolant, resulting in high clad temperatures.

The ratio of the heat flux causing DNB at a particular core location (as predicted by the WRB-1 correlation) to the existing heat flux at the same core location is the DNB ratio. A DNB ratio of 1.17 for the WRB-1 correlation corresponds to a 95-percent probability at a 95-percent confidence level that DNB does not occur. The DNB ratio for the W-3 correlation is 1.3 for pressure from 1000 to 2300 psia and 1.45 for pressure from 500 to 1000 psia.

3.1.3.4 Mechanical Limits

3.1.3.4.1 Reactor Internals

The reactor internal components are designed to withstand the stresses resulting from startup, steady state operation with any number of pumps running, and shutdown conditions. No damage to the reactor internals occurs as a result of loss of pumping power.

Lateral deflection and torsional rotation of the lower end of the core barrel is limited to prevent excessive movements resulting from seismic disturbances and thus prevent interference with rod cluster control assemblies. Core drop in the event of failure of the normal supports is limited so that the rod cluster control assemblies do not disengage from the fuel assembly guide thimbles.

The internals are further designed to maintain their functional integrity in the event of a major loss-of-coolant accident. The dynamic loading resulting from the pressure oscillations because of a loss-of-coolant accident does not cause sufficient deformation to prevent rod cluster control assembly insertion.

3.1.3.4.2 Fuel Assemblies

The fuel assemblies are designed to perform satisfactorily throughout their lifetime. The loads, stresses, and strains resulting from the combined effects of flow induced vibrations, earthquakes, reactor pressure, fission gas pressure, fuel growth, thermal strain, and differential expansion during both steady state and transient reactor operating conditions have been considered in the design of the fuel rods and fuel assemblies. The assemblies are also structurally designed to withstand handling and shipping loads prior to irradiation, and to maintain sufficient integrity at the completion of design burnup to permit safe removal from the core, subsequent handling during cooldown, shipment, and fuel reprocessing.

The fuel rods are supported at nine locations along their length within the fuel assemblies by grid assemblies, which are designed to maintain control of the lateral spacing between the rods through the design life of the assemblies. The magnitude of the support loads provided by the grids is established to minimize possible fretting without overstressing the cladding at the points of contact between the grids and fuel rods and without imposing restraints of sufficient magnitude to result in buckling or distortion of the rods. In addition, there are 3 Intermediate Flow Mixing (IFM) grids spaced along the fuel assembly and a protective grid (P-grid) on the bottom of the assembly. These grids do not provide any support function.

The fuel rod cladding is designed to withstand operating pressure loads without rupture and to maintain encapsulation of the fuel throughout the design life.

3.1.3.4.3 Rod Cluster Control Assemblies

The criteria used for the design of the cladding on the individual absorber rods in the rod cluster control assemblies are similar to those used for the fuel rod cladding. The cladding is designed to be free standing under all operating conditions and will maintain encapsulation of the absorber material throughout the absorber rod design life. Allowance for wear during operation is included for the rod cluster control assembly cladding thickness.

Adequate clearance is provided between the absorber rods and the guide thimbles, which position the rods within the fuel assemblies so that coolant flow along the length of the absorber rods is sufficient to remove the heat generated without overheating of the absorber cladding. The clearance is also sufficient to compensate for any misalignment between the absorber rods and guide thimbles and to prevent mechanical interference between the rods and guide thimbles under any operating conditions.

3.1.3.4.4 Control Rod Drive Assembly

Each control rod drive assembly is designed as a hermetically sealed unit to prevent leakage of reactor coolant. All pressure-containing components are designed to meet the requirements of the ASME Code, Section III, Nuclear Vessels for Class A vessels.

The control rod drive assemblies for the full length rods provide rod cluster control assembly insertion and withdrawal rates consistent with the required reactivity changes for reactor operational load changes. This rate is based on the worths of the various rod groups, which are established to limit power-peaking flux patterns to design values. The maximum reactivity addition rate is specified to limit the magnitude of a possible nuclear excursion resulting from a control system or operator malfunction. Also, the control rod drive assemblies for the full length rods provide a fast insertion rate during a "trip" of the rod cluster control assemblies, which results in a rapid shutdown of the reactor for conditions that cannot be handled by the reactor control system.

REFERENCES FOR SECTION 3.1

1. Westinghouse Proprietary, "Power Distribution Control in Westinghouse Pressurized Water Reactors," WCAP-7208, 1968.

3.2 REACTOR DESIGN

3.2.1 Nuclear Design And Evaluation

This section presents the nuclear characteristics of the core and an evaluation of the characteristics and design parameters, which are significant to design objectives. The capability of the reactor to achieve these objectives while performing safely under normal operational modes, including both transient and steady state, is demonstrated.

3.2.1.1 Nuclear Characteristics of the Design

A summary of the reactor nuclear design characteristics is presented in Table 3.2-1 for Cycle 1 and Table 3.2-1A for the current cycle. Some of the presented parameters will change from cycle to cycle depending on reload core designs.

3.2.1.1.1 Reactivity Control Aspects

Reactivity control is provided by neutron absorbing control rods and by a soluble chemical neutron absorber (boric acid) in the reactor coolant. The concentration of boric acid is varied as necessary during the life of the core to compensate for:

1. changes in reactivity, which occur with changes in temperature of the reactor coolant from cold shutdown to the hot operating, zero power conditions,
2. changes in reactivity associated with changes in the fission product poisons xenon and samarium,
3. reactivity losses associated with the depletion of fissile inventory and buildup of long-lived fission product poisons (other than xenon and samarium), and
4. changes in reactivity due to burnable poison burnup.

The control rods provide reactivity control for:

1. fast shutdown,
2. reactivity changes associated with changes in the average coolant temperature above hot zero power (core average coolant temperature is increased with power level),
3. reactivity associated with any void formation, and
4. reactivity changes associated with the power coefficient of reactivity.

3.2.1.1.1.1 Chemical Shim Control

Control to render the reactor sub-critical at temperatures below the operating range is provided by a chemical neutron absorber (boron). The boron concentration during refueling following Cycle 1 has been established as shown in Table 3.2-1, line 29. This concentration together with the control rods provides approximately 10-percent shutdown margin for these operations. The concentration is also sufficient to maintain the core shutdown without any rod cluster control rods during refueling. For cold shutdown, at the beginning of core life, a concentration (shown in Table 3.2-1, line 37) is sufficient for 1-percent shutdown with all but the highest worth rod inserted. The boron concentration (Table 3.2-1, line 29) for refueling is equivalent to less than 2-percent by weight boric acid (H_3BO_3) and is well within solubility limits at ambient temperature. This concentration is also maintained in the spent fuel pit since it is directly connected with the refueling canal during refueling operations.

The refueling boron concentration requirement for the current cycle is shown in Table 3.2-1A.

For example, the initial Cycle 1 full power boron concentration without equilibrium xenon and samarium was 1186 ppm. As the fission product poisons built up, the boron concentration was reduced to 890 ppm.

This initial boron concentration was that which permitted the withdrawal of the control banks to their operational limits. The Cycle 1 xenon-free, hot zero power shutdown (k = 0.99) with all but the highest worth rod inserted, could be maintained with the boron concentration of 677 ppm. This concentration is less than the full power operating value with equilibrium xenon.

3.2.1.1.1.2 Control Rod Requirements

Neutron-absorbing control rods provide reactivity control to compensate for more rapid variations in reactivity. The rods are divided into two categories according to their function. Some rods compensate for changes in reactivity due to variations in operating conditions of the reactor such as power or temperature. These rods comprise the control group of rods. The remaining rods, which provide shutdown reactivity, are termed shutdown rods. The total shutdown worth of all the rods is also specified to provide adequate shutdown with the most reactive rod stuck out of the core as discussed in Sections 14.1, 14.2.5, and 14.2.6.

Control rod reactivity requirements at beginning- and end-of-life for Cycle 1 are summarized in Table 3.2-2. The installed worth of the control rods for Cycle 1 is shown in Table 3.2-3. These values will vary from cycle to cycle depending on the reload core design.

The difference is available for excess shutdown upon reactor trip. The control rod requirements are discussed below.

3.2.1.1.1.3 Total Power Reactivity Defect

Control rods must be available to compensate for the reactivity change incurred with a change in power level due to the Doppler effect. The magnitude of this change was determined by measurement during the Cycle 1 Startup test program.

The average temperature of the reactor coolant is increased with power level in the reactor. Since this change is actually a part of the power dependent reactivity change, along with the Doppler effect and void formation, the associated reactivity change must be controlled by rods. The largest amount of reactivity that must be controlled is at the end-of-life when the moderator temperature coefficient has its most negative value. The Cycle 1 moderator density coefficient range is given in Table 3.2-1, line 44, while the cumulative reactivity change for Cycle 1 is shown in the first line of Table 3.2-2. By the end of the fuel cycle, the nonuniform axial depletion causes a severe power peak at low power. The reactivity associated with this peak is part of the power defect.

3.2.1.1.1.4 Operational Maneuvering Band

The control group is operated at full power within a prescribed band of travel in the core to compensate for periodic changes in boron concentration, temperature, or xenon. The band has been defined as the operational maneuvering band. When the rods reach either limit of the band, a change in boron concentration must be made to compensate for any additional change in reactivity, thus keeping the control group within the maneuvering band.

3.2.1.1.1.5 Control Rod Bite

If sufficient boron is present in a chemically shimmed core, the inherent operational control afforded by the negative moderator temperature coefficient is lessened to such a degree that the major control of transients resulting from load variations must be compensated for by control rods. The ability of the plant to accept major load variations is distinct from safety considerations, since the reactor would be tripped and the plant shut down safely if the rods could not follow the imposed load variations. In order to meet required reactivity ramp rates resulting from load changes, the control rods were inserted a given distance into the core. The reactivity worth of this insertion has been defined as control rod bite.

The reactivity insertion rate was sufficient to compensate for reactivity variation due to changes in power and temperature caused either by a ramp load change of 5-percent/min, or by a step load change of 10-percent. An insertion rate of 3×10^{-5} $\Delta\rho$/sec is determined by the transient analysis of the core and plant to be adequate for the most adverse combinations of power and moderator coefficients.

Beginning with Cycle 18, plant operation with Control Bank D at bite position is no longer required. The analysis in support of the stretch power uprate eliminated the need for control rod bite.

3.2.1.1.1.6 Xenon Stability Control

Control rods are capable of suppressing xenon-induced power oscillations in the axial direction, should they occur. Ex-core instrumentation is provided to obtain necessary information concerning power distribution. This instrumentation is adequate to enable the operator to monitor and control xenon-induced power oscillations. Extensive analyses, with confirmation of methods by spatial transient experiments at Haddam Neck, have shown that any induced radial or diametral xenon transients would die away naturally. A full discussion of xenon stability control can be found in Reference 1. In assessing potential power distribution instabilities arising from spatial xenon redistribution and in determining stability indices, primary reliance has been placed on time-dependent digital calculations in three dimensions representing feedback reactivity effects by means of semi-empirically fitted expressions whose coefficients were determined from other calculations using standard design analytical techniques and computer codes (e.g., LEOPARD code).

To assess the level of credibility and range of uncertainty attached to xenon stability analyses, conservative values of the reactivity feedback parameters were used to arrive at a reasonable upper limit for the stability index. This technique gives reasonable assurance that the reactor will in fact be stable toward diametral xenon oscillations. Means are available to increase the moderator temperature feedback term in order to stabilize the reactor response to diametral xenon oscillation (Reference 2). The reference three-dimensional time dependent digital calculations have indicated that the Indian Point Unit 2 reactor is stable against diametral xenon transients.

Cross-coupled transients are discussed in Reference 3 on the basis of full three-dimensional analyses of xenon transients.

"Second overtone" xenon transients from quadrant to quadrant (X-Y transient) are also discussed in Reference 3 in the form of radial transients. That is to say that a power-xenon perturbation is introduced by moving the center control rod. "Second overtone" xenon transients from top-to-bottom (axial transient) have been analyzed with results presented in References 1 and 4. For clarity, such transients are presented here specifically for the Indian Point Unit 2 reactor.

Figure 3.2-1 shows a cross-plot of the axial peaking factor and axial offset for transients at three points in core life. This plot is completely in agreement with such plots shown in Reference 1. In short, the ex-core detector based protection system is capable of detecting (by means of axial offset) transients, which result from perturbations to the "second axial overtone" of the power distribution.

The separation of dimensions is a conceptual artifice, which greatly facilitates analysis of xenon transients. Full three-dimensional transient analyses have been performed, as reported in References 2 and 3. Conclusions relating to power distribution stability against spatial xenon redistribution are based on results of these analyses. Cross-coupling between axial and diametral xenon oscillations are inherently accounted for in the three- dimensional time dependent calculations. Results of these calculations do not reveal any unique problems arising from cross-coupling. See Appendix 3B for additional discussion.

If the core was originally operating with a symmetric quadrant-to-quadrant power distribution, the effect of excess xenon poisoning would be to flatten the power distribution because the xenon excess would be greatest where the equilibrium power has been greatest. While this additional power flattening would tend to decrease the stability of the reactor, the analytical evaluation described in Reference 3 had already assumed a power distribution flatter than expected in the actual reactors; thus, an allowance has already been made in the Reference 3 analysis to account for this effect. Furthermore, the high xenon inventory present under the postulated conditions (i.e., maximum xenon buildup) would decrease the required boron concentration and lead to a more stable reactor response from the enhanced negative moderator reactivity feedback effect.

As burnup progresses, the required boron concentration further decreases resulting in an increasingly more stable reactor response to diametral xenon oscillations. This effect is greater than the effect of burnup on radial power flattening.

If the equilibrium power was not balanced from quadrant to quadrant, the effect of the excess xenon poisoning might be to cause the quadrant power to reverse and perhaps to increase in magnitude. If the quadrant power tilt were to reach the limit given in the Technical Specifications, the operator would take action to maintain core thermal margins.

In any event, the excess xenon, which might be present under the conditions postulated would decay naturally and cannot be regarded as a continuing source of power distribution anomalies. Similarly, a top-to-bottom power imbalance could be temporarily increased by the excess xenon poisoning. Such an imbalance cannot be a safety problem because the reactor protection system is cognizant of the axial power imbalance and if necessary will reduce trip setpoints accordingly.

3.2.1.1.1.7 Excess Reactivity Insertion Upon Reactor Trip

The control requirements are nominally based on providing 1-percent shutdown at hot, zero power conditions with the highest worth rod stuck in its fully withdrawn position or to prevent return to criticality following a credible steamline break, whichever is the more limiting. The condition where excess reactivity insertion is most critical is at the end of a cycle when the steam break accident is considered. For example, the excess control available at the end of Cycle 1, hot zero power condition with the highest worth rod stuck out is 2.08-percent $\Delta\rho$ after allowing a 10-percent margin for uncertainty in control rod worth as shown in Table 3.2-3.

3.2.1.1.1.8 Calculated Rod Worths

The complement of 53 full length control rods arranged in the pattern shown in Figure 3.2-2 meets the shutdown requirements. Table 3.2-3 lists the calculated worths of this rod configuration for beginning and end of the first cycle.

The calculated reactivity worths listed are decreased in the design by 10-percent to account for any errors or uncertainties in the calculation. This worth is established for the condition that the highest worth rod is stuck in the fully withdrawn position in the core.

A comparison between calculated and measured rod worths in the operating reactor shows the calculation to be well within the allowed uncertainty of 10-percent.

3.2.1.2 Reactor Core Power Distribution

The accuracy of power distribution calculations has been confirmed through approximately 1000 flux maps during some 20 years of operation under conditions very similar to those expected for the plant described herein. Details of this confirmation are given in Reference 5.

3.2.1.2.1 Definitions

Power distributions are quantified in terms of hot channel factors. These factors are a measure of the peak pellet power within the reactor core and the total energy produced in a coolant channel and are expressed in terms of quantities related to the nuclear or thermal design, namely:

- Power density is the thermal power produced per unit volume of the core (kW/liter).

- Linear power density is the thermal power produced per unit length of active fuel (kW/ft). Since fuel assembly geometry is standardized, this is the unit of power density most commonly used. For all practical purposes it differs from kW/liter by a constant factor, which includes geometry and the fraction of the total thermal power that is generated in the fuel rod.

- Average linear power density is the total thermal power produced in the fuel rods divided by the total active fuel length of all rods in the core.

- Local heat flux is the heat flux at the surface of the cladding (BTU-ft^{-2}-hr^{-1}). For nominal rod parameters this differs from linear power density by a constant factor.

- Rod power or rod integral power is the length integrated linear power density in one rod (kW).

- Average rod power is the total thermal power produced in the fuel rods divided by the number of fuel rods (assuming all rods have equal length).

The hot channel factors used in the discussion of power distributions in this section are defined as follows:

F_Q, Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

F_Q^N, Nuclear Heat Flux Hot Channel Factor, is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod parameters.

F_Q^E, Engineering Heat Flux Hot Channel Factor, is the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod, and eccentricity of the gap between pellet and

pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

$F^N_{\Delta H}$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

Manufacturing tolerances, hot channel power distribution and surrounding channel power distributions are treated explicitly in the calculation of the DNBR.

It is convenient for the purposes of discussion to define subfactors of F_Q. However, design limits are set in terms of the total peaking factor.

F_Q = total peaking factor (or heat flux hot-channel factor)

$$= \frac{\text{maximum kW / ft}}{\text{average kW / ft}}$$

without densification effects

$$F_Q = F^N_Q \times F^E_Q$$

$$= F^N_{XY} \times F^N_Z \times F^E_Q \times F^N_U$$

where:

F^N_Q and F^E_Q are defined above.

F^N_U = factor for conservatism, assumed to be 1.05.

F^N_{XY} = ratio of peak power density to average power density in the horizontal plane of peak local power

F^N_Z = ratio of the power per unit core height in the horizontal plane of peak local power to the average value of power per unit core height. If the plane of peak local power coincides with the plane of maximum power per unit core height, then F^N_Z is the core average axial peaking factor.

To include the allowance made for densification effects, which are height dependent, the following quantities are defined.

S(Z) = the allowance made for densification effects at height Z in the core.

P(Z) = ratio of the power per unit core height in the horizontal plane at height Z to the average value of power per unit core height.

Then:

$$F_Q = \frac{\text{maximum kW/ft}}{\text{average kW/ft}}$$

Including densification allowance

$$F_Q = \text{max} \quad [F^N_{XY} \text{ (Z) x P(Z) x S(Z) x } F^N_U \text{ x } F^E_Q]$$

3.2.1.2.2 Radial Power Distributions

The power shape in horizontal sections of the core at full power is a function of the fuel and burnable absorber loading patterns, and the presence or absence of a single bank of control rods. Thus, at any time in the cycle, a horizontal section of the core can be characterized as (1) unrodded, or (2) with group D control rods. These two situations combined with burnup effects determine the radial power shapes, which can exist in the core at full power. The effect on radial power shapes of power level, xenon, samarium, and moderator density effects are also considered but these are quite small. The effect of nonuniform flow distribution is negligible. While radial power distributions in various planes of the core are often illustrated, the core radial enthalpy rise distribution as determined by the integral of power up each channel is of greater interest. As an example, Historical Figures 3.2-3 through 3.2-6 show representative radial cycle 1 power distributions for one quarter of the core for representative operating conditions. These conditions are (1) Hot Full Power (HFP) - beginning-of-life (BOL) - unrodded - no xenon, (2) HFP-BOL - Bank D in - equilibrium xenon - unrodded, (3) HFP - end-of-life (EOL) - unrodded - equilibrium xenon, (4) HFP, BOL, no xenon, part-length rods in. Figure 3.2-6 is of historical significance only, since part-length rods have been removed.

Since the position of the hot channel varies from time to time a single reference radial design power distribution is selected for DNB calculations. This reference power distribution is chosen conservatively to concentrate power in one area of the core, minimizing the benefits of flow redistribution. Assembly powers are normalized to core average power.

3.2.1.2.3 Axial Power Distributions

The shape of the power profile in the axial or vertical direction is largely under the control of the operator through either the manual operation of the control rods or automatic motion of rods responding to manual operation of the soluble boron system. Nuclear effects, which cause variations in the axial power shape include moderator density, Doppler effect on resonance absorption, spatial xenon, and burnup. Automatically controlled variations in total power output and rod motion are also important in determining the axial power shape at any time. Signals are available to the operator from the ex-core ion chambers, which are long ion chambers outside the reactor vessel running parallel to the axis of the core. Separate signals are taken from the top and bottom halves of the chambers. The difference between top and bottom signals from each of four pairs of detectors is displayed on the control panel and called the flux difference, ΔI. Calculations of core average peaking factor for many plants and measurements from operating plants under many operating situations are associated with either ΔI or axial offset in such a way that an upper bound can be placed on the peaking factor. For these correlations axial offset is defined as:

$$\text{axial offset} = \frac{\phi t - \phi b}{\phi t + \phi b}$$

and ϕt and ϕb are the top and bottom detector readings.

3.2.1.2.4 Local Power Peaking

Fuel densification, which has been observed to occur under irradiation in several operating reactors, causes the fuel pellets to shrink both axially and radially. The pellet shrinkage combined with random hang-up of fuel pellets results in gaps in the fuel column when the pellets below the hung-up pellet settle in the fuel rod. The gaps vary in length and location in the fuel rod. Because of decreased neutron absorption in the vicinity of the gap, power peaking occurs in the adjacent fuel rods resulting in an increased power peaking factor. A quantitative measure of this local peaking is given by the power spike factor S(Z) where Z is the axial location in the core.

The method used to compute the power spike factor is described in Reference 6. Results reported in Reference 14 show that fuel manufactured by Westinghouse will not densify and therefore no power spike penalty should be included in the safety analysis

The power spike factor due to densification is assumed to be a local perturbation applicable to overpower transients. Thus, densification affects F_Q but not $F^N_{\Delta H}$. The magnitude of the increased power peaking increases from no effect at the bottom of the core to a few percent at the top of the core. For fuel produced by a process other than those for which Reference 6 is applicable, specifications will be followed to ensure that the effects of densification will be no greater than has been allowed for in the design. The specifications for qualifying the extent of densification will be based on the NRC report on fuel densification (Reference 7).

Results reported in a Westinghouse Topical Report concerning the spike penalty in LOCA analysis (Reference 8) show that the power spike penalty does not have to be included in the LOCA envelope.

3.2.1.2.5 Limiting Power Distributions

According to the ANS classification of plant conditions, Condition I occurrences are those which are expected frequently or regularly in the course of power operation, maintenance, or maneuvering of the plant. As such, Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter, which would require either automatic or manual protective action. Inasmuch as Condition I occurrences occur frequently or regularly, they must be considered from the point of view of affecting the consequences of fault conditions (Conditions II, III, and IV). In this regard, analysis of each fault condition described is generally based on a conservative set of initial conditions corresponding to the most adverse set of conditions, which can occur during Condition I operation.

Implicit in the definition of normal operation is proper and timely action by the reactor operator. That is, the operator follows recommended operating procedures for maintaining appropriate power distributions and takes any necessary remedial actions when alerted to do so by the plant instrumentation. Thus, as stated above, the worst or limiting power distribution, which can occur during normal operation is to be considered as the starting point for analysis of Condition II, III, and IV events.

Improper procedural actions or errors by the operator are assumed in the design as occurrences of moderate frequency (Condition II). Therefore, the limiting power shapes, which result from such Condition II events are those power shapes, which deviate from the normal operating condition at the recommended axial offset band, e.g., due to lack of proper action by the operator during a xenon transient following a change in power level brought about by control rod motion. Power shapes, which fall in this category are used for determination of the reactor protection system setpoints so as to maintain margin to overpower or DNB limits.

The means for maintaining power distributions within the required hot channel factor limits are described in the Technical Specifications. A complete discussion of power distribution control in Westinghouse PWRs is included in Reference 9. Detailed background information on the design constraints on local power density in a Westinghouse PWR, on the defined operating procedures, and on the measures taken to preclude exceeding design limits is presented in the Westinghouse topical report on power distribution control and load following procedures (Reference 4). The following paragraphs summarize these reports and describe the calculations used to establish the upper bound on peaking factors.

The calculations used to establish the upper bound on peaking factors, F_Q and $F^N_{\Delta H}$, include all of the nuclear effects, which influence the radial and/or axial power distributions throughout core life for various modes of operation including load follow, reduced power operation, and axial xenon transients.

Radial power distributions are calculated for the full power condition and fuel and moderator temperature feedback effects are included for the average enthalpy plane of the reactor. The steady-state nuclear design calculations are done for normal flow with the same mass flow in each channel and flow redistribution effects neglected. The effect of flow redistribution is calculated explicitly where it is important in the DNB analysis of accidents. The effect of xenon on radial power distribution is small but is included as part of the normal design process. Radial power distributions are relatively fixed and easily bounded with upper limits.

The core average axial profile, however, can experience significant changes, which can occur rapidly as a result of rod motion and load changes and more slowly due to xenon distribution. For the study of points of closest approach to axial power distribution limits, several thousand cases are examined. Since the properties of the nuclear design dictate what axial shapes can occur, boundaries in the limits of interest can be set in terms of the parameters, which are readily observed in the plant. Specifically, the nuclear design parameters, which are significant to the axial power distribution analysis are:

1. Core power level.
2. Core height.
3. Coolant temperature and flow.
4. Coolant temperature program as a function of reactor power.
5. Fuel cycle lifetimes.
6. Rod bank worths.
7. Rod bank overlaps.

Normal operation of the plant assumes compliance with the following conditions:

1. Control rods in a single bank move together with no individual rod insertion differing from the bank demand position by more than the Technical Specification limit.

2. Control banks are sequenced with overlapping banks.
3. The control bank insertion limits are not violated.
4. Axial power distribution procedures, which are given in terms of flux difference control and control bank position, are observed.

The axial power distribution procedures referred to above are part of the required operating procedures, which are followed in normal operation. Briefly they require control of the axial offset (flux difference divided by fractional power) at all power levels within a permissible operating band of a target value corresponding to the equilibrium full power value. In the first cycle, the target value changes from about +10 to -3-percent linearly through the life of the cycle. This minimizes xenon transient effects on the axial power distribution since the procedures essentially keep the xenon distribution in phase with the power distribution.

Calculations are performed for normal operation of the reactor including load following maneuvers. Beginning, middle, and end of cycle conditions are included in the calculations. Different histories of operation are assumed prior to calculating the effect of load follow transients on the axial power distribution. These different histories assume base loaded operation and extensive load following. For a given plant and fuel cycle, a finite number of maneuvers are studied to determine the general behavior of the local power density as a function of core elevation.

These cases represent many possible reactor states in the life of one fuel cycle, and they have been chosen as sufficiently definitive of the cycle by comparison with much more exhaustive studies performed on some 20 or 30 different, but typical, plant and fuel cycle combinations. The cases are described in detail in Reference 4, and they are considered to be necessary and sufficient to generate a local power density limit, which, when increased by 5-percent for conservatism, will not be exceeded with a 95-percent confidence level. Many of the points do not approach the limiting envelope. However, they are part of the time histories, which lead to the hundreds of shapes, which do define the envelope. They also serve as a check that the reactor studied is typical of those more exhaustively studied.

Thus, it is not possible to single out any transient or steady-state condition, which defines the most limiting case. It is not even possible to separate out a small number, which form an adequate analysis. The process of generating a myriad of shapes is essential to the philosophy that leads to the required level of confidence. A maneuver, which provides a limiting case for one reactor fuel cycle is not necessarily a limiting case for another reactor or fuel cycle with different control bank worths, enrichments, burnup, coefficients, etc. Each shape depends on the detailed history of operation up to that time and on the manner in which the operator conditioned xenon in the days immediately prior to the time at which the power distribution is calculated.

The calculated points are synthesized from axial calculations combined with radial factors appropriate for rodded and unrodded planes in the first cycle. In these calculations, the effects on the unrodded radial peak of xenon redistribution that occurs following the withdrawal of a control bank (or banks) from a rodded region is obtained from two-dimensional X-Y calculations. A 1.03 factor to be applied on the unrodded radial peak was obtained from calculations in which xenon distribution was preconditioned by the presence of control rods and then allowed to redistribute for several hours. A detailed discussion of this effect may be found in Reference 4. The calculated values have been increased by a factor of 1.05 for conservatism and a factor of 1.03 for the engineering factor F^E_Q.

The envelope drawn over the calculated ([F_Q x Power] max) points in Figure 3.2-7 represents an upper bound envelope on local power density versus elevation in the core for Cycles 1 through 12. For Cycles 13 and on, the anticipated normalized F_Q times power versus core height limiting K(z) curve is shown in the Unit 2 COLR. It should be emphasized that this envelope is a conservative representation of the bounding values of local power density. Expected values are considerably smaller and, in fact, less conservative bounding values may be justified with additional analysis or surveillance requirements. Additionally, Figure 3.2-7 is based on a radial power distribution invariant with core elevation.

Finally, as previously discussed, this upper bound envelope is based on procedures of load follow, which require operation within an allowed deviation from a target equilibrium value of axial flux difference. [***Note*** - Per Confirmatory Order for Indian Point Unit 2 of February 11, 1980, (letter from W. J. Cahill, Con Edison, to A. Schwencer, NRC), Indian Point Unit 2 is not operated presently in a load follow mode of operation.] These procedures are detailed in the Technical Specifications and are followed by relying only upon ex-core surveillance supplemented by the normal monthly incore core map requirement and by computer based alarms on deviation and time of deviation from the allowed flux difference band.

To determine reactor protection system setpoints with respect to power distributions, three categories of events are considered, namely rod control equipment malfunctions, operator errors of commission and operator errors of omission. In evaluating these three categories of events, the core is assumed to be operating within the four constraints described above.

The first category comprises uncontrolled rod withdrawal (with rods moving in the normal bank sequence). Also included are motions of the banks below their insertion limits, which could be caused, for example, by uncontrolled dilution or primary coolant cooldown. Power distributions are calculated throughout these occurrences assuming short term corrective action, that is, no transient xenon effects are considered to result from the malfunction. The event is assumed to occur from typical normal operating situations, which include normal xenon transients.

It is further assumed in determining the power distributions that total core power level will be limited by reactor trip to below 120-percent. Since the study is to determine protection limits with respect to power and axial off-set, no credit is taken for trip setpoint reduction due to flux difference. The peak power density, which can occur in such events, assuming reactor trip at or below 120-percent, is less than that required for centerline melt, including uncertainties and densification effects.

The second category assumes that the operator malpositions the rod bank in violation of the insertion limits and creates short-term conditions not included in normal operating conditions.

The third category assumes that the operator fails to take action to correct a flux difference violation. The resulting F_Q is multiplied by 102-percent power including an allowance for calorimetric error. It should be noted that a reactor overpower accident is not assumed to occur coincident with an independent operator error.

Analysis of possible operating power shapes shows that the appropriate hot channel factors F_Q and $F^N_{\Delta H}$ for peak local power density and for DNB analysis at full power are the values addressed in the Technical Specifications.

F_Q can be increased with decreasing power as shown in the Technical Specifications. Increasing

Increasing $F^N_{\Delta H}$ with decreasing power is permitted by the DNB protection setpoints and allows radial power shape changes with rod insertion to the insertion limits. It has been determined that Technical Specifications are met provided that, during normal operation of the plant, there is compliance with the four conditions listed earlier in this section.

When a situation is possible in normal operation, which could result in local power densities in excess of those assumed as the precondition for a subsequent hypothetical accident, but which would not itself cause fuel failure, administrative controls and alarms are provided for returning the core to a safe condition.

3.2.1.2.6 Power Distribution Anomalies

A discussion of the means provided to monitor and control power distributions anomalies caused by misplaced control rods and xenon oscillations is given in Appendix 3B.

A description of the protective function in the event of axial xenon oscillations, including calculated peaking factors and DNBRs, and the automatic trip setpoint reduction is given in References 1 and 4. Additional information on the response to ex-core ion chambers, including comparison with experimental information, is given in References 2 and 10. X-Y control rods are not required nor are they employed in the Indian Point Unit 2 reactor. A discussion of the consequences of control rod malposition is given in Appendix 3B and in Reference 2.

No automatic protective function is necessary, since even the complete misalignment of a control rod in the most limiting case (see Reference 2) cannot lead to a DNBR less than the applicable safety analysis limit at operating conditions. Furthermore, (1) rod position indicators are provided, (2) the existence of an asymmetric control rod misalignment would be revealed by the ex-core instrumentation, and (3) both asymmetric and symmetric control rod misalignments can readily be detected by the incore thermocouple system as indicated in References 2 and 10.

3.2.1.2.7 Reactivity Coefficients

The response of the reactor core to plant conditions or operator adjustments during normal operation, as well as the response during abnormal or accidental transients, is evaluated by means of a detailed plant simulation. In these calculations, reactivity coefficients are required to couple the response of the core neutron multiplication to the variables, which are set by conditions external to the core. Since the reactivity coefficients change during the life of the core, a range of coefficients is established to determine the response of the plant throughout life and to establish the design of the reactor control and protection system.

3.2.1.2.7.1 Moderator Temperature Coefficient

The moderator temperature coefficient in a core controlled by chemical shim is less negative than the coefficient in an equivalent rodded core. One reason is that control rods contribute a negative increment to the coefficient and in a chemical shim core, the rods are only partially inserted. Also, the chemical poison density is decreased with the water density upon an increase in temperature. This gives rise to a positive component of the moderator temperature coefficient due to boron being removed from the core. This is directly proportional to the amount of reactivity controlled by the dissolved poison.

In order to reduce the dissolved poison requirement for control of excess reactivity, burnable absorber rods have been incorporated in the core design. The result is that changes in the coolant density will have less effect on the density of poison and the moderator temperature coefficient will be reduced. The moderator temperature coefficient is negative at the operating coolant temperature with burnable absorber rods installed.

The original burnable absorber was in the form of borated Pyrex glass rods clad in stainless steel. The rods (1412 in the initial core) in the form of clusters were distributed throughout the initial core in vacant rod cluster control guide tubes as illustrated in Figures 3.2-8 and 3.2-9. Information regarding research, development, and nuclear evaluation of the burnable poison rods can be found in Reference 11. These rods initially controlled 7.2-percent $\Delta\rho$ of the installed excess reactivity and their addition resulted in a reduction of the initial hot zero power boron concentration in the coolant to 1318 ppm.

Starting with Cycle 8, the Wet Annular Burnable Absorber (WABA) design has been used. The WABA design is described in Section 3.2.3.2.1.5.

Starting with Cycle 11, the Integral Fuel Burnable Absorber (IFBA) design has also been used. The IFBA design is described in Section 3.2.3.2.1.5.

The effect of burnup on the moderator temperature coefficient is calculated and the coefficient becomes more negative with increasing burnup. This is due to the buildup of fission products with burnup and dilution of the boric acid concentration with burnup. The reactivity loss due to equilibrium xenon is controlled by boron, and as xenon builds up, boron is taken out. For example, the calculated net effect and the predicted unrodded moderator temperature coefficient equilibrium xenon for Cycle 1 at full power BOL was -0.55 x 10^{-4}/^{o}F. With core burnup, the coefficient became more negative as boron was removed due to the buildup of plutonium and fission products. At Cycle 1 end-of-life with no boron or rods in the core, the moderator coefficient was -3.0 x 10^{-4}/^{o}F.

Variation in moderate temperature can be seen, for example, from the Cycle 1 Figures 3.2-10 through 3.2-12.

3.2.1.2.7.2 Moderator Pressure Coefficient

The moderator pressure coefficient has an opposite sign to the moderator temperature coefficient. Its effect on core reactivity and stability is small because of the small magnitude of the pressure coefficient, a change of 50 psi in pressure having no more effect on reactivity than a one-half degree change in moderator temperature. The calculated Cycle 1 beginning and end-of-life pressure coefficients are specified in Table 3.2-1, line 43.

3.2.1.2.7.3 Moderator Density Coefficient

A uniform moderator density coefficient is defined as a change in the neutron multiplication per unit change in moderator density. The range of the moderator density coefficient for Cycle 1, for example, from BOL and EOL is specified in Table 3.2-1.

3.2.1.2.7.4 Doppler and Power Coefficients

The Doppler coefficient is defined as the change in neutron multiplication [***Note*** *- Neutron multiplication is defined as the ratio of the average number of neutrons produced by fission in*

each generation to the total number of corresponding neutrons absorbed.] per degree change in fuel temperature. The coefficient is obtained by calculating neutron multiplication as a function of effective fuel temperature. As an example, the Cycle 1 results, using the LEOPARD code (Reference 12), are shown in Figure 3.2-13.

In order to know the change in reactivity with power, it is necessary to know the change in the effective fuel temperature with power as well as the Doppler coefficient. It is very difficult to predict the effective temperature of the fuel using a conventional heat transfer model because of uncertainties in predicting the behavior of the fuel pellets. Therefore, an empirical approach is taken to calculate the power coefficient, based on operating experience of existing Westinghouse cores. As an example, Figure 3.2-14 shows the power coefficient as a function of power for Cycle 1 obtained by this method. The results presented do not include any moderator coefficient even though the moderator temperature changes with power level.

As the fuel pellet temperature increases with power, the resonance absorption in U-238 increases due to Doppler broadening of the resonances. A large temperature drop occurs across the fuel pellet-clad gap. Under certain conditions, this gap may be closed, thus resulting in lower pellet temperature. The net effect is a lower effective fuel temperature, a higher (more negative) Doppler coefficient, and a lower (less negative) power coefficient than that which exists with a pellet-clad gap. For example, the power coefficient for Cycle 1, which was determined using a closed gap model, is shown in Figure 3.2-15.

Calculations indicate the stability of the reactor to xenon oscillations is relatively insensitive to the thermal model used to obtain the power coefficient. The damping factor associated with the fuel Doppler effect is:

$$\alpha_f = \frac{\partial K_{eff}}{\partial T} \times \frac{\partial T}{\partial P}$$

where:

T = fuel temperature

P = power

The quantity $\frac{\partial T}{\partial P}$ is larger for the gap model than for the no gap case but since the Doppler coefficient varies as $T^{-1/2}$ the term $\frac{\partial K_{eff}}{\partial T}$ is smaller.

The net effect is that α_f is relatively insensitive to the thermal model in the range of power 0.5 to 1.5 of core average, which is the range of interest for stability.

3.2.1.3 Nuclear Evaluation of Current Core

Three principal computer codes have been used in the nuclear design on this reload cycle. These are: PHOENIX-P (two-dimensional), ANC (two-dimensional and three-dimensional) and APOLLO (one-dimensional). Descriptions and uses for these codes are given below:

PHOENIX-P is a two dimensional, multi-group transport theory code which utilizes a 70 energy-group cross section library. It provides the capability for cell lattice modeling on an assembly

level. In this design, PHOENIX-P is used to provide homogenized, two-group cross sections for nodal calculations and feedback models. Also, PHOENIX-P is used to generate appropriately weighted constants for the baffle/reflector regions.

ANC is an advance nodal code capable of two-dimensional and three-dimensional calculations. In this design, ANC is employed as the reference model for all safety analysis calculations, power distributions, peaking factors, critical boron concentrations, control rod worth, reactivity coefficients, etc. In addition, 3D ANC is used to validate one and two-dimensional results and to provide information about radial (x-y) peaking factors as a function of axial position. It has the capability of calculating discrete pin powers from the nodal information as well.

APOLLO, an advanced version of PANDA, is a two group, one-dimensional diffusion-depletion code. It uses cross sections generated by a radial averaging of the corresponding 3D model cross sections and is used as a one-dimensional axial model. Thermal feedback is included in the calculational models. The axial model is used for computing axial power distributions, differential rod worths, control rod operating limits (insertion limits, return to power limits), etc.

Additional support codes are used for special calculations such as determining fuel temperatures.

3.2.2 Thermal and Hydraulic Design and Evaluation

[***Note*** *- A large amount of material has been retained as historical background.*]

3.2.2.1 Thermal and Hydraulic Characteristics of the Design

3.2.2.1.1 Central Temperature of the Hot Pellet

The temperature distribution in the pellet is mainly a function of the uranium-dioxide thermal conductivity and the local power density. The absolute value of the temperature distribution is affected by the cladding temperature and the thermal conductance of the gap between the pellet and the cladding.

The gap conductance model is selected such that when combined with the UO_2 thermal conductivity model, the calculated fuel centerline temperatures reflect the inpile temperature measurements. A more detailed discussion of the gap conductance model has been provided in References 83 and 84. The temperature drop across the gap is calculated by assuming an annular gap conductance model of the following form:

$$h = \frac{K_{gas}}{\frac{\delta}{2} + \delta_r}$$

where:

h = contact conductance, Btu/hr-ft^2-^{o}F

K_{gas} = thermal conductivity of the gas mixture including a correction factor (Reference 25) for the accommodation coefficient for light gases, e.g., helium, Btu/hr-ft-°F

δ = diametral gap size, ft

δ_r = effective gap spacing due to surface roughness, ft

or an empirical correlation derived from thermocouple and melt radius data.

The larger gap conductance value from the equation above and the empirical correlation is used to calculate the temperature drop across the gap for finite gaps.

For evaluations in which the pellet-clad gap is closed, a contact conductance is calculated. The contact conductance between UO_2 and zircaloy has been measured and found to be dependent on the contact pressure, composition of the gas at the interface, and the surface roughness (References 25 and 26). This information, together with the surface roughness found in Westinghouse fuel, leads to the following correlation:

$$h = 0.6P + \frac{K_{gas}}{\delta_r}$$

δ_r	=	effective gap spacing due to surface roughness, ft
h	=	contact conductance, Btu/hr-ft^2-^{o}F
P	=	contact pressure, psi
K_{gas}	=	thermal conductivity of gas mixture at the interface including a correction factor (Reference 25) for the accommodation coefficient for light gases, e.g., helium, Btu/hr-ft-°F

The thermal conductivity of uranium-dioxide was evaluated from data reported by Howard, et al.,[27] Lucks, et al.,[28] Daniel, et al.,[29] Feith,[30] Vogt, et al.,[31] Nishijima, et al.,[32] Ainscough, et al.,[33] Godfrey, et al.,[34] Stora, et al.,[35] Bush,[36] Asamoto, et al.,[37] Kruger,[38] and Gyllander.[39]

At the higher temperatures, thermal conductivity is best obtained by utilizing the integral conductivity to melt, which can be determined with more certainty. From an examination of the data, it has been concluded that the best estimate for the value of $\int_0^{2800°C} Kdt$ is 93 watts/cm. This conclusion is based on the integral values reported by Gyllander,[39] Lyons, et al.,[40] Duncan,[41] Bain,[42] and Stora.[43]

The design curve for the thermal conductivity is shown in Figure 3.2-38. The section of the curve at temperatures between 0^{o}C and 1300^{o}C is in excellent agreement with the recommendation of the IAEA panel.[44] The section of the curve above 1300^{o}C is derived for an integral value of 93 watts/cm.[39,41,43]

Thermal conductivity for UO_2 at 95-percent theoretical density can be represented best by the following equation:

$$K = \frac{1}{11.8 + 0.0238T} + 8.775 \times 10^{-13} T^3$$

where:

K = watt/cm$^\circ$C
T = $^\circ$C

Based upon the above considerations, the maximum central temperature of the hot pellet at steady state is shown in Table 3.2-6. This temperature is well below the melting temperatures of the irradiated UO_2, which is taken as 5080°F (Reference 45) unirradiated and decreasing by 58°F per 10,000 MWd/metric ton uranium.

Westinghouse experience with fuel rods operating at high power ratings has been summarized in Appendix A, Indian Point No. 2 Preliminary Safety Analysis Report (Docket 50-247) and in Appendix-Section IX of the Preliminary Safeguards Report for the Saxton Reactor Operating at 35 MWt (Docket 50-146). These reports presented considerable statistical evidence of successful operation of high performance zircaloy clad fuel rods in CVTR (1368 rods) and Shipping port core I Blanket (94,920 rods). Since the date of these reports, a significant amount of additional information has been developed relating to the integrity of free standing zircaloy-clad oxide fuel rods at high power ratings. In addition, a comprehensive experimental program was initiated to extend the operating experience to higher power and to higher exposures for many of these fuel rods. This information is summarized in Figure 3.2-39.

More detailed information about Westinghouse experience with high power fuel rod bars has been provided in Reference 46.

3.2.2.1.2 Heat Flux Ratio and Data Correlation

Departure from nucleate boiling (DNB) is predicted upon a combination of hydrodynamic and heat transfer phenomena and is affected by the local and upstream conditions including the flux distribution. In reactor design, the heat flux associated with DNB and the location of DNB are both important. The W-3 based L-grid DNB correlation was used in design of the LOPAR fuel assemblies. The WRB-1 DNB correlation, Reference 76, is the primary DNB correlation for the analysis of the optimized and VANTAGE+ fuel assemblies. The W-3 DNB correlation[47] was developed to predict the DNB flux and the location of DNB equally well for uniform and an axially nonuniform heat flux distribution. This correlation replaced the WAPD q" and ΔH DNB correlations published in Nucleonics,[48] May 1963, in order to eliminate the discontinuity of the latter at the saturation temperature and to provide a single unambiguous criterion of the design margin.

The W-3 correlation, and several modifications of it, have been used in Westinghouse critical heat flux (CHF) calculations. The W-3 correlation was originally developed from single tube data,[49] but was subsequently modified to apply to the "L" grid[50] rod bundle data. These modifications to the W-3 correlation have been demonstrated to be adequate for reactor rod bundle design.

The W-3 DNB correlation[47] incorporates both local and system parameters in predicting the local DNB heat flux. This correlation includes the nonuniform flux effect and the upstream effect, which includes inlet enthalpy or length. The local DNB heat flux ratio (defined as the ratio of the DNB heat flux to the local heat flux) is indicative of the contingency available in the local heat flux without reaching DNB.

The sources of the data used in developing the correlation were:

WAPD-188	(1958)	CU-TR-No. I (NW-208)	(1964)
ASME Paper 62-WA-297	(1962)	CISE-R-90	(1964)
CISE-R-63	(1962)	DP-895	(1964)
ANL-6675	(1962)	AEEW-R-356	(1964)

GEAP-3766	(1962)	BAW-3238-7	(1965)
AEEW-R-213 and 309	(1963)	AE-RTL-778	(1965)
CISE-R-74	(1963)	AEEW-355	(1965)
CU-MPR-XIII	(1963)	EUR-2490.e	(1965)

The comparison of the W-3 measured to predicted DNB flux of this correlation is given in Figure 3.2-40. The local flux DNB ratio versus the probability of not reaching DNB is plotted in Figure 3.2-41. This plot indicates that with a DNBR of 1.3 the probability of not reaching DNB is 95-percent at a 95-percent confidence level. The comparison of the "L"-grid measured to predicted DNB flux is given in Figure 3.2-42.

Rod bundle data without mixing vanes agree very well with the predicted DNB flux as shown in Figure 3.2-43. Rod bundle data with mixing vanes (Figure 3.2-44) show on the average an 8-percent higher value of DNB heat flux than predicted by the W-3 DNB correlation. The L-grid modified spacer factor has been formulated to reflect the improvement in DNB heat flux due to the presence of mixing vanes.

It should be emphasized that the inlet subcooling effect of the W-3 correlation was obtained from both uniform and non-uniform data. The existence of an inlet subcooling effect has been demonstrated to be real and hence the actual subcooling should be used in the calculations. The W-3 correlation was developed from tests with flow in tubes and rectangular channels. Good agreement was obtained when the correlation is applied to test data for rod bundles.

3.2.2.1.3 Definition of Departure From Nucleate Boiling Ratio

The DNB heat flux ratio (DNBR) as applied to the design when all flow cell walls are heated is:

$$DNBR = \frac{q''_{DNB,N} \ x \ F'_{S'} \ x \ 0.986}{q''_{loc}} \qquad \text{(for LOPAR Fuel)}$$

The DNB heat flux ratio (DNBR) as applied to typical cells (flow cells with all walls heated) and thimble cells (flow cells with heated and unheated walls) is defined as:

$$DNBR = \frac{q''_{DNB,N}}{q''_{loc}}$$

where:

$$q''_{DNB,N} = \frac{q''_{DNB,EU}}{F}$$

and $q''_{DNB,EU}$ is the uniform DNB heat flux as predicted by the WRB-1 DNB Correlation and the W-3 DNB correlation (Reference 47) when all flow cells are heated. The flux shape factor to account for nonuniform axial heat flux distributions is F (Reference 47) with the "C" term modified as in Reference 49.

F_s is the modified spacer factor, which uses an axial grid spacing coefficient, K_S, and a thermal diffusion coefficient, TDC, based on the 20-in. grid spacing data.

q''_{loc} is the actual local heat flux.

The DNBR as applied to the W-3 DNB correlation when a cold wall is present is:

$$DNBR = \frac{q''_{DNB,N,CW}}{q''_{loc}}$$

where:

$$q''_{DNB,N,CW} = \frac{q''_{DNB,EU,Dh} \times CWF}{F}$$

$q''_{DNB,EU,Dh}$ is the uniform DNB heat flux as predicted by the W-3 cold wall DNB correlation (Reference 49) when not all flow cell walls are heated (thimble cold wall cell).

W-3 Cold Wall Factor:

$$CWF = \frac{q''_{coldwall}}{q''_{W-3,D_h}} = 1.0 - R_u\{13.76 - 1.372e^{1.78\chi} - 4.732\left(\frac{G}{10^6}\right)^{-0.0535}$$

$$- 0.0619\left(\frac{P}{1000}\right)^{0.14} - 8.509 D_h^{0.107}]$$

where:

$R_u = (D_h - D_e / D_h)$, q''_{W-3,D_h} = prediction of W - 3 correlation using D_h for D_e

The equivalent uniform DNB flux $q'_{DNB,EU}$ is calculated for the W-3 equivalent uniform flux DNB correlation as follows:

$$\frac{q''_{DNB,EU}}{10^6} = \left[(2.022 - 0.0004302p) + (0.1722 - 0.0000984p)\, e^{(18.177 - 0.004129\rho)\,\chi}\right]$$

$$\times \left[1.037 + \frac{G}{10^6}(0.1484 - 1.596\chi + 0.1729\chi\,|\chi|)\right] \times [1.157 - 0.869\chi]$$

$$\times[0.2664 + 0.8357e^{-3.151De}] \times [0.8258 + 0.000794\,(H_{sat} - H_{in})]$$

The heat flux is in Btu/hr-ft^2 and the units of the parameters are as listed below. The ranges of parameters for applicability of the W-3 DNB correlation are:

System pressure, p = 500 to 2400 psia
Mass velocity, G = 1.0 x 10^6 to 5.0 x 10^6 lb/hr ft^2
Equivalent diameter, D_e = 0.2 to 0.7-in.
Quality, $_{loc}$ = -0.15 to +0.15
Inlet enthalpy, no limit, Btu/lb
Length, L = 10 to 168-in.

$$\frac{\text{Heated perimeter}}{\text{Wetted perimeter}} = 0.88 \text{ to } 1.00$$

Geometries - circular tubes and rectangular channels

Flux = Uniform and non-uniform heat flux converted from non-uniform data by using F-factor of Reference 47.

For the LOPAR fuel, which uses the "L"-grid modified W-3 DNB correlation with the modified spacer factor (Reference 50):

$$F'_S = (1.445 - .0371L)\left(\frac{P}{225.896}\right)^{.5}\left(e^{(\chi + .2)^2} - .73\right) + K_S \frac{G}{10^6}\left(\frac{TDC}{.019}\right)^{.35}$$

Where: K_S = spacer factor dependent on grid type and axial spacing

P = inlet pressure, psia

G = inlet mass velocity, lbm/hr-ft^2

TDC = thermal diffusion coefficient

L = test length, ft

. = local quality, fraction

The ranges of the parameters listed above are:

$1460 \leq P \leq 2430$ psia
$1.96 \times 10^6 \leq G \leq 3.68 \times 10^6$ lbm/hr-ft^2
$-0.15 \leq \chi \leq +0.15$ fraction
$8 \leq L \leq 14$-ft

Geometries ("L"-grid correlation) - circular tubes and rectangular channels

Flux = Various nonuniform heat fluxes

Local Nonuniform DNB Flux

The local nonuniform $q''_{DNB,N}$ is calculated as follows:

$$q''_{DNB,N} = \frac{q''_{DNB,EU}}{F}$$

where:

$$F = \frac{C}{q''_{\text{local at } \ell_{DNB}} \times (1 - e^{-C\ell_{DNB}})} \int_0^{\ell_{DNB}} q''(z) e^{-C(\ell_{DNB} - z)} dz$$

ℓ_{DNB} = distance from the inception of local boiling to the point of DNB, in inches.
Z = distance from the inception of local boiling measured in the direction of flow, in inches.

$$C = 0.15 \frac{(1 - \chi_{DNB})^{4.31}}{(G/10^6)^{0.478}} \text{ in.}^{-1} \qquad \text{(Reference 49)}$$

Where:

G = mass velocity, lb/hr-ft^2

χ_{DNB} = quality of the coolant at the location where DNB flux is calculated

In determining the F-factor, the value of q''_{local} at ℓ_{DNB} in the above equation for the F-factor was measured as z = ℓ_{DNB}, the location where the DNB flux is calculated. For a uniform flux, F becomes unity so that $q''_{DNB,N}$ reduces to $q''_{DNB,EU}$ as expected. The criterion for determining the predicted location of DNB is to evaluate the ratio of the predicted DNB flux to the local heat flux along the length of the channel. The location of the minimum DNB ratio is considered to be location of DNB.

3.2.2.1.4 Procedure for Using W-3 L-grid Correlation

In predicting the local DNB flux in a nonuniform heat flux channel, the following two steps are required:

1. The uniform DNB heat flux, $q''_{DNB,EU}$, is computed with the W-3 L-grid correlation using the specified local reactor conditions.

2. This equivalent uniform heat flux is converted into corresponding nonuniform DNB heat flux, $q''_{DNB,N}$, for the nonuniform flux distribution in the reactor. This is accomplished by dividing the uniform DNB flux by the F-factor.[47] Since F is generally greater than unity, $q''_{DNB,N}$ will be smaller than $q''_{DNB,EU}$.

To calculate the DNBR of a reactor channel, the values of

$$\frac{q''_{DNB,N}}{q''_{loc}}$$

along the channel are evaluated and the minimum value is selected as the minimum DNBR incurred in that channel.

The W-3 L-grid correlation depends on both local and inlet enthalpies of the actual system fluid, and the upstream conditions are accommodated by the F-factor. Hence, the correlation provides a realistic evaluation of the safety margin on heat flux.

3.2.2.1.5 The WRB-1 DNB Correlation

The WRB-1[76] correlation was developed based exclusively on the large bank of mixing vane grid rod bundle critical heat flux data (in excess of 1100 points) that Westinghouse has collected. The WRB-1 correlation, based on local fluid conditions, represents the rod bundle data with better accuracy over a wide range of variables than the previous correlation used in design (namely the W-3 correlation). This correlation accounts directly for both typical and thimble cold wall cell effects, uniform and non-uniform heat flux profiles, and variations in rod heated length and in grid spacing.

The applicable range of variables is:

Pressure	$1440 \leq P \leq 2490$ psia
Local Mass Velocity	$0.9 \leq G_{loc}/10^6 \leq 3.7$ lb/ft^2-hr
Local Quality	$-0.2 \leq \chi_{loc} \leq 0.3$
Heated Length, Inlet to CHF Location	$L_h \leq 14$ feet
Grid Spacing	$13 \leq g_{sp} \leq 32$ inches
Equivalent Hydraulic Diameter	$0.37 \leq d_e \leq 0.60$ inches
Equivalent Heated Hydraulic Diameter	$0.46 \leq d_h \leq 0.68$ inches

Figure 3.2-44A shows measured critical heat flux plotted against predicted critical heat flux using the WRB-1 correlation.

A correlation limit DNBR of 1.17 for the WRB-1 correlation has been approved by the NRC for the 15x15 fuel.

3.2.2.1.6 The W-3 DNB Correlation

The W-3 DNB correlation[47, 49 and 95] is used for both fuel types where the primary DNB correlation is not applicable. The WRB-1 correlation is developed based on mixing vane data and therefore is only applicable in the heated rod spans above the first mixing vane grid. The W-3 correlation, which does not take credit for mixing vane grids, is used to calculate DNBR values in the heated region below the first mixing vane grid. In addition, the W-3 correlation is applied in the analysis of accident conditions where the system pressure is below the range of the primary correlation. For system pressure in the range of 500 to 1000 psia, the W-3 correlation is 1.45 [89]. For system pressure greater that 1000 psia, the W-3 correlation is limited to 1.30. A cold wall factor[50] is applied to the W-3 DNB correlation to account for the pressure of the unheated thermal surface.

3.2.2.1.7 Film Boiling Heat Transfer Coefficient

Heat transfer after departure from nucleate boiling was conservatively assumed to be limited by film boiling immediately, and the period of transition boiling neglected.

The correlation used to evaluate these film boiling heat transfer coefficients was developed by Tong, Sandberg and Bishop[51] and is shown in Figure 3.2-45.

$$\left(\frac{hD}{k}\right)_f = 0.0193 \left(\frac{DG}{\mu}\right)_f^{0.80} \left(\frac{C_p\,\mu}{k}\right)_f^{1.23} \left(\frac{\rho_g}{\rho_b}\right)^{0.68} \left(\frac{\rho_g}{\rho_\ell}\right)^{0.068}$$

where: $\rho_b = \rho_g\alpha + \rho_\ell(I - \alpha)$ and

C_P = heat capacity at constant pressure, Btu/lb-^{o}F
D = equivalent diameter of flow channel, ft
h = heat transfer coefficient, Btu/hr-ft^2-^{o}F
G = mass flow rate, lb/hr-ft^2
k = thermal conductivity, Btu/hr-ft-^{o}F
α = void fraction
ρ = density, lbs/ft^3
μ = viscosity, lbs/ft-hr

Subscripts:

g = Evaluation of the property at the saturated vapor condition
ℓ = Evaluation of the property at the saturated liquid condition
f = Evaluation of the property at the average film temperature
w = Evaluation of the property at the wall temperature
b = Evaluation of the property at the average bulk fluid condition

The heat transfer correlation was developed for flow rates equal or greater than 0.8 x 10^6 lb/hr ft^2 over a pressure range of 580 to 3190 psia, for qualities as high as 100-percent, and heat flux from 0.1 to 0.65 x 10^6 Btu/hr ft^2.

3.2.2.2 Hot Channel Factors

The total hot channel factors for heat flux and enthalpy rise are defined as the maximum-to-core average ratios of these quantities. The heat flux factor considers the local maximum linear heat generation rate at a point (the "hot spot"); the enthalpy rise hot channel factor involves the maximum integrated value along a channel (the "hot channel").

3.2.2.2.1 Definition of Engineering Hot Channel Factor

Each of the total hot channel factors is composed of a nuclear hot channel factor describing the neutron flux distribution and an engineering hot channel factor, which allows for variations in flow conditions and fabrication tolerances. The engineering hot channel factors are made up of subfactors, which account for the influence of the variations of fuel pellet diameter, density, enrichment and eccentricity; fuel rod diameter; inlet flow distribution; flow redistribution; and flow mixing.

3.2.2.2.2 Heat Flux Engineering Subfactor, F_Q^E

The heat flux engineering hot channel factor is used to evaluate the maximum linear heat generation rate in the core. This subfactor is determined by statistically combining the fabrication variations for fuel pellet diameter, density, enrichment and variation in fuel rod diameter, and has a value of 1.03 to be applied to the fuel rod surface heat flux. As shown in Reference 52, no DNB penalty need be taken for the short relatively low intensity heat flux spikes caused by variations in the above parameters or fuel pellet eccentricity

3.2.2.2.3 Enthalpy Rise Engineering Subfactor, $F_{\Delta H}^E$

The effect of variations in flow conditions and fabrication tolerances on the hot channel enthalpy rise in reload analysis is directly considered in the Westinghouse version of VIPRE-01 Code [100,101] thermal subchannel analysis under any reactor operating condition (refer to Section 3.2.2.4, DNB Analysis Method for VIPRE description). The items presently considered contributing to the enthalpy rise engineering hot channel factor are discussed below:

1. Pellet diameter, density and enrichment

   Variation in pellet diameter, density and enrichment are considered, statistically, in establishing the limit DNBR's (see section 3.2.2.4) for the Revised Thermal

Design Procedure[97] employed in this application. Uncertainties in the variables are determined from sampling of manufacturing data.

2. Inlet Flow Maldistribution

Data have been considered from several one-seventh scale hydraulic reactor model tests (References 53, 54, and 55) in arriving at the core inlet flow maldistribution criteria to be used in the THINC[56] analyses. THINC analyses made, using these data, have indicated that a conservative design basis is to consider a 5-percent reduction in the flow to the hot assembly (Reference 78). The design basis of 5% flow reduction to the hot assembly is also used in the VIPRE analysis for the 1.4% power uprate.

3. Flow Redistribution

The flow redistribution accounts for the reduction in flow in the hot channel resulting from the high flow resistance in the channel due to the local or bulk boiling. The effect of the nonuniform power distribution is inherently considered in the VIPRE analysis for every operating condition, which is evaluated.

4. Flow Mixing

The mixing vanes incorporated in the spacer grid design induce additional flow mixing between the various flow channels in a fuel assembly as well as between adjacent assemblies. This mixing reduces the enthalpy rise in the hot channel resulting from local power peaking or unfavorable mechanical tolerances. The subchannel mixing model now incorporated in the VIPRE code and used in reload reactor design is based on experimental data (Reference 57).

3.2.2.3 Core Pressure Drop and Hydraulic Loads

Core and vessel pressure losses are calculated by equations of the form:

$$\Delta P_L = \left(K + F\frac{L}{D_e}\right)\frac{\rho V^2}{2\,g_c\,(144)}$$

where:

ΔP_L = unrecoverable pressure drop, $lb_f/in.^2$
ρ = fluid density, lb_m/ft^3
L = length, ft
D_e = equivalent diameter, ft
V = fluid velocity, ft/sec
g_c = 32.174, $lb_m\text{-}ft/lb_f\text{-}sec^2$
K = form loss coefficient, dimensionless
F = friction loss coefficient, dimensionless

Fluid density is assumed to be constant at the appropriate value for each component in the core and vessel. Because of the complex core and vessel flow geometry, precise analytical values for the form and friction loss coefficients are not available. Therefore, experimental values for these coefficients are obtained from geometrically similar models.

The results of full scale tests of core components and fuel assemblies are utilized in developing the core pressure loss characteristic in reload reactor design. The pressure drop for the vessel has been obtained by combining the core loss with correlation of one-seventh scale model hydraulic test data on a number of vessels (References 53 and 54) and form loss relationships (Reference 58). Moody (Reference 59) curves have been used to obtain the single phase friction factors.

The fuel assembly hold-down springs are designed to keep the fuel assemblies in contact with the lower core plate under all Condition I and II events, with the exception of the turbine overspeed transient associated with a loss of external load. The hold-down springs are designed to tolerate the possibility of an over deflection associated with fuel assembly liftoff for this case and provide contact between this transient. More adverse flow conditions occur during a loss-of-coolant accident. Hydraulic loads at normal operating conditions are calculated considering the best estimate flow and accounting for the minimum core bypass flow based on manufacturing tolerances. Core hydraulic loads at cold plant startup conditions are based on the best estimate flow, but are adjusted to account for the coolant density difference. Conservative core hydraulic loads for a pump overspeed transient, which could possibly create flow rates 20-percent greater than the best estimate flow, are evaluated to be approximately twice the fuel assembly weight. The hydraulic forces are not sufficient to lift a rod control cluster during normal operation even if the rod cluster is detached from its coupling.

3.2.2.4 Thermal and Hydraulic Design Parameters

The thermal and hydraulic design parameters are given in Table 3.2-6, Sheets 1-3.
Sheet 3 shows parameters over a range of vessel average temperatures, giving flexibility to operate at full licensed power at various plant operating conditions.

DNB Design Basis

There will be at least a 95-percent probability that departure from nucleate boiling (DNB) will not occur on the limiting fuel rods during normal operation and operational transients and any transient conditions arising from faults of moderate frequency (Condition I and II events), at a 95-percent confidence level. Historically, this has been conservatively met by adhering to the following thermal design basis: there must be at least a 95-percent probability that the minimum departure from nucleate boiling ratio (DNBR) of the limiting power rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used. The DNBR limit for the correlation is established based on the variance of the correlation such that there is a 95-percent probability with 95-percent confidence that DNB will not occur when the calculated DNBR is at the DNBR limit.

DNB Analysis Method

The THINC IV [77, 78] computer program, beginning in Cycle 10, was used to perform the thermal/hydraulic calculations for both fuel types. The THINC IV code is used to calculate coolant density, mass velocity, enthalpy, void fractions, static pressure and DNBR distributions along flow channels within a reactor core under all expected operating conditions. References 77 and 78 contain details of the THINC IV computer program, including models and correlations used. The Westinghouse version of the VIPRE-01 (VIPRE) code is used. The VIPRE code is equivalent to the THIC-VI (THINC) code and has been approved by the NRC for licensing applications to replace the THINC code. The use of VIPRE is in full compliance with the

conditions specified in the NRC Safety Evaluation Report (SER) on WCAP-14565-P-A (Reference 101). The design method employed for both fuel types to meet the DNB design basis is the Revised Thermal Design Procedure.[97] Uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95-percent probability that the minimum DNBR will be greater than or equal to the limit DNBR for the limiting power rod. Plant parameter uncertainties are used to determine the plant DNBR uncertainty. The DNBR uncertainty, combined with the DNBR limit, establishes a design DNBR value, which must be met in plant safety analyses. Since the parameter uncertainties are considered in determining the design DNBR value, the plant safety analyses are performed using values of input parameters without uncertainties. In addition, the limit DNBR values are increased to values designated as the safety analysis limit DNBR's. The plant allowances available between the safety analysis limit DNBR values and the design limit DNBR values is not required to meet the design basis. This allowance will be used for flexibility in the design and operation of this plant.

For this design, the WRB-1 correlation is used for analysis of the Vantage+ fuel assemblies with a correlation limit of 1.17 (both typical and thimble cells).

The design method employed for both fuel types to meet the DNB design basis is the Revised Thermal Design Procedure (RTDP)[97]. With the RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes and DNB correlation predictions are considered statistically to obtain DNB uncertainty factors. Based on the DNB uncertainty factors, RTDP design limit DNBR values are determined such that there is at least a 95% probability at a 95% confidence level that DNB will not occur on the most limiting fuel rod during normal operation and operational transients and during transient conditions arising from faults of moderate frequency (Condition I and II events as defined in ANSI N18.2)

To maintain DNBR margin to offset DNB penalties such as those due to fuel rod bow (see section 3.2.2.6) and potential transition core (see 3.2.2.5.1), the safety analyses were performed to DNBR limits higher than the design limit DNBR values. The difference between the design limit DNBRs and the safety analysis limit DNBRs results in available DNBR margin. The net DNBR margin, after consideration of all penalties, is available for operating and design flexibility. The Standard Thermal Design Procedure (STDP) is used for those analyses where RTDP is not applicable. In the STDP method the parameters used in analysis are treated in a conservative way from a DNBR standpoint. The parameter uncertainties are applied directly to the plant safety input values to give the lowest minimum DNBR. The DNBR limit for STDP is the appropriate DNB correlation limit increased by sufficient margin to offset the applicable DNBR penalties.

For this design, the WRB-1 correlation is used for analysis of both fuel types with a correlation limit of 1.17 (both typical and thimble cells). When the core condition is outside the range of the WRB-1 correlation, the W-3 correlation is applied with a correlation limit of 1.30 (both cell types).

DNB With Physical Burnout

Westinghouse[60] has conducted DNB tests in a 25-rod bundle where physical burnout occurred with one rod. After this occurrence, the 25-rod test section was used for several days to obtain more DNB data from the other rods in the bundle. The burnout and deformation of the rod did not affect the performance of neighboring rods in the test section during the burnout or the

validity of the subsequent DNB data points as predicted by the W-3 correlation. No occurrences of flow instability or other abnormal operations were observed.

DNB With Return to Nucleate Boiling

Additional DNB tests have been conducted by Westinghouse[61] in 19 and 21 rod bundles. In these tests, DNB without physical burnout was experienced more than once on single rods in the bundles for short periods of time. Each time, a reduction in power of approximately 10-percent was sufficient to reestablish nucleate boiling on the surface of the rod. During these and subsequent tests, no adverse effects were observed on this rod or any other rod in the bundle as a consequence of operating in DNB.

Hydrodynamic and Flow Power Coupled Instability

Thermohydrodynamic instabilities will not occur under Condition I and II modes of operation for Westinghouse PWR reactor designs. A large power margin exists to predicted inception of such instabilities. Analysis has been performed which shows that minor plant to plant differences in Westinghouse reactor designs such as fuel assembly arrays, core power to flow ratios and fuel assembly length will not result in gross deterioration of the above power margins.

3.2.2.5 Hydraulic Compatibility

3.2.2.5.1 Transition Core Effects

The entire core is now 15x15 Upgraded fuel therefore there are no transition core effects.

3.2.2.5.2 DNB Performance When Transitioning Cores

The Westinghouse transition core DNB methodology is given in References 90, 91, and 92 and has been approved by the NRC via Reference 93 and 94. Using this methodology, transition cores are analyzed as if the entire core consists of one assembly type.

3.2.2.5.3 Compatibility

The hydraulic resistance of the two assemblies is based on full scale hydraulic flow test data. The design hydraulic loss coefficients are verified with a confirmatory hydraulic test in the Fuel Assembly Compatibility Test System (FACTS). The results are evaluated to determine the values of the pressure drop loss coefficients.

Fuel assembly vibration testing is conducted to confirm that the fuel assembly does not experience flow-induced vibration. Hydraulic vibration tests are performed in the FACTS test loop.

Side-by-side VIPER tests are performed. Test results are used to demonstrate the acceptable fretting behavior design between the two designs is insignificant.

A crossflow analysis using the THINC code is completed to determine the crossflow velocity profile in the transition core in the IP2 plant conditions. The crossflow is caused by the mid-grid and IFM pressure drop mismatch between adjacent fuel assemblies.

3.2.2.6 Effects of Rod Bow on DNBR

The phenomenon of fuel rod bowing, as described in Reference 80 must be accounted for in the DNBR safety analysis of Condition I and Condition II events for each plant application. Applicable generic credits for margin resulting from retained conservatisms in the evaluation of DNBR can be used to offset the effects of rod bow.

For safety analysis of Indian Point Unit 2, sufficient margin was maintained in the design of the fuel [***Note** - Margin maintained between design limit DNBR and safety analysis limit DNBR +As a result of analyses performed for OFA transition, maintaining plugging devices in core is optional.*] to accommodate full and low flow rod bow DNBR penalties (less than 1-percent for the worst cast, which is at a burnup of 24,000 MWd/MTU identified in Reference 81) with the incorporation of the L^2/I scaling factor (I=bending moment of inertia, L=span length) to account for the 9-grid fuel span lengths. The rod bow DNBR penalties in the Intermediate Flow Mixer (IFM) grid spans are less than those in the mixing vane grid spans.

The maximum rod bow penalties accounted for in the design safety analysis are based on an assembly average burnup of 24,000 MWd/MTU. At burnups greater than 24,000 MWd/MTU, credit is taken for the effect of F-delta-H burndown, due to the decrease in fissionable isotopes and the buildup of fission product inventory, and no additional rod bow penalty is required.

3.2.3 Mechanical Design And Evaluation

The reactor core and reactor vessel internals are shown in cross-section in Figure 3.2-46 and in elevation in Figure 3.2-47. The core, consisting of the fuel assemblies, control rods, source rods, burnable poison rods, and plugging devices+, provides and controls the heat source for the reactor operation.

The internals, consisting of the upper and lower core support structure, are designed to support, align, and guide the core components, direct the coolant flow to and from the core components, and to support and guide the incore instrumentation. A listing of the core mechanical design parameters is given in Table 3.2-7.

The fuel assemblies are arranged in a checkerboard and/or roughly circular zoned pattern. The fuel assemblies contain fuel of different enrichments depending on the location of the assembly within the core.

The fuel is in the form of slightly enriched uranium-dioxide ceramic pellets. The pellets are stacked to an active height of 144-in. (previously 142 in.) within ZIRLO™ (previously Zircaloy-4) tubular cladding, which is plugged and seal welded at the ends to encapsulate the fuel. The enrichments of the fuel for the first three regions in the core are given in Table 3.2-7. Reload fuel enrichment may vary up to the maximum value allowed in the Technical Specifications. Heat generated by the fuel is removed by demineralized light water, which flows upward through the fuel assemblies and acts as both moderator and coolant.

The core is divided into fuel assembly regions of different enrichments. The loading arrangement for the initial cycle is indicated on Figure 3.2-48. In the past refueling took place generally in accordance with an inward loading schedule. Starting from Cycle 6 a low leakage loading pattern for core refueling design has been adopted and starting from Cycle 13, a low, low leakage loading pattern was used. This will reduce neutron fluence at the reactor vessel wall.

The control rods, designated as rod cluster control assemblies, consist of groups of individual absorber rods, which are held together by a spider assembly at the top end and actuated as a group. In the inserted position, the absorber rods fit within hollow guide thimbles in the fuel assemblies. The guide thimbles are an integral part of the fuel assemblies and occupy locations within the regular fuel rod pattern where fuel rods have been deleted. In the withdrawn position, the absorber rods are guided and supported laterally by guide tubes, which form an integral part of the upper core support structure. Figures 3.2-49 and 3.2-50 show a typical rod cluster control assembly in a fuel assembly. As shown in Figure 3.2-47, the fuel assemblies are positioned and supported vertically in the core between the upper and lower core plates. The core plates are provided with pins, which index into closely fitting mating holes in the fuel assembly top and bottom nozzles. The pins maintain the fuel assembly alignment, which permits free movement of the control rods from the fuel assembly into the guide tubes in the upper support structure without binding or restriction between the rods and their guide surfaces.

Operational or seismic loads imposed on the fuel assemblies are transmitted through the core plates to the upper and lower support structures and ultimately to the internals support ledge at the pressure vessel flange in the case of vertical loads or to the lower radial support and internals support ledge in the case of horizontal loads. The internals also provide a form fitting baffle surrounding the fuel assemblies, which confine the upward flow of coolant in the core area to the fuel bearing region.

3.2.3.1 <u>Reactor Internals</u>

3.2.3.1.1 Design Description

The reactor internals are designed to support and orient the reactor core fuel assemblies and control rod assemblies, absorb the control rod dynamic loads, and transmit these and other loads to the reactor vessel flange, provide a passageway for the reactor coolant, and support incore instrumentation. The reactor internals are shown in Figure 3.2-47.

The internals have been designed to withstand the forces due to weight, preload of fuel assemblies, control rod dynamic loading, vibration, and earthquake acceleration. The internals were analyzed in a manner similar to Connecticut Yankee, San Onofre, Zorita, Saxton, and Yankee. Under the loading conditions, including conservative effects of design earthquake loading, the structure satisfies stress values prescribed in Section III, ASME Nuclear Vessel Code.

The reactor internals are equipped with bottom-mounted incore instrumentation supports. These supports are designed to sustain the applicable loads outlined above.

The components of the reactor internals are divided into three parts consisting of the lower core support structure (including the entire core barrel and thermal shield), the upper core support structure, and the incore instrumentation support structure.

3.2.3.1.1.1 Lower Core Support Structure

The major component and support member of the reactor internals is the lower core support structure, shown in Figure 3.2-51. This support structure assembly consists of the core barrel, the core baffle, the lower core plate and support columns, the thermal shield, the intermediate diffuser plate, and the bottom support plate, which is welded to the core barrel. All the major

material for this structure is type 304 stainless steel. The core support structure is supported at its upper flange from a ledge in the reactor vessel head flange and its lower end is restrained in its transverse movement by a radial support system attached to the vessel wall. Within the core barrel are axial baffle and former plates, which are attached to the core barrel wall and form the enclosure periphery of the assembled core. The lower core plate is positioned at the bottom level of the core below the baffle plates and provides support and orientation for the fuel assemblies.

The lower core plate is a 2-in. thick member through which the necessary flow distributor holes for each fuel assembly are machined. Fuel assembly locating pins (two for each assembly) are also inserted into this plate. Columns are placed between this plate and the lower core support of the core barrel in order to provide stiffness and to transmit the core load to the lower core support. Intermediate between the support plate and lower core support plate a perforated plate is positioned to diffuse uniformly the coolant flowing into the core.

The one-piece thermal shield is fixed to the core barrel at the top with rigid bolted connections. The bottom of the thermal shield is connected to the core barrel by means of axial flexures. This bottom support allows for differential axial growth of the shield with respect to the core barrel but restricts radial or horizontal movement of the bottom of the shield. Rectangular tubing, in which vessel material samples can be inserted and irradiated during reactor operation, are welded to the thermal shield and extend to the top of the thermal shield. These samples are held in the rectangular tubing by a preloaded spring device at the top and bottom.

The lower core support structure and principally the core barrel serve to provide passageways and control for the coolant flow. Inlet coolant flow from the vessel inlet nozzles proceeds down the annulus between the core barrel and the vessel wall, flows on both sides of the thermal shield, and then into a plenum at the bottom of the vessel. It then turns and flows up through the lower support, passes through the intermediate diffuser plate and then through the lower core plate. The flow holes in the diffuser plate and the lower core are arranged to give a very uniform entrance flow distribution to the core. After passing through the core, the coolant enters the area of the upper support structure and then flows generally radially to the core barrel outlet nozzles and directly through the vessel outlet nozzles.

A small amount of water also flows between the baffle plates and core barrel to provide additional cooling of the barrel. Similarly, a small amount of the entering flow is directed into the vessel head plenum to provide cooling of the head. Both these flows eventually are directed into the upper support structure plenum and exit through the vessel outlet nozzles.

Vertically downward loads from weight, fuel assembly preload, control rod dynamic loading, and earthquake acceleration are carried by the lower core plate partially into the lower core plate support flange on the core barrel shell and partially through the lower support columns to the lower core support and then through the core barrel shell to the core barrel flange supported by the vessel head flange. Transverse loads from earthquake acceleration, coolant cross flow, and vibration are carried by the core barrel shell to be distributed to the lower radial support to the vessel wall, and to the core barrel flange. Transverse acceleration of the fuel assemblies is transmitted to the core barrel shell by direct connection of the lower core plate to the barrel wall and by a radial support type connection of the upper core plate to slab-sided pins pressed into the core barrel.

The main radial support system of the core barrel is accomplished by "key" and "keyway" joints to the reactor vessel wall. At equally spaced points around the circumference, an Inconel block

is welded to the vessel inside diameter. Another Inconel block is bolted to each of these blocks, and has a "keyway" geometry. Opposite each of these is a "key", which is attached to the internals. At assembly, as the internals are lowered into the vessel, the keys engage the keyways in the axial direction. With this design, the internals are provided with a support at the furthest extremity, and may be viewed as a beam fixed at the top and simply supported at the bottom.

Radial and axial expansions of the core barrel are accommodated but transverse movement of the core barrel is restricted by this design. With this system, cycle stresses in the internal structures are within the ASME Section III limits. This eliminates any possibility of failure of the core support.

In the event of downward vertical displacement of the internals, energy absorbing devices limit the displacement by contacting the vessel bottom head. The load is transferred through the energy devices of the internals.

The energy absorbers, cylindrical in shape, are contoured on their bottom surface to the reactor vessel bottom head geometry. Their number and design are determined so as to limit the forces imposed to less than yield. Assuming a downward vertical displacement, the potential energy of the system is absorbed mostly by the strain energy of the energy absorbing devices.

The free fall in the hot condition is on the order of 0.50-in. and there is an additional strain displacement in the energy absorbing devices of approximately 0.75-in. Alignment features in the internals prevent cocking of the internals structure during this postulated drop. The control rods are designed to provide assurance of control rod insertion capabilities under this assumed drop of internals condition. The drop distance of about 1.25-in. is not enough to cause the tips of the shutdown group of rod cluster control assemblies to come out of the guide tubes in the fuel assemblies.

3.2.3.1.1.2 Upper Core Support Assembly

The upper core support assembly, shown in Figure 3.2-52, consists of the top support plate, deep beam sections, and upper core plate between which are contained 48 support columns and 61 guide tube assemblies. The support columns establish the spacing between the top support plate, deep beam sections, and the upper core plate. They are fastened at top and bottom to these plates and beams. The support columns transmit the mechanical loadings between the two plates and serve the supplementary function of supporting thermocouple guide tubes. The guide tube assemblies, shown on Figure 3.2-53, sheath and guide the control rod drive shafts and control rods and provide no other mechanical functions. They are fastened to the top support plate and are guided by pins in the upper core plate for proper orientation and support. Additional guidance for the control rod drive shafts is provided by the control rod shroud tube, which is attached to the upper support plate and guide tube.

The upper core support assembly, which is removed as a unit during refueling operation, is positioned in its proper orientation with respect to the lower support structure by flat-sided pins pressed into the core barrel, which in turn engage in slots in the upper core plate. At an elevation in the core barrel where the upper core plate is positioned, the flat-sided pins are located at angular positions of 0, 90, 180, and 270 degrees. Four slots are milled into the core plate at the same positions. As the upper support structure is lowered into the main internals, the slots in the plate engage the flat-sided pins in the axial direction. Lateral displacement of the plate and of the upper support assembly is restricted by this design. Fuel assembly locating pins

pins protrude from the bottom of the upper core plate and engage the fuel assemblies as the upper assembly is lowered into place. Proper alignment of the lower core support structure, the upper core support assembly, the fuel assemblies and control rods is thereby assured by this system of locating pins and guidance arrangement. The upper core support assembly is restrained from any axial movements by a large circumferential spring, which rests between the upper barrel flange and the upper core support assembly and is compressed by the reactor vessel head flange.

Vertical loads from weight, earthquake acceleration, hydraulic loads, and fuel assembly preload are transmitted through the upper core plate via the support columns to the deep beams and top support plate and then to the reactor vessel head. Transverse loads from coolant cross flow, earthquake acceleration, and possible vibrations are distributed by the support columns to the top support plate and upper core plate. The top support plate is particularly stiff to minimize deflection.

3.2.3.1.1.3 Incore Instrumentation Support Structures

The incore instrumentation support structures consist of an upper system to convey and support thermocouples penetrating the vessel through the head and a lower system to convey and support flux thimbles penetrating the vessel through the bottom.

The upper system utilizes the reactor vessel head penetrations. Instrumentation port columns are slip-connected to inline columns that are in turn fastened to the upper support plate. These port columns protrude through the head penetrations. The thermocouples are carried through these port columns and the upper support plate at positions above their readout locations. The thermocouple conduits are supported from the columns of the upper core support system. The thermocouple conduits are sealed stainless steel tubes.

In addition to the upper incore instrumentation, there are reactor vessel bottom port columns, which carry the retractable, cold-worked stainless steel flux thimbles that are pushed upward into the reactor core. Conduits extend from the bottom of the reactor vessel down through the concrete shield area and up to a thimble seal line. The minimum bend radii are about 144-in. and the trailing ends of the thimbles (at the seal line) are extracted approximately 15-ft during refueling of the reactor in order to avoid interference within the core. The thimbles are closed at the leading ends and serve as the pressure barrier between the reactor pressurized water and the containment atmosphere.

Mechanical seals between the retractable thimbles and the conduits are provided at the seal line. During normal operation, the retractable thimbles are stationary and move only during refueling or for maintenance, at which time a space of approximately 15-ft above the seal line is cleared for the retraction operation. Sections 7.4 and 7.6 contain more information on the layout of the incore instrumentation system. The incore instrumentation support structure is designed for adequate support of instrumentation during reactor operation and is rugged enough to resist damage or distortion under the conditions imposed by handling during the refueling sequence.

3.2.3.1.2 Evaluation of Core Barrel and Thermal Shield

The internals design is based on analysis, test, and operational information. Troubles in previous Westinghouse PWRs have been evaluated and information derived has been considered in this design. For example, the Westinghouse design uses a one-piece thermal

shield, which is attached rigidly to the core barrel at one end and flexured at the other. The early designs that malfunctioned were multi-piece thermal shields that rested on vessel lugs and were not rigidly attached at the top.

Early core barrel designs that have malfunctioned in service, now abandoned, employed threaded connections such as tie rods, joining the bottom support to the bottom of the core barrel, and a bolted connection that tied the core barrel to the upper barrel. The malfunctioning of core barrel designs in earlier service was believed to have been caused by the thermal shield, which was oscillating, thus creating forces on the core barrel. Other forces were induced by unbalanced flow in the lower plenum of the reactor. In today's rod cluster control design there are no fuel followers to necessitate a large bottom plenum in the reactor. The elimination of these fuel followers enabled Westinghouse to build a shorter core barrel.

The Connecticut Yankee reactor and the Zorita reactor core barrels are of the same construction as the Indian Point Unit 2 reactor core barrel. Deflection measuring devices employed in the Connecticut Yankee reactor during the hot functional test, and deflection and strain gauges employed in the Zorita reactor during the hot-functional test provided important information that was used in the design of the present day internals, including that for Indian Point. When the Connecticut Yankee thermal shield was modified to the same design as for Southern California Edison, it, too, operated satisfactorily as was evidenced by the examination after the hot-functional test. After hot-functional tests on all of these reactors, a careful inspection of the internals was examined for any differential movement; upper core plate inside supports were examined, and the thermal shield attachments to the core barrel, including all lockwelds on the devices used to lock the bolt, were checked; no malfunctions were found.

Substantial scale model testing was performed at WAPD. This included tests, which involved a complete full-scale fuel assembly, which was operated at reactor flow, temperature, and pressure conditions. Tests were run on a one-seventh scale model of the Indian Point Unit 2 reactor. Measurements taken from those tests indicated very little shield movement, on the order of a few mils when scaled up to Indian Point Unit 2. Strain gauge measurements taken on the core barrel also indicated very low stresses. Testing to determine thermal shield excitation due to inlet flow disturbances was included. Information gathered from these tests was used in the design of the thermal shield and core barrel. It was concluded, from the experience gained during the testing program and the analyses, that the design as employed on the Indian Point Unit 2 plant is adequate.

In order to confirm the internals design, deflection gauges were mounted on the thermal shield top and bottom for the hot-functional test. Gauges were mounted in the top of the thermal shield equidistant from the fixed supports, and at the bottom of the thermal shield, equidistant from the six flexures, and next to the flexure supports. The internals inspection, just before the hot-functional test, included looking at mating bearing surfaces, main welds, and welds that are used on bolt locking devices. At the conclusion of the hot-functional test, measurement readings were taken from the deflectometers on the shield and the internals were re-examined at all key areas for any evidence of malfunction.

3.2.3.2 Core Components

3.2.3.2.1 Design Description

3.2.3.2.1.1 Fuel Assembly

The 15x15 Upgraded fuel assembly, introduced in Cycle 17, is shown in Figure 3.2-61C. The assemblies are square in cross section, nominally 8.426-in. on a side, and have an overall height of approximately 159.975 inches. The fuel rods in a fuel assembly are arranged in a square array with 15 rod locations per side and a nominal centerline-to-centerline pitch of 0.563-in. between rods. Of the total possible 225 rod locations per assembly, 20 are occupied by guide thimbles for the rod cluster control rods and one for incore instrumentation. The remaining 204 locations contain fuel rods. In addition to fuel rods, a fuel assembly is composed of a top nozzle, a bottom nozzle, ten grid assemblies (plus 3 intermediate flow mixing grids starting with Cycle 13), 20 absorber rod guide thimbles, and one instrumentation thimble.

The guide thimbles in conjunction with the grid assemblies and the top and bottom nozzles comprise the basic structural fuel assembly skeleton. The grid assemblies are bulge attached to the guide thimbles at each location along the height of the fuel assembly at which lateral support for the fuel rods is required. Within this skeletal framework the fuel rods are contained and supported and the rod-to-rod centerline spacing is maintained along the assembly.

The original fuel design for Indian Point 2 was the Westinghouse High Parasitic (HIPAR) fuel assembly. This consisted of Zircoloy clad fuel rods, 9 Inconel grids and stainless steel instrumentation and guide thimbles. Burnable absorbers used were pyrex glass.

Starting with Cycle 5, the Westinghouse Low Parasitic (LOPAR) fuel assembly was introduced. This design consisted of Zircaloy-4 clad fuel rods, 9 Inconnel grids and Zircaloy-4 instrumentation and guide thimbles.

For the Cycle 8, Wet Annular Burnable Absorbers (WABA) were introduced.

For Cycle 10, the Westinghouse Optimized Fuel Assembly (OFA) was introduced. This consisted of Zircaloy-4 clad fuel rods, 2 Inconel grids (top & bottom), 7 Zircaloy-4 grids and Zircaloy-4 instrumentation and guide thimbles. In addition, thimble plugs were removed from the core this cycle based on analysis performed to support removal. The assembly top nozzle design was changed to a Reconstitutable Top Nozzle (RTN) design to facilitate reconstitution of failed fuel.

For Cycle 11, the OFA fuel assembly design incorporated Debris Filter Bottom Nozzles (DFBN) and Integral Fuel Burnable Adsorbers (IFBA).

For Cycle 13, the Westinghouse Vantage+ fuel design was introduced (see Figures 3.2-54 and 3.2-56B). This design included ZIRLO clad fuel rods, 2 Inconel grids, 7 low pressure drop (LPD) Zircaloy-4 grids, 3 Zircaloy Integral Flow Mixing grids (IFM), ZIRLO instrumentation and guide thimbles, annular axial blankets along with the DFBN, IFBA and RTN. Use of WABAs was continued. See Reference 13.

For Cycle 15, the Vantage+ fuel assembly design incorporated Performance+ features of ZIRLO grids and IFMs and a hardened coating of zirconium oxide on the bottom section of the fuel rod clad to increase debris resistance.

For Cycle 16, the Vantage+ fuel assembly design was further enhanced with Performance+ features that include debris mitigation features of an additional grid located at the bottom end plug of the fuel rod, a longer fuel rod end plug and a revised DFBN. Other performance+ enhancements include longer fuel rods and longer annual axial blanket (see Figure 3.2-61B).

In addition to the above fuel design changes, the design burnup of the fuel assemblies has also been increasing to 62,000 MWD/MTU lead rod burnup for Cycle 16. See References 15, 16 and 17.

For Cycle 17, the 15x15 Upgraded fuel assembly design was used. This design has features to address grid-to-rod fretting fuel failures. These include I-spring mid-grids, enhanced IFMs and balanced mixing vanes. In addition, the tube-in-tube thimble design was incorporated with a single-dashpot, which improves straightness.

For Cycle 18, solid axial blanket pellets were introduced for the non-IFBA fuel rods.

For Cycle 19, the top nozzle spring design was changed from the Vantage+ design to the standard spring design.

Cycle 20 fuel is the same as Cycle 19, there are no fuel design changes.

Bottom Nozzle

Two types of nozzle designs were used for the HIPAR fuel assemblies. One type, which is square in cross section, is fabricated from type 304 stainless steel consisting of four side plates, 12 cross bars and four pads or feet. The side plates are welded together at the corners to form a plenum for inlet coolant to the fuel assembly. The cross bars are welded at each end to the top edges of the side plate and function as the bottom end support for the fuel rods. The bottom support surface for the fuel assembly is formed by the four pads, which are welded to the side plates in the corners. This design was used in a majority of the first core fuel assemblies. The previously used LOPAR and OFA fuel incorporate an equivalent bottom nozzle design utilizing a square perforated plate rather than the cross bars and side plate. On both designs, their respective cross bars and perforated plate prevent the fuel rods from falling through the bottom nozzles of the assembly.

Coolant flow to the fuel assembly is directed from the plenum in the bottom nozzle upward to the interior of the fuel assembly and to the channel between assemblies.

Axial loads imposed on the assembly, as well as the weight of the assembly are distributed through the guide thimble and the bottom nozzle to the lower core support plate. Indexing and positioning of the fuel assembly in the core is controlled through two holes in diagonally opposite pads, which mate with locating pins in the lower core plate. Lateral loads imposed on the fuel assembly are also transferred to the core support structures through the locating pins.

The OFA and VANTAGE+ bottom nozzle used the reconstitutable feature found on the previously installed LOPAR fuel design, which uses a locking cup to lock the thimble screws on the guide thimble assembly, instead of the lockwire used in earlier LOPAR designs. The OFA nozzle assembly is shorter when compared to the previously installed LOPAR assembly to enhance fuel rod growth allowances.

The two bottom nozzle designs used in the OFAs are both square in cross section and fabricated from 304 stainless steel. The design used in earlier regions consists of a perforated plate, four angle legs, and four pads of feet. The angle legs are fastened to the plate forming a plenum space for the coolant inlet to the fuel assembly.

The remaining OFA regions and the VANTAGE+ and 15x15 Upgraded fuel regions (starting with Cycle 13, Region 15) incorporate an equivalent bottom nozzle design denoted as the Debris Filter Bottom Nozzle (DFBN). This nozzle adds side plates or "skirts" to the previous design increasing structural capability for abnormal loads and providing a more defined plenum space below the nozzle. Additionally, the relatively large adapter plate flow holes of the earlier design are replaced with a new pattern of smaller flow holes. The decrease in size of the holes provides a "screen" for larger debris particles, which would otherwise cause damage if allowed to pass into the assembly.

In both designs, the adaptor plates prevent accidental downward ejection of the fuel rods from the fuel assembly. The nozzles are fastened to the assembly guide tubes by stainless steel screws, which penetrate through the nozzle and mate with a threaded plug in each guide tube (Figure 3.2-57). The screw possesses a circular locking cup around the screw head, which is crimped into mating detentes (lobes) on the bottom nozzle, preventing the screw from loosening.

Top Nozzle

The Reconstitutable Top Nozzle (RTN) used in both the OFA, VANTAGE+ and 15x15 Upgraded fuel assemblies is a box-like structure, which functions as the fuel assembly upper structural element and forms a plenum space where the heated fuel assembly discharge coolant is mixed and directed toward the flow holes in the upper core plate. The nozzle is comprised of an adaptor plate enclosure, top plate, clamps, hold-down leaf springs and assorted hardware. Each nozzle has four sets of leaf springs. All parts, with the exception of the springs and their hold-down bolts/screws, are constructed of type 304 stainless steel. The springs are made from age hardenable Inconel 718 and the bolts/screws from Inconel 600 for Region 16 and earlier regions, and from shotpeened Inconel 718 for Regions 17 and 18.

The adaptor plate portion of the nozzle is square in cross section, and is perforated by machined slots to provide for coolant flow through the plate. At assembly, the top ends of the LOPAR thimble stainless sleeves are fitted through individual bored holes in the plate and welded to the plate around the circumference of each hole. In the OFA removable top nozzle design, a groove is provided in each thimble thru-hole in the nozzle plate into which a stainless steel nozzle insert is mechanically connected by means of a preformed circumferential bulge near the top of the insert. Thus, the adaptor plate acts as the fuel assembly top end plate, and provides a means of distributing evenly among the guide thimbles any axial loads imposed on the fuel assemblies.

The nozzle enclosure is actually a square tubular structure, which forms the plenum section of the top nozzle. The bottom end of the enclosure is pinned and welded to the periphery of the adaptor plate and the top end is welded to the periphery of the top plate.

The top plate is square in cross section with a square central hole. The hole allows clearance for the rod cluster control absorber rods to pass through the nozzle into the guide thimbles in the fuel assembly and for coolant exit from the fuel assembly to the upper internals area. Two pads containing axial through-holes, which are located on diametrically opposite corners of the top

top plate provide a means of positioning and aligning the top of the fuel assembly. As with the bottom nozzle, alignment pins in the upper core plate mate with the holes in the top nozzle plate. Hold-down forces of sufficient magnitude to oppose the hydraulic lifting forces on the fuel assembly are obtained by means of the leaf spring sets, which are mounted on the top plate. The springs are fastened in pairs to the top plate at the two corners where alignment holes are not used and radiate out from the corners parallel to the sides of the plate. Fastening of each pair of springs is accomplished with a clamp, which fits over the ends of the springs and two bolts/screws (one per spring set), which pass through the clamp and spring, and thread into the top plate. At assembly, the spring mounting bolts/screws are torqued sufficiently to preload against the maximum spring load and then lockwelded to the clamp, which is counterbored to receive the bolt/screw head. The spring load is obtained through deflection of the spring pack by the upper core plate. The spring pack form is such that it projects above the fuel assembly and is depressed by the core plate when the internals are loaded into the reactor. The free end of the spring pack is bent downward and captured in a key slot in the top plate to guard against loose parts in the reactor in the event (however remote) of spring fracture. The capture of the loose end has been deleted in latter designs.

Starting with Cycle 14, Region 16, the fuel has a cast top nozzle. This is a two-piece design incorporating a machined stainless steel adapter plate welded to a low-cobalt investment casting. The cast top nozzle is functionally interchangeable with the previous design and meets design criteria for the top nozzle.

In addition to its plenum and structural functions, the nozzle provides a protective housing for components, which mate with the fuel assembly. In handling a fuel assembly with a control rod inserted, the control rod spider is contained within the nozzle. During operation in the reactor, the nozzle protects the absorber rods from coolant cross flows in the unsupported span between the fuel assembly adaptor plate and the end of the guide tube in the upper internals package. Plugging devices, [*Note - As a result of analyses performed for OFA transition, maintaining plugging devices in the core is optional.*], which fill the ends of the fuel assembly thimble tubes at unrodded core locations and the source rods and burnable absorber rods, are all contained within the fuel top nozzle.

For the RTN design, a stainless steel nozzle insert is mechanically connected to the top nozzle adaptor plate (Figure 3.2-58A) via the engagement of the preformed circumferential bulge near the top of the insert and the mating groove in the wall of the adapter plate thimble tube through-hole. The insert has four equally spaced axial slots, which allow the insert to deflect inwardly at the elevation of the bulge, thus permitting the installation and removal of the nozzle. The insert bulge is positively held in the adapter plate mating groove by placing a lock tube with a uniform OD identical to that of the thimble tube into the insert. The lock tube is secured in place by a top flare, which creates a tight fit and six non-yielding projections on the OD, which interface with the concave side of the insert to preclude escape during core component transfer. The adaptor plate acts as the fuel assembly top end plate and provides a means of evenly distributing any axial loads imposed on the fuel assemblies to the guide thimbles.

Guide Thimbles

The control rod guide thimbles in the fuel assemblies provide guided channels for the absorber rods during insertion and withdrawal of the control rods. Up to and including Region 18 (VANTAGE+), they are fabricated from a single piece of tubing, which is drawn to two different diameters. The OFA thimbles are Zircaloy-4 and the VANTAGE+ thimble material ZIRLO™. The larger inside diameter at the top provides a relatively large annular area for rapid insertion

during a reactor trip and accommodates a small amount of upward cooling flow during normal operations. The bottom portion of the guide thimble has two sections of reduced diameter producing a "double dashpot" action when the absorber rods near the end of travel in the guide thimbles during a reactor trip. The transition zones at the dashpot sections are conical in shape so that there are no rapid changes in diameter in the tube.

Starting with Region 19 (15x15 Upgraded fuel design), the guide thimbles incorporate the tube-in-tube dashpot design. The tube-in-tube design utilizes a separate dashpot tube assembly that is inserted into the guide thimble assembly pulled to a press fit over the thimble end plug and bulged into place. To maintain the same diametrical clearance between the guide thimble ID and the dashpot OD, the 15x15 upgraded nominal dashpot thickness was reduced from 0.0165 to 0.0160 inches. As the dashpot tube in the design can provide additional lateral support in that bottom thimble span, it is expected that there will be additional resistance to lateral deformation and Incomplete Rod Insertions as a result of the design modification. The 15x15 Upgraded fuel thimbles are ZIRLO™.

Flow holes are provided just above the first dashpot transition to permit the entrance of cooling water during normal operation, and to accommodate the outflow of water from the dashpot during reactor trip.

The dashpot is open at the bottom by means of the drainage hole in the thimble screws that secure the bottom nozzle to the welded end plugs of the guide thimbles. This geometry is shown in Figure 3.2-57.

The top ends of the thimble tubes are mechanically attached to the sleeve of the top grid. An insert is also bulge attached to the thimble and the insert upper end is in turn mechanically attached to the top nozzle as shown in Figure 3.2-58A.

VANTAGE+ Grids

Prior to Region 18, the VANTAGE+ assembly has twelve grids. Starting with Region 18, a thirteenth grid, the protective grid (P-grid), was added to the VANTAGE+ assembly. The top and bottom grids, as in the OFA assembly, and the P-grid are Inconel 718 non-mixing vane grids. The top and bottom grids are Inconel-718 non-mixing vane grids. Low Pressure Drop (LPD) Zircaloy grids are used for the middle grids with Zircaloy IFMs located in the three uppermost middle grid spans. The VANTAGE+ fuel assembly with PERFORMANCE+ options has ZIRLO™ grids for the 3IFMs and 7 mid grids. The LPD grids have mixing vanes, diagonal springs and a reduced grid height, relative to the OFA grids. The LPD grid cells use the standard four dimples and two springs per cell for support locations. The IFMs provide mid-span flow mixing in the hottest fuel assembly spans. Each IFM cell contains four dimples, which are designed to prevent midspan channel closure and fuel rod contact with the mixing vanes. With the additional Performance+ enhancements added to the fuel starting with Region 18, a new Protective Bottom Grid (PBG) has been added. The PBG is a wider, extra grid at the very bottom of the fuel assembly that protects the fuel from debris. Its purpose is to filter out debris and hold it at an elevation below the bottom of the active core. The PBG is not a structural grid. The bottom of the PBG lies below the tops of the lower end plugs within the fuel rod. This means that any debris caught in the PBG cannot fret through the cladding and expose fuel pellets.

All VANTAGE+ outside grid straps contain mixing vanes, which also act as guides during fuel handling. The grids are also attached to the thimble tubes via the bulging mechanism as shown

in Figure 3.2-61A. Top grid nozzle attachment is shown in Figure 3.2-58A. All grids employ the anti-snag outer strap design. A mixing vane grid is shown in Figure 3.2-59.

15x15 Upgraded Design Grids

The 15x15 Upgraded fuel design still contains twelve grids with the top and bottom grids unchanged from the Vantage+ design. The thirteenth grid, the protective gird (P-grid) also remains the same as the Vantage+ design. The middle grids have changed to an I-spring design. The changes were made to improve fuel rod fretting margin. In addition to the spring change the size of the dimples was increased. The strap thickness was decreased to help offset pressure drop increase due to the I-spring and increased grid strap height. The strap height increased to create space to accommodate the increased dimples and the I-spring. The IFM grid design was enhanced to increase contact area also.

Fuel Rods

The fuel rods consist of uranium-dioxide ceramic pellets in slightly cold worked ZIRLO™ tubing, which is plugged and seal welded at the ends to encapsulate the fuel. Sufficient void volume and clearances are provided within the rod to accommodate fission gases released from the fuel, differential thermal expansion between the cladding and the fuel, helium released from poison burnup (IFBA rods), and fuel swelling due to accumulated fission products without overstressing of the cladding or seal welds. Shifting of the fuel within the cladding is prevented during handling or shipping prior to core loading by a stainless steel helical compression spring, which bears on the top of the fuel.

At assembly, the pellets are stacked in the cladding to the required fuel height. The compression spring is then inserted into the top end of the fuel and the end plugs pressed into the ends of the tube and welded. A hold-down force of approximately four times the weight of the fuel is obtained by compression of the spring between the top end plug and the top of the fuel pellet stack. All fuel rods are internally pressurized with helium in order to minimize compressive clad stresses and creep due to coolant operating pressures.

The fuel pellets are in the form of a right circular cylinder and consist of slightly enriched uranium-dioxide powder, which is compacted by cold pressing and sintering to the required density. The ends of each pellet are dished slightly to allow the greater axial expansion at the center of the pellets to be taken up within the pellets themselves and not in the overall fuel length. The 15x15 Upgrade fuel has mid-enriched annular (IFBA) and solid (non-IFBA) pellets in the axial blanket region of the fuel rod and optimized plenum spring to maximize the available plenum volume for increased burnup. The 15x15 Upgrade fuel has a longer fuel rod to allow higher fission gas release due to longer cycles. This is allowable due to the ZIRLO cladding, which has less rod growth on irradiation.

For the first core, the pellets in the outer region had a density of approximately 10.3 g/cm^3 (94-percent of theoretical density) while those in the two inner regions (checkerboard pattern, see Figure 3.2-48) had a density of 10.4 g/cm^3 corresponding to 95-percent of theoretical density. Lower pellet densities were used to compensate for the effects of the higher burnup, which the fuel experienced in those regions.

Reload cores contain 15x15 Upgraded fuel arranged in a zoned and/or checkerboard pattern. Different fuel enrichments, as listed in Table 3.2-7, are used for each of the core regions for all core loadings.

Each fuel rod is marked with a permanent traceability code. This aids in ensuring that rods of the proper enrichment will be loaded into each fuel assembly. The identification numbers on the fuel assembly top nozzles will then maintain the enrichment identity and ensure that the assemblies with the correct enrichment are loaded into the proper core region.

Each assembly is assigned a core loading position. A record is then made of the core loading position, serial number, and enrichment. During the core loading, two independent checks are made to ensure that the actual loading position agrees with the position assigned.

During initial core loading and subsequent refueling operations, detailed handling and checkoff procedures are used throughout the sequence. The initial core was loaded in accordance with the core loading diagram similar to Figure 3.2-48, which shows the location for each of the three enrichment types of fuel assemblies used in the loading.

3.2.3.2.1.2 Rod Cluster Control Assemblies

The control rods or rod cluster control assemblies consist of a group of individual absorber rods fastened at the top end to a common hub or spider assembly. These assemblies, one of which is shown in Figure 3.2-49, are provided to control the reactivity of the core under operating conditions. The absorber material used in the control rods is silver-indium-cadmium alloy, which is essentially "black" to thermal neutrons and has sufficient additional resonance absorption to increase its worth significantly. The alloy is in the form of extruded single-length rods, which are sealed in stainless steel tubes to prevent the rods from coming in direct contact with the coolant.

The overall control rod length is such that when the assembly has been withdrawn through its full travel, the tips of the absorber rods remain engaged in the guide thimbles so that alignment between rods and thimbles is always maintained. Since the rods are long and slender, they are relatively free to conform to any small misalignments with the guide thimble. Prototype tests have shown that the rod cluster control assemblies are very easily inserted and not subject to binding even under conditions of severe misalignment.

The spider assembly is in the form of a center hub with radial vanes supporting cylindrical fingers from which the absorber rods are suspended. Handling detents and detents for connection to the drive shaft are machined into the upper end of the hub. A spring pack is assembled into a skirt integral to the bottom of the hub to stop the rod cluster control assembly and absorb the energy from the impact at the end of a trip insertion. The radial vanes are joined to the hub and the fingers are joined to the vanes by furnace brazing. A centerpost, which holds the spring pack and its retainer is threaded into the hub within the skirt and welded to prevent loosening in service. All components of the spider assembly are made from type 304 stainless steel except for the springs, which are Inconel X-750 alloy and the retainer, which is of 17-4 pH material.

The absorber rods are secured to the spider so as to ensure trouble free service. The rods are first threaded into the spider fingers and then pinned to maintain joint tightness, after which the pins are welded in place. The end plug below the pin position is designed with a reduced section to permit flexing of the rods to correct for small operating or assembly misalignments.

In construction, the silver-indium-cadmium rods are inserted into cold-worked stainless steel tubing, which is then sealed at the bottom and the top by welded end plugs. Sufficient diametral

and end clearance are provided to accommodate relative thermal expansions and to limit the internal pressure to acceptable levels.

The bottom plugs are made bullet-nosed to reduce the hydraulic drag during a reactor trip and to guide smoothly into the dashpot section of the fuel assembly guide thimbles. The upper plug is threaded for assembly to the spider and has a reduced end section to make the joint more flexible. Stainless steel clad silver-indium-cadmium alloy absorber rods are resistant to radiation and thermal damage thereby ensuring their effectiveness under all operating conditions. Rods of similar design have been successfully used in a number of operating nuclear plants.

3.2.3.2.1.3 Neutron Source Assemblies

Six neutron source assemblies were utilized in the first cycle core. These consisted of two assemblies with four secondary source rods each, and four assemblies with one secondary source rod and one primary source rod each. The rods in each assembly were fastened to a spider at the top end. The spider for the four secondary source rod assemblies was similar to the rod cluster control assembly spiders, while the primary source assembly spider was similar to that of the burnable poison and plugging device assemblies. Various source assembly designs are used in the reload cycles.

In the first cycle core, the neutron source assemblies were inserted into the rod cluster control guide thimbles in fuel assemblies at unrodded locations. The location and orientation of each of the assemblies in the core is shown in Figure 3.2-62.

The primary and secondary source rods both utilize the same type of cladding material as the absorber rods (cold-worked type 304 stainless steel tubing). The secondary source rods contain Sb-Be pellets. The primary source rods contained capsules of Pu-Be source material in the initial core loading; for reload cores, this material may vary. Design criteria similar to that for the fuel rods is used for the design of the source rods; i.e., the cladding is free standing, internal pressures are always less than reactor operating pressure, and internal gaps and clearances are provided to allow for differential expansions between the source material and cladding.

3.2.3.2.1.4 Plugging Devices

In order to limit bypass flow through the rod cluster control guide thimbles in fuel assemblies, which do not contain either control rods, source assemblies or burnable absorber rods, the fuel assemblies at those locations were fitted with plugging devices. The plugging devices consist of a flat plate with short rods suspended from the bottom surface and a spring pack assembly. At installation in the core, the plugging devices fit with the fuel assembly top nozzles and rest on the adaptor plate. The short rods project into the upper ends of the thimble tubes to reduce the bypass flow area. The spring pack is compressed by the upper core plate when the upper internals package is lowered into place. Similar short rods are also used on the source assemblies to fill the ends of all vacant fuel assembly guide thimbles. All components in the plugging device, except for the springs, are constructed from type 304 stainless steel. The springs are wound from an age hardenable nickel base alloy to obtain higher strength.

Coincident with implementation of the Indian Point Unit 2 OFA transition, removal of thimble plugging devices from the core was allowed. This included the removal of the thimble plugs from the OFA assemblies, previously installed LOPAR assemblies, and all new core component clusters (burnable absorbers and sources).

As part of the implementation of the power uprate, Cycle 17 core will contain thimble plugs for all assemblies that do not contain inserts e.g. RCCAs, WABAs or secondary sources.

3.2.3.2.1.5 Burnable Absorber Rods

The burnable absorber rods are statically suspended and positioned in vacant rod cluster control thimble tubes within the fuel assemblies at nonrodded core locations. The absorber rods in each fuel assembly are grouped and attached together at the top end of the rods by a flat plate, which fits with the fuel assembly top nozzle and rests on the top adaptor plate.

The plate with the absorber rods is held down and restrained against vertical motion with a spring pack, which is attached to the plate and is compressed by the upper core plate when the reactor upper internals package is lowered into the reactor. This ensures that the absorber rods cannot be lifted out of the core by flow forces.

The absorber rods used during Cycles 1 through 7 consisted of borated Pyrex glass tubes contained within type 304 stainless steel tubular cladding, which was plugged and seal welded at the ends to encapsulate the glass. The glass was also supported along the length of its inside diameter by a thin-wall type 304 stainless steel tubular inner liner.

Starting in Cycle 8, Wet Annular Burnable Absorber (WABA) rods were used and are described in Reference 74. As shown in Figures 3.2-69 and 3.2-70, WABA rods are composed of annular pellets containing aluminum oxide-boron carbide (Al_2O_3 - B_4C) burnable absorber material contained within two concentric Zircaloy tubes. The Zircaloy tubes are plugged and seal welded at the ends to enclose the annular stack of absorber material. The tubes are also the inner and outer cladding of the annular burnable absorber rod. A hold-down device is placed on top of the pellet stack to hold the stack in position and to allow for pellet stack growth. The hold-down device is a C-shape Zircaloy polygonal ring clip. Within the rod is an annular plenum to allow for helium gas release from the absorber material during boron depletion. Reactor coolant flows through the inner tube and outside the outer tube of the annular rod. The annular rods are grouped and attached at the top end to a hold-down assembly and retaining plate in the same way as the borosilicate glass absorber rod. WABA rods are used in preference to standard BPRAs to provide smaller residual burnup penalty.

Starting with Cycle 11, Integral Fuel Burnable Absorbers (IFBA) were used in conjunction with the WABA rods. The IFBA features a zirconium diboride coating on the fuel pellet surface on the central portion of the enriched UO_2 pellets. IFBA's provide power peaking and moderator temperature coefficient control. IFBA's are described in Reference 88.

3.2.3.2.2 Evaluation of Core Components

3.2.3.2.2.1 Fuel Evaluation

The fission gas release and the associated buildup of internal gas pressure in the fuel rods are calculated by overall fuel rod design models, which incorporate time-dependent fuel densification (References 68 and 69). The increase of internal pressure in the fuel rod due to this phenomena is included in the determination of the maximum cladding stresses at the end of core life when the fission product gap inventory is a maximum. Modifications to the initial core fuel design and evaluations are given in the Indian Point Unit 2 Fuel Densification Reports[70 and 71]. The VANTAGE+ fuel rod design bases and evaluation are given in Reference 88. The fuel rod design has not been changed as part of the 15x15 Upgraded fuel design.

The maximum allowable tensile strain in the cladding during steady-state operation, considering the combined effects of internal gas pressure, external coolant pressure, fuel pellet swelling and thermal expansion, and cladding creep is less than 1% from the unirradiated condition throughout core life. For Condition II transients, the total tensile strain during the transient is less than 1% from the pre-transient value. The associated stresses are below the yield strength of the material under steady-state and Condition II transient conditions.

To assure that manufactured fuel rods meet a high standard of excellence from the standpoint of functional requirements, many inspections and tests are performed both on the raw material and the finished product. These tests and inspections include chemical analysis, tensile testing of fuel tubes, dimensional inspection, X-ray of both end plug welds, ultrasonic testing, and helium leak tests.

In the event of cladding defects, the high resistance of uranium-dioxide fuel pellets to attack by hot water protects against fuel deterioration or decrease in fuel integrity. Thermal stress in the pellets, while causing some fracture of the bulk material during temperature cycling, does not result in pulverization or gross void formation in the fuel matrix. As shown by operating experience and extensive experimental work in the industry, the thermal design parameters conservatively account for any changes in the thermal performance of the fuel element due to pellet fracture.

The consequences of a breach of cladding are greatly reduced by the ability of uranium-dioxide to retain fission products including those, which are gaseous or highly volatile. This retentiveness decreases with increasing temperature or fuel burnup, but remains a significant factor even at full power operating temperature in the maximum burnup element.

Data on fuel behavior in high burnup uranium-dioxide show that it is possible to conservatively define the fuel swelling as a function of burnup and as-fabricated uranium-dioxide porosity (References 68 and 69).

Actual fuel rod damage limits depend upon neutron exposure and normal variation of material properties and are greater than the design limits. For the life of the fuel rod, the actual stresses and strains are below the design limits. Thus, significant margins exist between actual operating conditions and the damage limits.

The other parameters having an influence on cladding stress and strain are as follows:

1. <u>Internal gas pressure</u>

   The maximum rod internal pressure under nominal conditions will be substantially less than the calculated pressure at the design limits. The end-of-life internal gas pressure is dependent upon the fuel rod power history and will not exceed the design limit defined in Section 3.1.2.1 (item 3).

2. <u>Cladding temperature</u>

   The strength of the fuel cladding is temperature dependent. The minimum ultimate strength reduces to the design yield strength at an average cladding temperature of approximately 850°F. The maximum average cladding temperature during normal operating conditions is given in Table 3.2-6.

3. Burnup

Fuel burnup results in fuel swelling, which, along with fuel thermal expansion, causes tensile cladding strain. Since rod power levels, and hence fuel temperature, decreases with burnup, the fuel pellet diameter increase with burnup is somewhat mitigated by the reduced thermal expansion. The strain design limits and stress design limits are met throughout the burnup lifetime of the fuel. These strain and stress design limits are below the cladding damage limits.

4. Fuel temperature and kW/ft

The fuel is designed so that the maximum fuel temperature will not exceed 4700°F during normal operating conditions or unanticipated malfunction transients (Condition II events).

3.2.3.2.2.2 Evaluation of Burnable Absorber Rods

The burnable absorber rods are positioned in the core inside rod cluster control assembly guide thimbles and held down in place by attachment to a retainer assembly compressed beneath the upper core plate and, hence, cannot be the source of any reactivity transient. Due to the low heat generation rate and the conservative design of the rods, there is no possibility for release of the poison as a result of helium pressure or clad heating during accident transients including loss-of-coolant.

3.2.3.2.2.3 Effects of Vibration and Thermal Cycling on Fuel Assemblies

Analyses of the effect of cyclic deflection of the fuel rods, grid spring, rod cluster control rods, and burnable absorber rods due to hydraulically induced vibrations and thermal cycling show that the design of the components is good for an infinite number of cycles.

In the case of the fuel rod grid spring support, the amplitude of a hydraulically induced motion of the fuel rod is extremely small (approximately 0.001-in.), and the stress associated with the motion is significantly small (<100 psi). Likewise, the reactions at the grid spring due to the motion is much less than the preload spring force and contact is maintained between the fuel clad and the grid spring and dimples. Fatigue of the clad and fretting between the clad and the grid support is not anticipated.

The effect of thermal cycling on the grid-clad support is a slight relative movement between the grid contact surfaces and the clad. Since the number of cycles of the occurrence is small over the life of a fuel assembly (approximately 3 years), negligible wear of the mating parts is expected. Incore operation of assemblies in the Yankee Rowe and Saxton reactors using similar clad support have verified the calculated conclusions. Additional test results under simulated reactor environment in the Westinghouse Reactor Evaluation Channel also support these conclusions.

The dynamic deflection of the control rods and the burnable absorber rods is limited by their fit with the inside diameter of either the upper portion of the guide thimble or the dashpot. With this limitation, the occurrence of truly cyclic motion is questionable. However, an assumed cyclic deflection through the available clearance gap results in an insignificantly low stress in either the

either the clad tubing or in the flexure joint at the spider or retainer plate. The above consideration assumes the rods are supported as cantilevers from the spider or the retainer plate in the case of the burnable absorber rods.

A calculation assuming the rods are supported by the surface of the dashpots and at the upper end by the spider or retainer results in a similar conclusion.

3.2.3.3 Transition Cores

The entire core is now 15x15 Upgraded fuel therefore there are no transition core effects.

3.2.3.4 Control Rod Drive Mechanism Design Description

3.2.3.4.1 Full Length Rods

The control rod drive mechanisms are used for withdrawal and insertion of the rod cluster control assemblies into the reactor core and to provide sufficient holding power for stationary support.

Fast total insertion (reactor trip) is obtained by simply removing the electrical power allowing the rods to fall by gravity. Design scram time is 2.4 seconds from gripper release to dashpot entry.

The complete drive mechanism, shown in Figure 3.2-65, consists of the internal (latch) assembly, the pressure vessel, the operating coil stack, the drive shaft assembly, and the rod position indicator coil stack.

Each assembly is an independent unit, which can be dismantled or assembled separately. Each mechanism pressure housing is threaded onto an adaptor on top of the reactor pressure vessel and seal welded. The operating drive assembly is connected to the control rod (directly below) by means of a grooved drive shaft. The upper section of the drive shaft is suspended from the working components of the drive mechanism. The drive shaft and control rod remain connected during reactor operation including tripping of the rods.

Main coolant fills the pressure containing parts of the drive mechanism. All working components and the shaft are immersed in the main coolant and depend upon it for lubrication of sliding parts.

Three magnetic coils, which form a removable electrical unit and surround the rod drive pressure housing, induce magnetic flux through the housing wall to operate the working components. They move two sets of latches, which lift, lower, and hold the grooved drive shaft.

The three magnets are turned on and off in a fixed sequence by solid-state switches for the full length rod assemblies.

The sequencing of the magnets produces step motion over the full length of normal control rod travel.

The mechanism develops a lifting force approximately two times the static lifting load. Therefore, extra lift capacity is available for overcoming mechanical friction between the moving and the stationary parts. Gravity provides the drive force for rod insertion and the weight of the whole rod assembly is available to overcome any resistance.

The mechanisms are designed to operate in water at 650°F and 2485 psig. The temperature at the mechanism head adaptor will be much less than 650°F because it is located in a region where there is limited flow of water from the reactor core while the pressure is the same as in the reactor pressure vessel. A multiconductor cable connects the mechanism operating coils to the 125-V DC power supply. The power supply is described in Section 7.3.2.

3.2.3.4.1.1 Latch Assembly

The latch assembly contains the working components, which withdraw and insert the drive shaft and attached control rod. It is located within the pressure housing and consists of the pole pieces for three electromagnets. They actuate two sets of latches, which engage the grooved section of the drive shaft.

The upper set of latches move up or down to raise or lower the drive rod by 5/8-in. The lower set of latches have a 1/16-in. axial movement to shift the weight of the control rod from the upper to the lower latches.

3.2.3.4.1.2 Pressure Vessel

The pressure vessel consists of the pressure housing and rod travel housing. The pressure housing is the lower portion of the vessel and contains the latch assembly. The rod travel housing is the upper portion of the vessel. It provides space for the drive shaft during its upward movement as the control rod is withdrawn from the core.

3.2.3.4.1.3 Operating Coil Stack

The operating coil stack is an independent unit, which is installed on the drive mechanism by sliding it over the outside of the pressure housing. It rests on a pressure housing flange without any mechanical attachment and can be removed and installed while the reactor is pressurized.

The three operating coils are made of round copper wire, which is insulated with a double layer of filament type glass yarn.

The design operating temperature of the coils is 200°C. Average coil temperature can be determined by resistance measurement. Forced air cooling along the outside of the coil stack maintains a coil casing temperature of approximately 120°C or lower.

3.2.3.4.1.4 Drive Shaft (Drive Rod) Assembly

The main function of the drive shaft is to connect the control rod to the mechanism latches. Grooves for engagement and lifting by the latches are located throughout the 144-in. of control rod travel. The grooves are spaced 5/8-in. apart to coincide with the mechanism step length and have 45-degree-angle sides.

The drive shaft is attached to the control rod by the coupling. The coupling has two flexible arms, which engage the grooves in the spider assembly.

A 0.25-in. diameter disconnect rod runs down the inside of the drive shaft. It uses a locking button at its lower end to lock the drive rod assembly and control rod assembly together. At its

upper end there is a disconnect assembly for remote disconnection of the drive rod assembly from the control rod assembly.

During plant operation, the drive shaft assembly remains connected to the control rod at all times. It can be attached and removed from the control rod only when the reactor vessel head is removed.

3.2.3.4.1.5 Position Indicator Coil Stack

The position indicator coil stack slides over the rod travel housing section of the pressure vessel. It detects drive rod position by means of a cylindrically wound differential transformer, which spans the normal length of the rod travel (144-in.).

3.2.3.4.1.6 Drive Mechanism Materials

All parts exposed to reactor coolant, such as the pressure vessel, latch assembly, and drive rod, are made of metals, which resist the corrosive action of the water.

Three types of metals are used exclusively: stainless steels, Inconel-X, and cobalt-based alloys. Wherever magnetic flux is carried by parts exposed to the main coolant, 400 series stainless steel is used. Cobalt-based alloys are used for the pins, latch tips, and bearing surfaces.

Inconel-X is used for the springs of both latch assemblies and type 304 stainless steel is used for all pressure containment. Hard chrome plating provides wear surfaces on the sliding parts and prevents galling between matting parts (such as threads) during assembly.

Outside of the pressure vessel, where the metals are exposed only to the reactor containment environment and cannot contaminate the main coolant, carbon and stainless steels are used. Carbon steel, because of its high permeability, is used for flux return paths around the operating coils. It is zinc-plated approximately 0.001-in. thick to prevent corrosion.

3.2.3.4.1.7 Principles of Operation

The drive mechanisms, shown schematically in Figure 3.2-66, withdraw and insert their respective control rods as electrical pulses are received by the operator coils.

ON and OFF sequence, repeated by switches in the power programmer, causes either withdrawal or insertion of the control rod. Position of the control rod is indicated by the transformer action of the position indicator coil stack surrounding the rod travel housing. The transformer output changes as the top of the ferromagnetic drive shaft assembly moves up the rod travel housing. Generally during plant operation the drive mechanisms hold the control rods withdrawn from the core in a static position and only one coil, the stationary gripper coil, is energized on each mechanism.

Control Rod Withdrawal

The control rod is withdrawn by repeating the following sequence:

1. Movable Gripper Coil - ON

   The movable gripper armature raises and swings the movable gripper latches into the drive shaft groove.

2. Stationary Gripper Coil - OFF

   Gravity causes the stationary gripper latches and armature to move downward until the load of the drive shaft is transferred to the movable gripper latches. Simultaneously, the stationary gripper latches swing out of the shaft groove.

3. Lift Coil - ON

   The 5/8-in. gap between the lift armature and the lift magnet pole closes and the drive rod raises one step length.

4. Stationary Gripper Coil - ON

   The stationary gripper armature raises and closes the gap below the stationary gripper magnetic pole, swings the stationary gripper latches into a drive shaft groove. The latches contact the shaft and lift it 1/16-in. The load is so transferred from the movable to the stationary gripper latches.

5. Movable Gripper Coil - OFF

   The movable gripper armature separates from the lift armature under the force of three springs and gravity. Three links, pinned to the movable gripper armature, swing the three movable gripper latches out of the groove.

6. Lift Coil - OFF

   The gap between the lift armature and the lift magnet pole opens. The movable gripper latches drop 5/8-in. to a position adjacent to the next groove.

Control Rod Insertion

The sequence for control rod insertion is similar to that for control rod withdrawal:

1. Lift Coil - ON

   The movable gripper latches are raised to a position adjacent to a shaft groove.

2. Movable Gripper Coil - ON

   The movable gripper armature raises and swings the movable gripper latches into a groove.

3. Stationary Gripper Coil - OFF

    The stationary gripper armature moves downward and swings the stationary gripper latches out of the groove.

4. Lift Coil - OFF

    Gravity separates the lift armature from the lift magnet pole and the control rod drops down 5/8-in.

5. Stationary Gripper Coil - ON

6. Movable Gripper Coil - OFF

    The sequences described above are termed as one step or one cycle and the control rod moves 5/8-in. for each cycle. Each sequence can be repeated at a rate of up to 72 steps/min and the control rods can therefore be withdrawn or inserted at a rate of up to 45-in./min. The sequence timing has been modified to preclude the rod withdrawal event described in NRC Generic Letter 93-04.

Control Rod Tripping

If power to the movable gripper coil is cut off, as for tripping, the combined weight of the drive shaft and the rod cluster control assembly is sufficient to move the latches out of the shaft groove. The control rod falls by gravity into the core. The tripping occurs as the magnetic field, holding the movable gripper armature against the lift magnet, collapses and the movable gripper armature is forced down by the weight acting upon the latches.

3.2.3.4.2 Part-Length Rods

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3.2.3.5 Fuel Assembly and Rod Cluster Control Assembly Mechanical Evaluation

To confirm the mechanical adequacy of the fuel assembly and full length rod cluster control assembly, functional test programs have been conducted on a full scale Indian Point Unit 2 prototype 12-ft canless fuel assembly and control rod. The prototype assembly was tested under simulated conditions of reactor temperature, pressure, and flow for approximately 1000 hr. The prototype mechanism accumulated 2,260,892 steps and 600 scrams. At the end of the test the control rod drive mechanism was still operating satisfactorily. A correlation was developed to predict the amplitude of flow excited vibration of individual fuel rods and fuel assemblies. Inspection of the fuel assembly and drive line components did not reveal significant fretting. The wear of the absorber rods, fuel assembly guide thimbles, and upper guide tubes was minimal. The control rod free fall time against 125-percent of nominal flow was less than 1.5 sec to the dashpot (10-ft of travel). Additional tests had previously been made on a full scale San Onofre mockup version of the fuel assembly and control rods (Reference 73).

3.2.3.5.1 One-Seventh Scale Mockup Tests

A one-seventh scale model of the Indian Point Unit 2 internals was designed and built for hydraulic and mechanical testing. The tests provided information on stresses and displacements at selected locations on the structure due to static loads, flow induced loads, and electromagnetic shaker loads. Flow distribution and pressure drop information were obtained. Results of the static tests indicated that mean strains in the upper core support plate and upper support columns are below design limits. Strains and displacements measured in the model during flow tests verified that no damaging vibration levels were present. Additional information gained from the tests was the natural frequency and damping of the thermal shield and other components in air and water. Model response can be related to the full scale plant for most of the expected exciting phenomena, but across-the-board scaling is not possible. Specifically exciting phenomena, which are strongly dependent on Reynolds number cannot be scaled. In areas where Reynolds number may be important, either (1) the measured vibration amplitudes were many times lower than a level that would be damaging, or (2) full scale vibration data have been obtained.

3.2.3.5.2 Loading and Handling Tests

Tests simulating the loading of the prototype fuel assembly into a core location have also been successfully conducted to determine that proper provisions had been made for guidance of the fuel assembly during refueling operation.

3.2.3.5.3 Axial and Lateral Bending Tests

Axial and lateral bending tests have been performed in order to simulate mechanical loading of the assembly during refueling operation. Although the maximum column load expected to be experienced in service is approximately 1000 lb, the fuel assembly was successfully loaded to 2200 lb axially with no damage resulting. This information is also used in the design of fuel handling equipment to establish the limits for inadvertent axial loads during refueling.

3.2.4 Fixed Incore Detectors

Eight fixed core neutron detectors are installed within the Unit 2 reactor as shown in Figure 3.2-67. They provide no input to plant instrumentation nor are they needed by the operator. The detectors have been retired and are cut and capped at the seal table.

The detectors are not movable when the primary system is at operating pressure. The assemblies are extracted downward from the core during refueling. The seal table used for the movable detector assemblies is also utilized for the fixed incore detector assemblies.

The installation of the fixed incore detector system is expected to cause no significant reactivity effect. If a fixed detector were to fail, the expulsion of reactor coolant would be accommodated by the charging pumps, which is common for ruptures of a very small cross section. This would enable the operator to execute an orderly shutdown. If a seal were to fail at the seal table, flow from the reactor coolant system would be through the annulus defined by the O.D. of the flux thimble and the I.D. of the conduit (approximately 0.13-in.). The fixed incore detector system is designed to Class I standards such that the likelihood of a failure causing a loss of coolant will be extremely remote.

3.2.4.1 Core Monitoring

Verification of axial and radial power distribution during full power operation is performed using the movable incore detector system. These movable detector locations are shown in Figure 3.2-68. The axial power distribution during operation is determined for each measured thimble location since the activity level is measured at several axial heights for each thimble. Comparison of the measured power distribution to design predictions provides confirmation of safe operation of the reactor and confidence in design predictions. To obtain temperature maps of the core, 65 fuel assembly outlet thermocouples are located as shown in Figure 3.2-68. These thermocouples are located in the upper coolant internals package above the corresponding assemblies. Based on the average activity determined for each thimble, the measured radial power distribution can be determined.

3.2.5 Plant Computer

The computer system provided for Indian Point Unit 2 is a DS&S Plant Integrated Computer System known as “PICS”. It is provided as an adjunct to the normal control room instrumentation to assist the operator in the operation of the reactor by monitoring reactor performance and displaying it in a consistent, well-ordered, usable form. The computer system performs functions such as scanning, signal converting, calculating, indicating, recording, and alarm annunciating. This system is not required for safety, and operation of the reactor is not in any way dependent upon the availability of the computer.

Briefly, the analog scanning includes reading all inputs in a pre-established manner, checking the readings, converting values to engineering units, storing them for future use, updating information, and checking alarm conditions. Some inputs are scanned once every second and status placed in memory; other inputs are not scanned periodically but are given immediate attention. The alarm program compares the values of the inputs against the fixed or variable alarm limits and indicates when off-normal conditions exist.

3.2.6 Current Operating Cycle

Indian Point Unit 2 is currently operating in the twentieth cycle. The core for this cycle uses 15x15 Upgraded fuel. In order to reduce neutron fluence to the reactor vessel shell, a low-low leakage loading pattern, shown in Figure 3.2-68A, is utilized. Integrated fuel burnable absorbers (IFBA) and wet annular burnable absorber (WABA) assemblies, described in References 74 and 88, are used. Figures 3.2-68 and 3.2-68B show the locations of core components and instrumentation. The cycle is designed for a burnup of 25,193 MWd/MTU which includes a power coastdown. The methodology and computer codes described in Reference 75, 82, 87, 96, and 97 were utilized for analysis of the current operating cycle. Fuel Temperatures were calculated using the fuel thermal models of References 69 and 84 (prior to Cycle 16 the analysis was done with Reference 83).

Beginning with Cycle 11, replacement fuel had debris filter bottom nozzles and integral fuel burnable absorbers. To prevent debris from reaching the core, the nozzles have a larger number of smaller holes than previous designs. The integral fuel burnable absorber (IFBA) fuel has a thin layer of zirconium diboride or enriched zirconium diboride coated directly onto selected fuel pellets to control power peaking.

A three dimensional model is used to track and predict core operating characteristics. The code predictions are compared to startup physics test results, measured core flux distributions, and

critical boron concentrations as a function of core burnup. It has been shown to provide a suitable and accurate means for predicting core operating conditions.

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TABLE 3.2-1 (Sheet 1 of 3)
Nuclear Design Data
Cycle 1 Values

STRUCTURAL CHARACTERISTICS

1.	Fuel weight (UO_2), lbs	217,800
2.	Zircaloy weight, lbs	44,600
3.	Core diameter, in.	132.75
4.	Core height, in.	
	Reflector thickness and composition	144
5.	Top water plus steel, in.	~ 10
6.	Bottom water plus steel, in.	~ 10
7.	Side water plus steel, in.	~ 15
8.	H_2O/U, (cold) core	3.91
9.	Number of fuel assemblies	193
10.	UO_2 rods per assembly	204

PERFORMANCE CHARACTERISTICS

11.	Heat output, MWt (initial rating)	2758
12.	Heat output, MWt (maximum calculated turbine rating)	3216
13.	Fuel burnup, MWd/metric ton uranium	16,100
	First cycle enrichments, w/o	
14.	Region 1	2.21
15.	Region 2	2.80
16.	Region 3	3.20
17.	Equilibrium enrichment	3.2
18.	Nuclear heat flux hot channel factor, $F_{Q\ 1}^{T}$	2.32
19.	Nuclear enthalpy rise hot channel factor, $F_{\Delta H}^{N}$	1.55

TABLE 3.2-1 (Sheet 2 of 3)
Nuclear Design Data
Cycle 1 Values

CONTROL CHARACTERISTICS

Effective multiplication (Beginning-of-Life)
with rods in, no Boron

20.	Cold, no power, clean		1.113
21.	Hot, no power, clean		1.057
22.	Hot, full power, clean		1.031
23.	Hot, full power, Xe and Sm equilibrium		1.001
24.	Material		5-percent Cd; 15-percent In; 80-percent Ag
25.	Full length rod cluster control assemblies, number		53
26.	Part length rod cluster control assemblies (removed)		
27.	Number of absorber rods per rod cluster control assemblies		7,8,9,12,16, or 20
28.	Total rod worth, BOL, percent		(See Table 3.2-2)
	Boron concentration for first core cycle loading with burnable poison rods		
29.	Fuel loading shutdown; rods in	(k = .86)	2000 ppm
		(k = .90)	1615 ppm
30.	Shutdown (k = .99) with rods inserted, clean, cold		849 ppm
31.	Shutdown (k = .99) with rods inserted, clean, hot		572 ppm
32.	Shutdown (k = .99) with no rods inserted, clean, hot		1405 ppm
33.	Shutdown (k = .99) with no rods inserted, clean, cold		1370 ppm

TABLE 3.2-1 (Sheet 3 of 3)
Nuclear Design Data
Cycle 1 Values

Boron concentration to maintain k = 1 at hot full power,
No rods inserted:

34.	Clean	1160 ppm
35.	Xenon	860 ppm
36.	Xenon and Samarium	780 ppm
37.	Shutdown, all but one rod inserted, clean, cold (k = .99)	915 ppm
38.	Shutdown, all but one rod inserted, clean, hot (k = .99)	677 ppm

BURNABLE POISON RODS

39.	Number and material	1412 Borated Pyrex Glass
40.	Worth hot Δp	8.2-percent
41.	Worth cold Δp	5.4-percent

KINETIC CHARACTERISTICS

42.	Moderator temperature coefficient at fuel power (°F^{-1})	-0.25×10^{-4} to -3.00×10^{-4}
43.	Moderator pressure coefficient (psi^{-1})	$+0.2 \times 10^{-6}$ to $+3.00 \times 10^{-6}$
44.	Moderator density coefficient, delta k/gm/cm^3	- 0.1 to .30
45.	Doppler coefficient (°F^{-1})	-1.1×10^{-5} to 1.8×10^{-5}
46.	Delayed neutron fraction, percent	0.50 to .72
47.	Prompt neutron lifetime, sec	1.50×10^{-5} to 2.0×10^{-5}

Note:
1. The total flux hot channel factor (F_Q^T) is a generic limit.
   The actual value is presented in the Technical Specifications.

TABLE 3.2-1A
Nuclear Design Data
Cycle 20 Values

		Cycle 19
1.	Heat output, MWt	3216
2.	Fuel loading shutdown boron concentration; rods in ($k \leq 0.95$)	$\geq$2050
3.	Most positive Moderator Temperature Coefficient (pcm/°F)	-9.77
4.	Least Negative Doppler- Only Power Coefficient, Zero to Full Power (pcm/% Power)	-13.35 to -8.53
5.	Most Negative Doppler- Only Power Coefficient, Zero to Full Power (pcm/% Power)	-14.74 to -8.98
6.	Effective average delayed neutron fraction Beff, percent	0.510 to 0.621
7.	Prompt neutron lifetime, µsec	11.60 to 14.17
8.	Design bases minimum shutdown (% Δ p)	1.3
9.	Nuclear Heat Flux Hot Channel Factor (F_q) limit	2.5
10.	Nuclear Enthalpy Rise Hot Channel Factor (FdH) limit	1.7

TABLE 3.2-2
Reactivity Requirements for Control Rods for Cycle 1[1]

Requirements	Percent $\Delta\rho$ Beginning-of-Life	Percent $\Delta\rho$ End-of-Life
Control		
Power defect	1.90	3.05
Operational maneuvering band	0.40	0.40
Control rod bite	0.10	0.10
X-Y xenon rods	0.20	0.20
Total control	2.60	3.75

Notes:
1. Design values used for performing preoperational calculations and analyses.

TABLE 3.2-3
Calculated Rod Worths, $\Delta\rho$ for Cycle 1

Core Condition	Rod Configuration	Worth (Percent)	Less 10-percent[1] (Percent)	Design Reactivity Requirements (Percent)	Shutdown Margin (Percent)
BOL, HFP	53 rods in	8.46			
	52 rods in highest worth rod stuck out	7.43	6.69	2.60	4.09
EOL, HFP	53 rods in	7.98			
	52 rods in; highest worth rod stuck out	6.48	5.83	3.75	2.08[2]

BOL = Beginning-of-life

EOL = End-of-life

HFP = Hot full power

Notes:
1. Calculated rod worth is reduced by 10-percent to allow for uncertainties.
2. The design basis minimum shutdown margin is 1.95-percent.

TABLE 3.2-4
DELETED

TABLE 3.2-5
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TABLE 3.2-6 (Sheet 1 of 3)
Thermal and Hydraulic Design Parameters

	All Hipar Fuel Core[5]	All Lopar Fuel Core[5]	All OFA Fuel Core[5]	All VANTAGE+ Fuel Core[6]
Total heat Output, MWt	2758	2758	3071.4	3216
Total Heat Output, Btu/hr	$9{,}413\times10^6$	$9{,}413\times10^6$	$10{,}483\times10^6$	$10{,}973\times10^6$
Heat generated in fuel, percent	97.4	97.4	97.4	97.4
Nominal system pressure, psia	2250	2250	2250	2250
Coolant Flow				
Total flow rate, $\times 10^6$ lbs/hr[1]	129.57	128.3	121.72	123.3
Avg velocity along fuel rods, ft/sec	14.8	14.6	13.0	13.80
Avg mass flow, $\times 10^6$ lb/hr-ft^2	2.42	2.39	2.21	2.24
Coolant temperature, ^{o}F				
Nominal inlet	541.6	541.3	547.7	538.2
Average rise in vessel	55.8	56.3	64.4	67.6
Average rise in core	58.2	58.7	67.9	71.8
Average in core	571.7	571.7	583.5	575.9
Average in vessel	569.5	569.5	579.7	572.0
Heat transfer				
Active heat transfer surface area,ft^2	51,400	52,100	52,100	52,100
Average heat flux, Btu/hr-ft^2	178,500	176,000	196,000	205,200
Maximum heat flux, Btu/hr-ft^2	414,000	408,300	490,000	513,100

TABLE 3.2-6 (Sheet 2 of 3)
Thermal and Hydraulic Design Parameters

	All Hipar Fuel Core$_5$	All LOPAR Fuel Core$_5$	All OFA Fuel Core$_5$	All VANTAGE+ Fuel Core$_6$
Heat Transfer (continued)				
Maximum thermal output for normal operation, KW/ft	13.4$_2$	13.4$_2$	15.86$_3$	16.6$_3$
Maximum clad surface temperature for normal operation, °F	657	657	663	NA
Fuel central temperatures for nominal fuel rod dimensions, °F				
Maximum at 100-percent power	<4700	<4700	<4700	<4700
DNB ratio				
Minimum DNB ratio at nominal operating conditions (Thimble)	1.95	1.84	2.45	2.40
Typical	NA	NA	2.33	2.50
Pressure drop, psi				
Across core	24.0	25.5	27.2	29.0
Across vessel, including nozzles	50.0	~51.5	~55.0	NA

TABLE 3.2-6 (Sheet 3 of 3)
Thermal and Hydraulic Design Parameters

	3230 MWt (Low Temp Extreme) 7	3230 MWt (High Temp Extreme) 7
NSSS Power, MWt	3230	3230
Core Power, MWt	3216	3216
Thermal Design Flow, Loop gpm	80,700	80,700
Reactor Thermal Design Flow, Total, 10^6 lbm/hr	126.8	123.3
Reactor Coolant Pressure, psia	2250	2250
Reactor Coolant Temperature, °F		
Core Outlet	588.1	610.0
Vessel Outlet	583.7	605.8
Core Average	552.6	575.9
Vessel Average	549.0	572.0
Vessel/Core Inlet	514.3	538.2
Zero Load Temperature	547	547
Percent Tube Plugging	10_8	10_8
Core Bypass Percent	6.5_4	6.5_4

Notes:
1. The thermal design flow rate for all the all HIPAR core reflects a 5-percent flow reduction (to account for postulated 25-percent steam generator tube plugging). The thermal design flow rate for the all LOPAR core reflects a 6-percent flow reduction (5-percent reduction to account for postulated 25-percent steam generator tube plugging and an additional 1-percent reduction to account for an all LOPAR fuel core). For all OFA and all VANTAGE+ Fuel Cores, the thermal design flow rate reflects a 5-percent flow reduction (to account for postulated 10-percent steam generator tube plugging).
2. This power level is based upon a peaking factor (F_q) of 2.32.
3. This power level is based upon a peaking factor (F_q) of 2.50
4. Increased bypass flow is due to thimble plug deletion, and IFMs.
5. This data is historic only.
6. This data reflects the current core with uprated power of 3216 MWt.
7. This data is for analysis extremes covering a range of vessel average temperatures.
8. The tube plugging level is supported by the Thermal-hydraulic safety analyses.

TABLE 3.2-7
Core Mechanical Design Parameters[1]

Fuel assemblies

Number	193
Rod array	15 x 15
Rods per assembly	204[2]
Rod pitch, in.	0.563
Overall dimensions	8.426 x 8.426 HIPAR[6]/LOPAR[6] 8.424 x 8.424 OFA 8.426 x 8.426 VANTAGE+/15x15 Upgraded
Number of grids per assembly (HIPAR[6]/LOPAR[6]/OFA)	9
(VANTAGE+/15x15 Upgraded)	13
Number of instrumentation thimbles	1
Number of guide thimbles	20
Diameter of guide thimbles, upper part, in., HIPAR[6]	0.545 O.D. x 0.515 I.D.
Diameter of guide thimbles, lower part, in., HIPAR[6]	0.484 O.D. x 0.454 I.D.
Diameter of guide thimbles, upper part, in., LOPAR[6]	0.546 O.D. x 0.512 I.D.
Diameter of guide thimbles, lower part, in., LOPAR[6]	0.489 O.D. x 0.455 I.D.
Diameter of guide thimbles, upper part, in., OFA/V+	0.533 O.D. x 0.499 I.D.
Diameter of guide thimbles, lower part, in., OFA/V+	0.489 O.D. x 0.455 I.D.
Diameter of guide thimbles, upper part, in., 15x15 Upgraded	0.533 O.D. x 0.499 I.D.
Diameter of guide thimbles, lower part, in., 15x15 Upgraded	0.487 O.D. x 0.455 I.D.

Fuel rods

Number	39,369 (+3 stainless steel rods)
Outside diameter, in.	0.422
Diametral gap, in.	0.0075
Clad thickness, in.	0.0243
Clad material	Zircaloy (HIPAR[6]/LOPAR[6]/OFA) ZIRLO™ (VANTAGE+/15x15 Upgraded)
Overall length	148.6, HIPAR[6] 151.9, LOPAR[6] 152.17, OFA 152.55, VANTAGE+ 152.88,V+ w/P+ Enhancements/15x15 Upgraded
Length of end cap, overall, in.	0.688, HIPAR[6] 0.265, LOPAR[6] 0.357 OFA/v+ (TOP) 0.430 OFA/V+ (BOTTOM) 0.350,V+w/P+/15x15 Upgraded (TOP) 0.810,V+w/P+/15x15 Upgraded (BOTTOM)
Length of end cap, inserted in rod	0.250, HIPAR[6] 0.200, LOPAR[6] 0.130, OFA/V+/15x15 Upgraded
Active fuel length, in.	142, HIPAR[6] 144, LOPAR[6] 144, OFA/V+/15x15 Upgraded

TABLE 3.2-7 (Cont.)
Core Mechanical Design Parameters[1]

Fuel pellets	
Material	UO_2 sintered
Density (percent of theoretical)	
Region 20B	94.49 15x15 Upgraded
Region 21A	95.85 15x15 Upgraded
Region 21B	95.74 15x15 Upgraded
Region 22A	95.50 15x15 Upgraded
Region 22B	95.50 15x15 Upgraded
Feed enrichments w/o[3]	
Region 20B	4.94 15x15 Upgraded
Region 21A	4.60 15x15 Upgraded
Region 21B	4.96 15x15 Upgraded
Region 22A	4.80 15x15 Upgraded
Region 22B	4.95 15x15 Upgraded
Diameter, in.	0.3659
Length, in.	0.4390 0.500 (Blanket)
Rod cluster control assemblies	
Neutron absorber	5-percent Cd, 15-percent In, 80-percent Ag
Cladding material	Type 304 SS - cold worked
Clad thickness, in.	0.019
Number of clusters	53
Number of control rods per cluster	20
Length of rod control, in.	156.436 (overall)
Length of absorber section, in.	142.00

TABLE 3.2-7 (Cont.)
Core Mechanical Design Parameters[1]

Core structure

Core barrel	
I.D., in.	148.0
O.D., in.	152.5
Thermal shield	
I.D., in.	158.5
O.D., in.	164.0

Wet Annular Burnable Absorber (WABA) Rods

Number	640
Pellet Stack Length	128”
Pellet Material	Al_2O_3-B_4C
Boron Loading(Natural)	.0243 g/cm
(B-10)	.0060 g/cm
Pellet O.D. /I.D.	.318"/.278"
Tube material	ZIRLO™
Outer tube O.D. /I.D.	.3810"/.3290"
Inner tube O.D. /I.D.	.2670"/.2250"

Integral Fuel Burnable Absorber (IFBA) Rods

Number	5396
Absorber	Zirconium Diboride
B-10 Loading (mg/inch)[4]	2.65 (1.5X)
	2.21 (1.25X)
IFBA Coating Length	120/128inches

Notes:
1. All dimensions are for cold conditions. Data is for all fuel types unless otherwise stated.
2. Twenty-one rods are omitted: Twenty provide passage for control rods and one contains incore instrumentation.
3. Reload fuel regions have variable enrichments depending on energy requirements, the number of assemblies being fed, and the degree of low leakage (i.e. number of feed assemblies on the periphery).
4. Nominal values.
5. Deleted
6. Symbol representing old and removed fuel assemblies.

3.2 FIGURES

3.3 TESTS AND INSPECTIONS

3.3.1 Reactivity Anomalies

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burnup and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to reflect actual core conditions. When full power is reached initially, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burnup and reactivity was compared with that predicted. This process of normalization is completed before a cycle burnup of 60 Effective Full Power Days (EFPDs) is reached. Thereafter, actual boron concentration is compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1-percent would be unexpected, and its occurrence would be thoroughly investigated and evaluated. The methods employed in calculating the reactivity of the core versus burnup and the reactivity worth of boron versus burnup are given in Section 3.2.1.

3.3.2 Thermal And Hydraulic Tests And Inspections

General hydraulic tests on models were used to confirm the design flow distributions and pressure drops.[1,2] Fuel assemblies and control and drive mechanisms were also tested. Onsite measurements were made to confirm the design flow rates.

Vessel and internals inspections were also reviewed to check such thermal and hydraulic design values as bypass flow. As part of startup physics testing, a series of core power distribution measurements were made over the entire range of operation in terms of design control rod configuration by means of the core movable detector system. These measurements were analyzed and the results compared with the analytical predictions upon which the safety analysis was based with regard to both radial and axial power distribution. The design hot-channel factors were used as criteria for acceptable results.

3.3.3 Core Component Tests And Inspections

To ensure conformance of all materials, components, and assemblies to the design requirements, a release point program is established with the assembly manufacturer, which requires upgrading of all raw materials, special processes (i.e., welding, heat treating, nondestructive testing, etc.) and those characteristics of detail parts, which directly affect the assembly and alignment of the reactor internals. The upgrading is accomplished by the issuance of an inspection release by quality control after conformance has been verified.

A resident quality control representative performs a surveillance/audit program at the manufacturer's facility and witnesses the required tests and inspections and issues the inspection releases. An example is the radiographic examination of the welds joining core barrel shell courses.

Components and materials supplied by Westinghouse to the assembly manufacturer are subjected to a similar program. Quality control engineers develop inspection plans for all raw materials, components, and assemblies. Each level of manufacturing is evaluated by a qualified inspector for conformance, i.e., witnessing the ultrasonic testing of core plant raw material. Upon completion of specified events, all documentation is audited prior to releasing the material or component for further manufacturing. All documentation and inspection releases are maintained in the quality control central records section. All materials are traceable to the mill heat number.

In conclusion, a set of "as-built" dimensions are taken to verify conformance to the design requirements and assure proper fit-up between the reactor internals and the reactor pressure vessel.

3.3.3.1 Quality Assurance Program

The quality assurance program plan of the Westinghouse Nuclear Fuel Division is summarized in Reference 3.

The program provides for control over all activities affecting product quality, commencing with design and development and continuing through procurement, materials handling, fabrication, testing and inspection, storage, and transportation. The program also provides for the indoctrination and training of personnel and for the auditing of activities affecting product quality through a formal auditing program.

Westinghouse drawings and product, process, and materials specifications identify the inspections to be performed.

3.3.3.2 Quality Control

Quality control philosophy is generally based on the following inspections being performed to a 95-percent confidence that at least 95-percent of the product meets specification, unless otherwise noted.

1. Fuel system components and parts

    The characteristics inspected depend upon the component parts; the quality control program includes dimensional and visual examinations, check audits of test reports, material certification, and nondestructive examination, such as X-ray and ultrasonic.

    All material used in this core is accepted and released by quality control.

2. Pellets

Inspection is performed for dimensional characteristics such as diameter, density, length, and squareness of ends. Additional visual inspections are performed for cracks, chips, and surface conditions according to approved standards.

Density is determined in terms of weight per unit length and is plotted on zone charts used in controlling the process. Chemical analyses are performed on a specified sample basis throughout pellet production.

3. Rod inspection

The fuel rod inspection consists of the following nondestructive examination techniques and methods, as applicable:

a. Each rod is leak tested using a calibrated mass spectrometer, with helium being the detectable gas.

b. Rod welds are inspected by ultrasonic test or X-ray in accordance with a qualified technique and Westinghouse specification.

c. All rods are dimensionally inspected prior to final release. The requirements include such items as length, camber, and visual appearance.

d. All fuel rods are inspected by gamma scanning or other approved methods to ensure proper plenum dimensions.

e. All fuel rods are inspected by gamma scanning, or other approved methods to ensure that no significant gaps exist between pellets.

f. All fuel rods are active gamma scanned to verify enrichment control prior to acceptance for assembly loading.

g. Traceability of rods and associated rod components is established by quality control.

4. Assemblies

Each fuel assembly is inspected for compliance with drawing and/or specification requirements. Other incore control component inspection and specification requirements are given in paragraph 4.2.3.4 of Reference 3.

5. Other inspections

The following inspections are performed as part of the routine inspection operation:

a. Tool and gauge inspection and control, including standardization to primary and/or secondary working standards. Tool inspection is performed at

prescribed intervals on all serialized tools. Complete records are kept of calibration and conditions of tools.

b. Audits are performed of inspection activities and records to ensure that prescribed methods are followed and that records are correct and properly maintained.

c. Surveillance inspection, where appropriate, and audits of outside contractors are performed to ensure conformance with specified requirements.

6. Process control

a. To prevent the possibility of mixing enrichments during fuel manufacture and assembly, strict enrichment segregation and other process controls are exercised.

The uranium-dioxide powder is kept in sealed containers. The contents are fully identified both by descriptive tagging and preselected color coding. A Westinghouse identification tag completely describing the contents is affixed to the containers before transfer to powder storage. Isotopic content is confirmed by analysis.

Powder withdrawal from storage can be made by only one authorized group, which directs the powder to the correct pellet production line. All pellet production lines are physically separated from each other and pellets of only a single nominal enrichment and density are produced in a given production line at any given time.

Finished pellets are placed on trays identified with the same color code as the powder containers and transferred to segregated storage racks within the confines of the pelleting area. Samples from each pellet lot are tested for isotopic content and impurity levels prior to acceptance by quality control. Physical barriers prevent mixing of pellets of different nominal densities and enrichments in this storage area. Unused powder and substandard pellets are returned to storage in the original color-coded containers.

Loading of pellets into the clad is performed in isolated production lines, and again only one enrichment and density loaded on a line at a time.

A serialized traceability code is placed on each fuel tube to provide unique identification. The end plugs are inserted and then inert-welded to seal the tube. The fuel tube remains coded and traceability identified until just prior to installation in the fuel assembly.

At the time of installation into an assembly, the traceability codes are removed and a matrix is generated to identify each rod in its position within a given assembly. The top nozzle is inscribed with a permanent identification number providing traceability to the fuel contained in the assembly.

Similar traceability is provided for burnable poison, source rods, and control rodlets, as required.

REFERENCES FOR SECTION 3.3

1. G. Hetsroni, "Hydraulic Tests of the San Onofre Reactor Model, "WCAP-3269-8, Westinghouse Electric Corporation, 1964.

2. G. Hetsroni, "Studies of the Connecticut-Yankee Hydraulic Model, "WCAP-2761, Westinghouse Electric Corporation, 1965.

3. J. Moore, "Nuclear Fuel Division Quality Assurance Program Plan;" WCAP-7800, Revision 5, Westinghouse Electric Corporation, November 1979.

Appendix 3A
EXPERIMENTAL VERIFICATION OF CALCULATIONS FOR BORON BURNABLE POISON RODS

A number of experiments were performed at the Westinghouse Reactor Evaluation Center to investigate the reactivity worth of Pyrex glass tubing similar to that employed in the Indian Point Unit 2 core as burnable poison rods. Several configurations with and without glass burnable poison rods and with fuel loadings representative of power reactors were tested. The reactor used was a rectangular core 4-ft high with 29 or 30 fuel rods on a side. In each case the water height was adjusted until the reactor was just critical.

Analyses were performed for each of the configurations measured to determine the adequacy of the methods used to calculate burnable poison rod worths in the design of the Indian Point Unit 2 core. The results of the calculations for the different experimental configurations are listed in Table 3A-1. In each case the eigenvalue should be compared to the appropriate reference eigenvalue (core with fuel only) to eliminate the systematic bias, which appears in the clean core calculation. The discrepancy between the eigenvalue calculated for the unpoisoned and poisoned cases has been related to the fractional error in the neutron current into the boron. This error is also given in Table 3A-1.

The burnable poison rods used in Indian Point Unit 2 correspond to the thick-walled tubes and in these cases the agreement is generally better than 5-percent.

TABLE 3A-1 (Sheet 1 of 2)
Calculations and Burnable Poison Rod Worths[1]

Case	Configuration	Loading	Just Critical Water Height, cm	Calculated K_{eff} For Critical Core	Error in Poison Absorption, Percent
1	No inserts - clean	25^2 = 625	64.10		
2	One solid glass rod	25^2-1 = 624	70.00		
3	No inserts clean core	29^2 = 841	43.25	.996915[2]	
4	Uniform thin wall glass	29^2-25 = 816	69.61	.994633	4.6
5	Uniform thick wall glass	29^2-25 = 816	83.81	.994800	3.5
6	Central assembly pattern - 72 water holes	29^2-72 = 769	39.52	1.000085	
7	Central assembly pattern - 36 thin wall glass 36 water holes	29^2-72 = 769	83.76	.996162	1.4
8	Central assembly pattern - 36 thick wall glass 36 water holes	29^2-72 = 769	114.95	.996392	0.8
9	4 water holes array	29^2-16 = 825	42.25		
10	No inserts - clean	30^2 = 900	40.69	.996596[2]	

TABLE 3A-1 (Sheet 2 of 2)
Calculations and Burnable Poison Rod Worths[1]

Case	Configuration	Loading	Just Critical Water Height, cm	Calculated K_{eff} For Critical Core	Error in Poison Absorption, Percent
11	Uniform water holes	30^2-36 = 864	38.91	.99777	
12	Uniform voids (SS cladding)	30^2-36 = 864	42.62		
13	Uniform thin wall glass tubes	30^2-36 = 864	67.18	.993915	4.8
14	Uniform thick wall glass tubes	30^2-36 = 864	82.56	.994875	2.5
15	Uniform - .260-in. Ag-In-Cd	30^2-32 = 868	92.34		
16	Alternate thin wall glass tubes	30^2-18 = 882	49.42	.994963	6.2
17	Alternate thick wall glass tubes	30^2-18 = 882	51.90	.994965	5.1
18	Alternate solid glass rods - bare	30^2-18 = 882	54.91	.995925	1.8
19	Alternate pattern - Ag-In-Cd (.260)	30^2-18 = 882	53.89		
20	Alternate pattern - Ag-In-Cd (.330)	30^2-18 = 882	59.94		
21	Four assembly pattern - water holes	30^2-80 = 820	37.13	.999352	
22	Four assembly pattern - 40 thick wall glass 40 water holes	30^2-80 = 820	68.73	.997573	1.8

Notes:
1. Reactivity data of glass rods, encased in 0.020-in. SS 0.395-in. I.D., 0.435-in. O.D.
Nominal glass rod data: solid rod, 0.396-in. O.D., L = 48-in.; thick wall I.D. = 0.228-in., O.D. = 0.375-in., L = 48-in. thin wall I.D. = 0.316-in., O.D. = 0.394-in., L = 48-in.

2. Reference case.

Appendix 3B
POWER DISTRIBUTION CONTROL

3B.1 GENERAL

The spatial stability of the xenon distribution in large PWRs has been the subject of extensive investigation by Westinghouse. General studies (performed in part under the Euratom Xenon Program) are reported in WCAP-3680-20, 21, 22, and 23[1-4] and specific studies related to the Indian Point Unit 2 reactor are reported in WCAP-7407-L[5] (Westinghouse Proprietary). Confidence that the reactor can be maintained within thermal limits (design nuclear hot channel factors) is provided by the following:

1. Results of the extensive analytical investigation of potential spatial instability arising from redistribution of xenon in the Indian Point Unit 2 reactor lead to the conclusions that (a) the reactor may be unstable toward axial spatial oscillations and (b) is stable toward radial or diametral (quadrant to quadrant) xenon spatial oscillations.

2. Stability towards diametral (X-Y) xenon oscillations was demonstrated during Cycle 1 startup tests per a report submitted by Con Edison to the NRC (Reference 6).

3. Continuous monitoring and appropriate alarm functions of both axial and diametral power tilts, using signals from the eight ex-core ion chambers, with additional information provided by the core exit thermocouples and moveable incore flux detectors.

4. Since the core is expected to be X-Y stable, automatic protection against diametral transients is not required. However, an alarm function is provided to alert the operator to the existence of such tilts before a limiting value on diametral power tilt is reached.

   Stability toward diametral oscillations was verified at startup. As burnup progresses, the reactor becomes increasingly stable toward diametral oscillations due to the decreasing soluble boron concentration and hence the continuously increasing moderator temperature coefficient feedback effect.

5. Control rod cluster malpositioning even under the most limiting case will not lead to a DNBR = 1.30 at operating conditions. Means for detecting such a misalignment are also provided.

3B.2 SPATIAL XENON STABILITY

3B.2.1 <u>AXIAL XENON STABILITY</u>

The potential existence of axial power distribution anomalies due to xenon redistribution have been reported in WCAP-7208.[7] Results of these studies have shown that the reactor will be unstable toward xenon oscillations in this dimension; consequently, power shaping devices (i.e., control rods) and automatic protection (i.e., trip setpoint reduction with excessive axial power imbalance) are provided. Operating philosophy and procedures for monitoring and controlling axial power anomalies have been described in References 7 and 8. The primary

8. The primary means of detecting axial power distortions will be by means of the ex-core ion chambers with appropriate operator display signals. Tests in the Connecticut Yankee reactor (References 5 and 9) have verified the capability of these ex-core ion chambers to detect significant axial power imbalances.

3B.2.2 DIAMETRAL XENON STABILITY

Results of the analytical investigations (primarily three-dimensional transient analyses reported in References 3 and 5) indicate that the Indian Point Unit 2 reactor will be stable toward diametral xenon oscillations; consequently, X-Y control rods are not required. Comparison with experimental results in the Connecticut Yankee reactor tend to confirm the validity of the less conservative calculations (see Reference 5, Figure 3-1). A test was performed at startup to demonstrate that artificially induced diametral oscillations decrease in amplitude as a function of time. Furthermore, extensive monitoring with appropriate display and alarm function is provided to alert the operators in the event a diametral power tilt should develop in the course of reactor operation. Consequently, no automatic safety protection against diametral xenon instability is required.

3B.2.3 ANALYTICAL TECHNIQUES

In assessing potential power distribution anomalies arising from spatial xenon redistribution, primary reliance has been placed on time-dependent two-group diffusion calculations in three-dimensions including pointwise feedback effects due to coolant density and fuel pellet temperature changes. Means of incorporating the reactivity feedback effects are described in References 2 and 3 using semi-empirically fitted expressions whose coefficients were determined by other calculations (e.g., LEOPARD). In some cases, survey calculations were performed in one or two dimensions using both digital and modal techniques (see Reference 1), to indicate trends and to identify the significance and relative importance of the various contributing parameters.

In performing three-dimensional time dependent stability analyses, standard design techniques (i.e., the LEOPARD Code) were used to compute the effect of the various feedback parameters on local reactivity. These results were fitted by a semi-empirical expression as described in Sections 2.2 and 3.3 of Reference 2. These analytical fits, with appropriate coefficients as determined from LEOPARD type calculations, were then used in the three-dimensional spatial power calculations, which included coupled thermal hydraulic effects.

3B.2.4 INSTRUMENTATION AND CONTROL

Instrumentation and appropriate display is provided to ensure that the reactor will be maintained within thermal limits (design hot nuclear channel factors) in the presence of power distribution anomalies caused by time-dependent xenon redistribution. Primary reliance is placed on the eight ex-core ion chambers supplemented by information derived from the core exit thermocouples and from the movable incore fission chambers.

The operator will have the ex-core detector information available, backed up by the core exit thermocouples and the movable incore detector readouts.

The following ex-core detector information is provided for the operator to alert him to the existence of any core instabilities, axial or diametral:

1. Four indicators, which indicate the difference between the top and bottom detectors. These signals will initiate alarms.

2. Eight indicators, which read out the individual currents of the four top and four bottom detectors.

3. One alarm for the four top detectors when the maximum to average flux is exceeded.

4. One alarm for the four bottom detectors when the maximum to average flux is exceeded.

5. Four 2-pen recorders; two detectors at 180 degrees are on the same recorder.

6. Two 2-pen recorders; power level proportional to total current, i.e., combined top and bottom detector outputs.

7. One total current deviation alarm, i.e., when any one top and bottom total current deviates by a pre-set amount from the other three total current outputs, the operator is alerted to this condition.

With these indications and alarms, the operator has many cross-checks and comparisons available to him. Failure of one top or bottom detector will provide the operator with instant indication and alarm. The ex-core detectors, backed by the movable incore detectors, provide more than adequate information. Operation with one ex-core ion chamber out of service does not compromise the safety of the plant.

3B.3 CONTROL ROD POSITIONING

Normal control rod operations have been described in Section 3.0. A deviation in the position of one or more control clusters relative to the position of the control bank can potentially lead to:

1. Asymmetric fuel depletion.
2. Reduction in shutdown margin.
3. Reduction in DNB margin.

Rod misalignment is not a safety problem, which requires automatic protection because (1) asymmetric fuel depletion could possibly lead to unacceptable power distributions, but only if the condition were to persist for many hundreds of hours, (2) misalignment of sufficient magnitude to consume the standard 1-percent Δk shutdown is not possible, because it would require an entire control bank to be several feet below the desired position; the complete misalignment of a single control cluster will reduce trip reactivity by not more than 0.2-percent Δk; and (3) misalignment of a single control cluster by as much as the entire height of the core with the most pessimistic xenon spatial distribution will not result in a DNBR less than 1.30 at operating conditions. Deviation of 15-in. will not result in a power distribution worse than design.

Misalignment of a rod is most limiting when the last control group (which may be partly inserted at full power) is fully inserted but one cluster is full-out. It has been shown for Indian Point Unit 2 (Reference 5) that this case cannot lead to DNBR less than 1.30 at operating conditions even with the worst possible xenon distribution and the control bank (less one cluster) fully inserted.

Each control cluster has its own position indicator channel. The rod position indicator channel is sufficiently accurate to detect a rod ±7.5-in. away from its demand position for indicated control rod position less than or equal to 210 steps withdrawn. An indicated misalignment ≤12 steps does not exceed the power peaking factor limits. A misaligned rod of +17 steps allows for greater instrumentation error when indicated control rod position is greater than or equal to 211 steps withdrawn. The reactivity worth of a rod at this core height (211 + steps) is not sufficient to perturb power shapes to the extent that peaking factors are affected.

The rod position indication system is the primary source of rod position information, but additional means, namely, ex-core ion chambers and movable incore fission chambers, are available.

Except for the central control rod cluster, a power tilt will result from any significant control rod misalignment and such a power tilt would be detected by the ex-core ion chambers. Also, the movable incore fission chamber system can also be used to detect and/or investigate a suspected control rod malpositioning.

REFERENCES FOR APPENDIX 3B

1. C. C. Poncelet and A. M. Christie, "Xenon-Induced Spatial Instabilities in Large Pressurized Water Reactors," WCAP-3680-20, Westinghouse Electric Corporation, March 1968.

2. F. B. Skogen and A. F. McFarlane, "Control Procedures for Xenon-Induced X-Y Instabilities in Large Pressurized Water Reactors," WCAP-3680-21, Westinghouse Electric Corporation, February 1969.

3. F. B. Skogen and A. F. McFarlane, "Xenon-Induced Spatial Instabilities in Three-Dimensions," WCAP-3680-22, Westinghouse Electric Corporation, September 1969.

4. A. M. Christie, et al., "Control of Xenon Instabilities in Large Pressurized Water Reactors," WCAP-3680-23, Westinghouse Electric Corporation, September 1969.

5. R. F. Barry, et al., "Power Maldistribution Investigations," WCAP-7407-L (Proprietary Class 2), Westinghouse Electric Corporation.

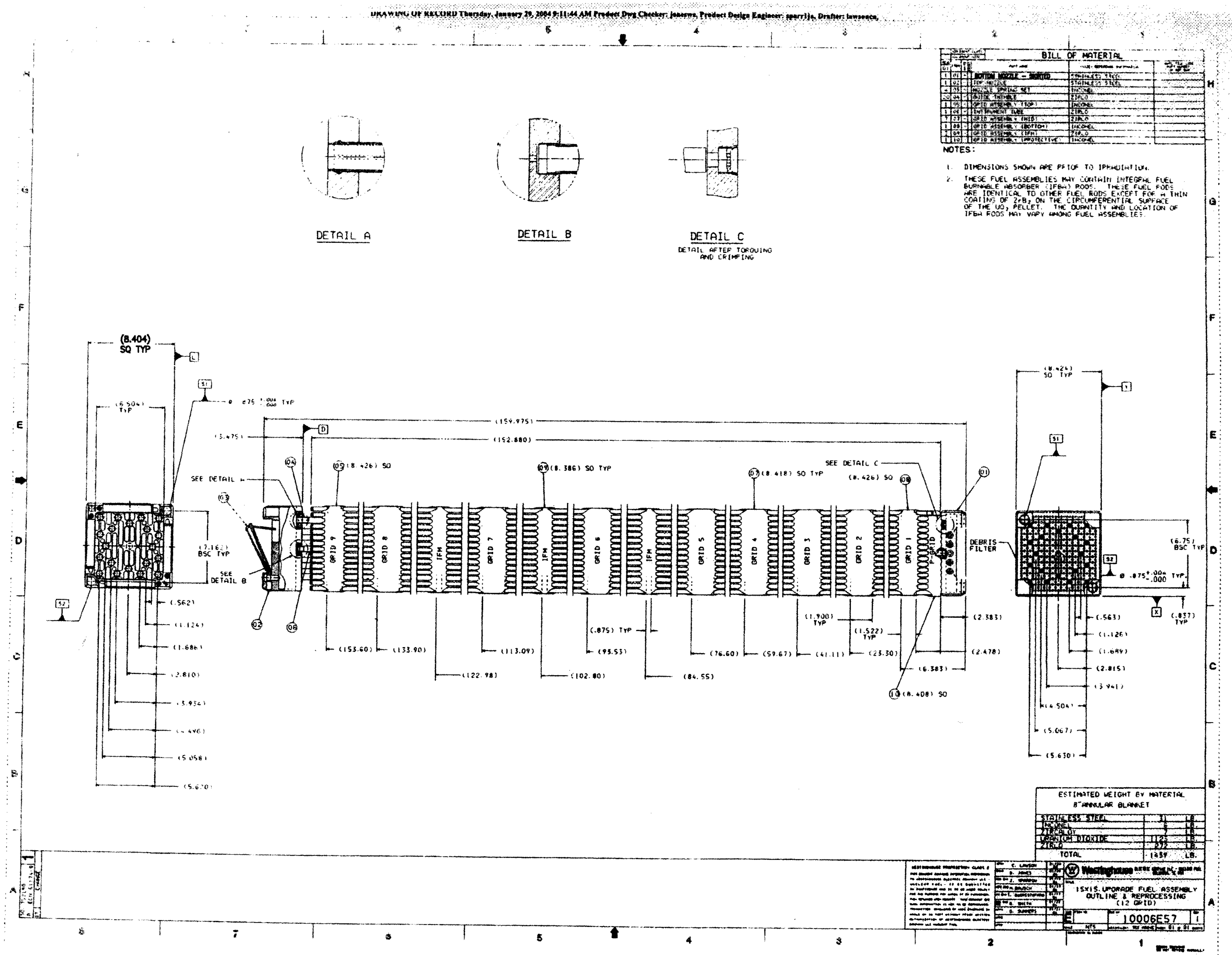
6. Letter from C. Newman, Con Ed, to AEC, Subject: Indian Point Unit No. 2 Results of the X-Y Xenon Stability Tests, dated October 17, 1974.

7. Westinghouse Electric Corporation, "Power Distribution Control of Westinghouse Pressurized Water Reactors," WCAP-7208 (APD Proprietary Class 2), September 1968.

8. T. Morita, et al., "Power Distribution Control and Load Following Procedures," WCAP-

WCAP-8385 (Proprietary), September 1974 and WCAP-8403 (Non-Proprietary), September 1974.

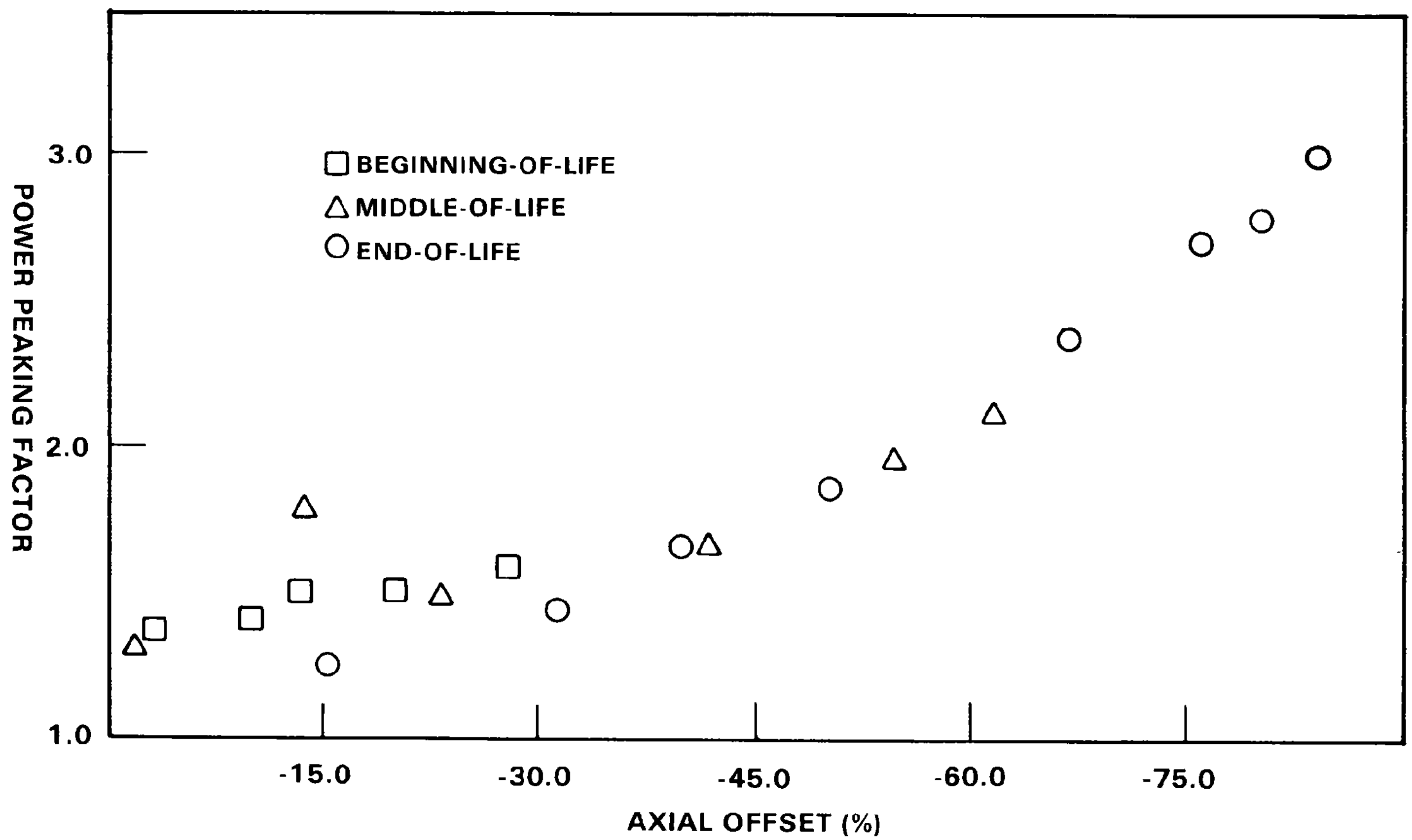
9. R. J. Johnson, "Connecticut-Yankee Tests on Detection of Power Maldistribution," WCAP-9010 (NES Proprietary Class 2), Westinghouse Electric Corporation, February 1969.



INDIAN POINT UNIT No. 2

15 X 15 UPGRADED FUEL ASSEMBLY

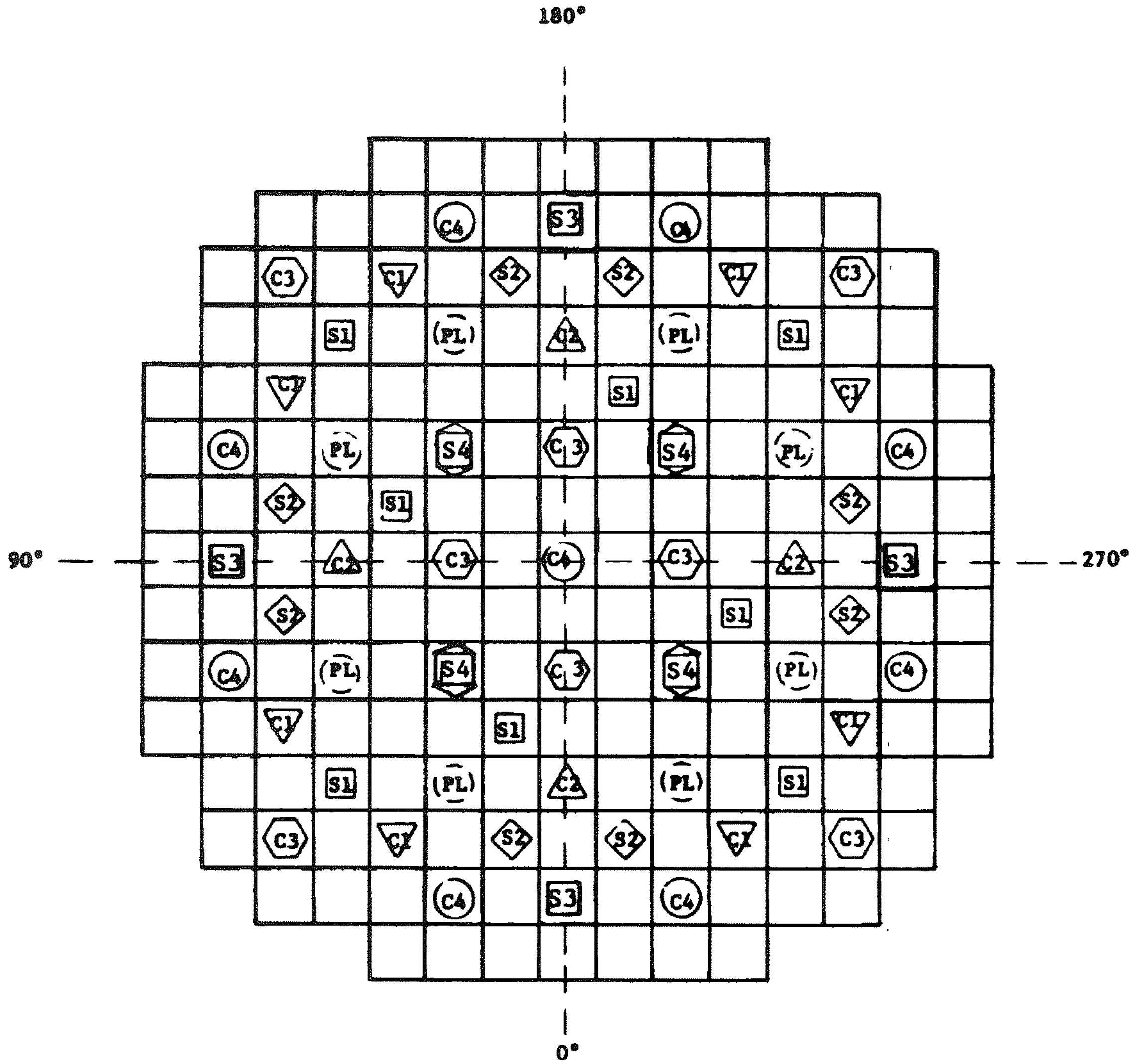
UFSAR FIGURE 3.2-61C | REV. No. 19



INDIAN POINT UNIT No. 2

UFSAR FIGURE 3.2-1
TYPICAL
POWER PEAKING FACTOR
VERSUS AXIAL OFFSET

MIC. No. 1999MC3583	REV. No. 17B



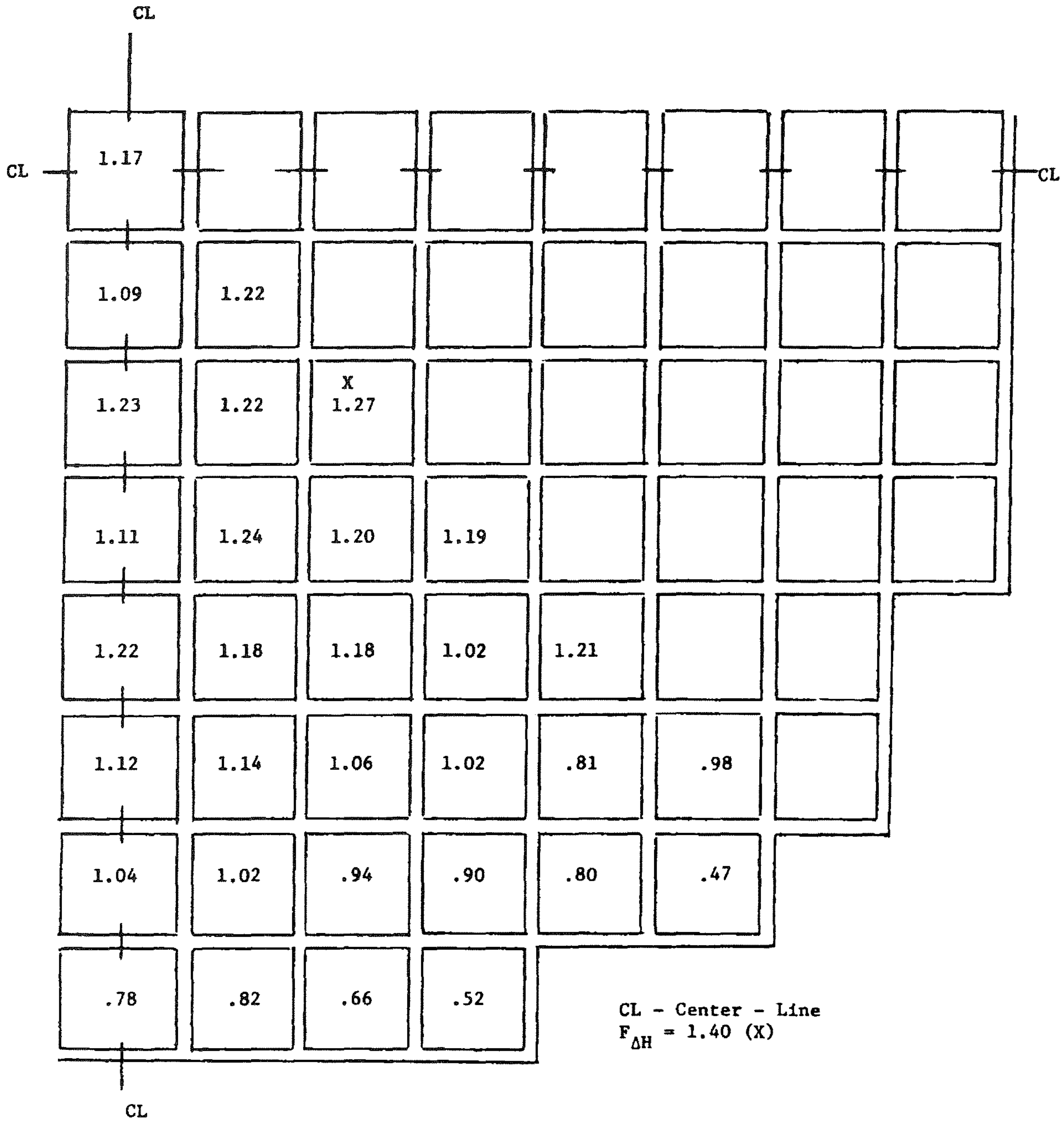
GROUP	SYMBOL	NUMBER OF ROD CLUSTERS
S1	□	8
S2	◇	8
S3	▭	4
S4	⬣	4
C1	▽	8
C2	△	4
C3	⬡	8
C4	○	9
PL (Part Length)	◌	8
		61

INDIAN POINT UNIT No. 2

UFSAR FIGURE 3.2-2

ROD CLUSTER GROUPS – CYCLE 1

MIC. No. 1999MC3584 | REV. No. 17A



INDIAN POINT UNIT No. 2

UFSAR FIGURE 3.2-3

ASSEMBLY AVERAGE POWER & BURNUP, CYCLE 1 CALCULATIONS, BOL, UNRODDED CORE

MIC. No. 1999MC3585	REV. No. 17A

CL

CL								CL
	1.03 16152							
	1.07 16279	1.03 16438						
	1.03 16468	1.08 17190	1.04 16705					
	1.08 16402	1.04 16539	1.11 17114	1.09 16281				
	1.06 16302	1.12 16864	1.08 16120	1.14 15741	X 1.19 17011			
	1.12 16249	1.08 15632	1.13 15716	1.08 14680	1.08 13526	1.04 13811		
	1.02 14068	1.12 14974	1.00 13271	1.04 13315	.88 11434	.58 7032		
	.79 10620	.81 10984	.76 9472	.60 7607				

CL

CL - Center - Line

$F_{\Delta H}$ = 1.43 (X)

INDIAN POINT UNIT No. 2

UFSAR FIGURE 3.2-4

ASSEMBLY AVERAGE POWER & BURNUP, CYCLE 1 CALCULATIONS, EOL, UNRODDED CORE

MIC. No. 1999MC3586 | REV. No. 17A

0.671	1.090	1.439	1.343	1.362	1.113	0.882	0.603
1.090	1.340	1.447	1.487	1.309	1.088	0.777	0.579
1.439	1.447	1.544 $F_{\Delta H}$ → X F_N → X	1.437	1.308	0.944	0.424	0.401
1.343	1.487	1.437	1.407	1.160	1.010	0.701	0.374
1.362	1.309	1.308	1.160	1.367	0.948	0.812	
1.113	1.088	0.944	1.010	0.948	1.109	0.535	
0.882	0.777	0.424	0.701	0.812	0.535		
0.603	0.579	0.401	0.374				

INDIAN POINT UNIT No. 2

UFSAR FIGURE 3.2-5

ASSEMBLY AVERAGE POWER DISTRIBUTION
CYCLE 1 CALCULATIONS, BOL,
GROUP C4 INSERTED

MIC. No. 1999MC3587 | REV. No. 17A

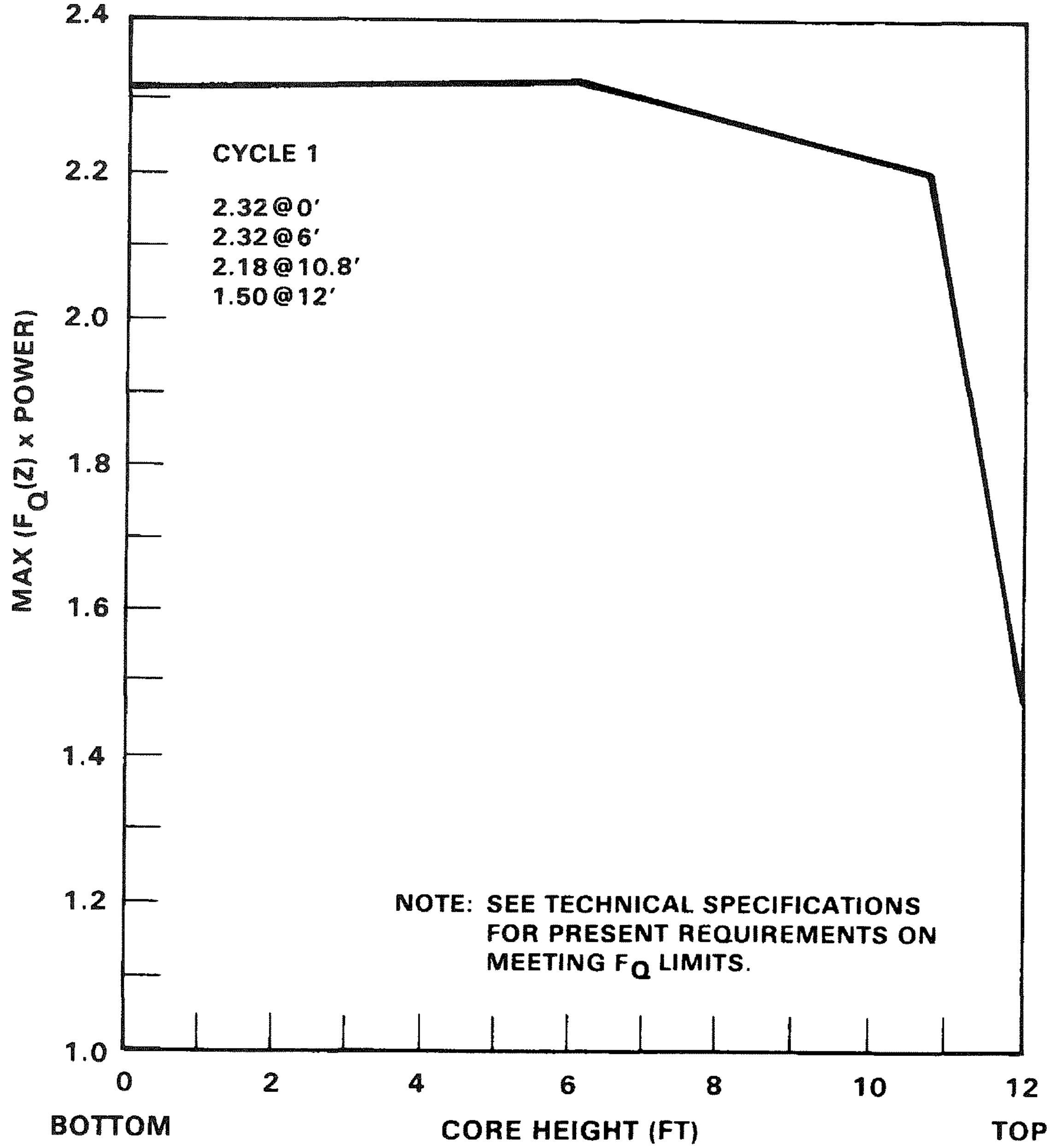
X X 1.402 X X	← $F^{N}_{\Delta H}$ 1.290	1.349	1.130	1.169	1.157	1.171	0.917
1.290	1.395	1.286	1.161	0.989	1.110	1.149	0.967
1.349	1.286	1.227	0.962	0.568	0.929	1.045	0.799
1.130	1.161	0.962	0.940	0.830	0.999	1.000	0.633
1.169	0.989	0.568	0.830	1.189	0.954	0.938	
1.157	1.110	0.929	0.999	0.954	1.159	0.584	
1.171	1.149	1.045	1.000	0.938	0.584		
0.917	0.867	0.799	0.633				

INDIAN POINT UNIT No. 2

UFSAR FIGURE 3.2-6

ASSEMBLY AVERAGE POWER DISTRIBUTION
CYCLE 1 CALCULATIONS, BOL,
PART-LENGTH RODS IN

MIC. No. 1999MC3588 | REV. No. 17A

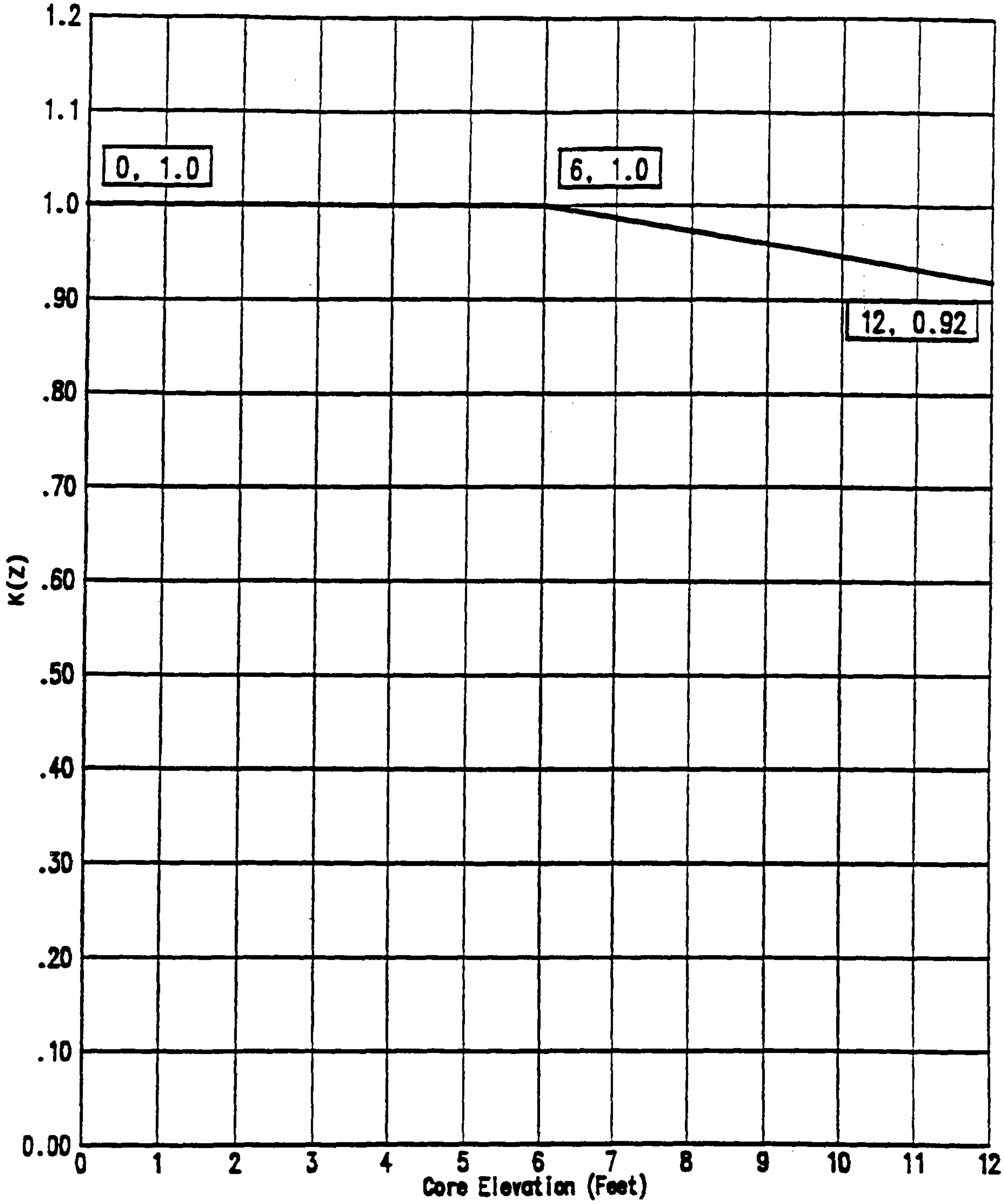


INDIAN POINT UNIT No. 2

UFSAR FIGURE 3.2-7

CYCLE 1 MAXIMUM F_Q x POWER VERSUS AXIAL HEIGHT DURING NORMAL OPERATIONS

MIC. No. 1999MC3589	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 3.2-7A NORMALIZED K (z) - Fq VERSUS AXIAL HEIGHT FOR CYCLE 16	
MIC. No. 1999MC3590	REV. No. 17B

15	14	13	12	11	10	9	8	7	6	5	4	3	2	1
					9		S 7		9					
		8		12		16		16		12		8		
	8		20		12		16		12		20		8	
		20		20		16		16		20		20		
	12		20		16		16		16		20		12	
9		12		16		20		20		16		12		9
	P 16		16		20		16		20		16		P 16	
7		16		16		16		16		16		16		7
	P 16		16		20		16		20		16		P 16	
9		12		16		20		20		16		12		9
	12		20		16		16		16		20		12	
		20		20		16		16		20		20		
	8		20		12		16		12		20		8	
		8		12		16		16		12		8		
					9		S 7		9					

TOTAL 1412

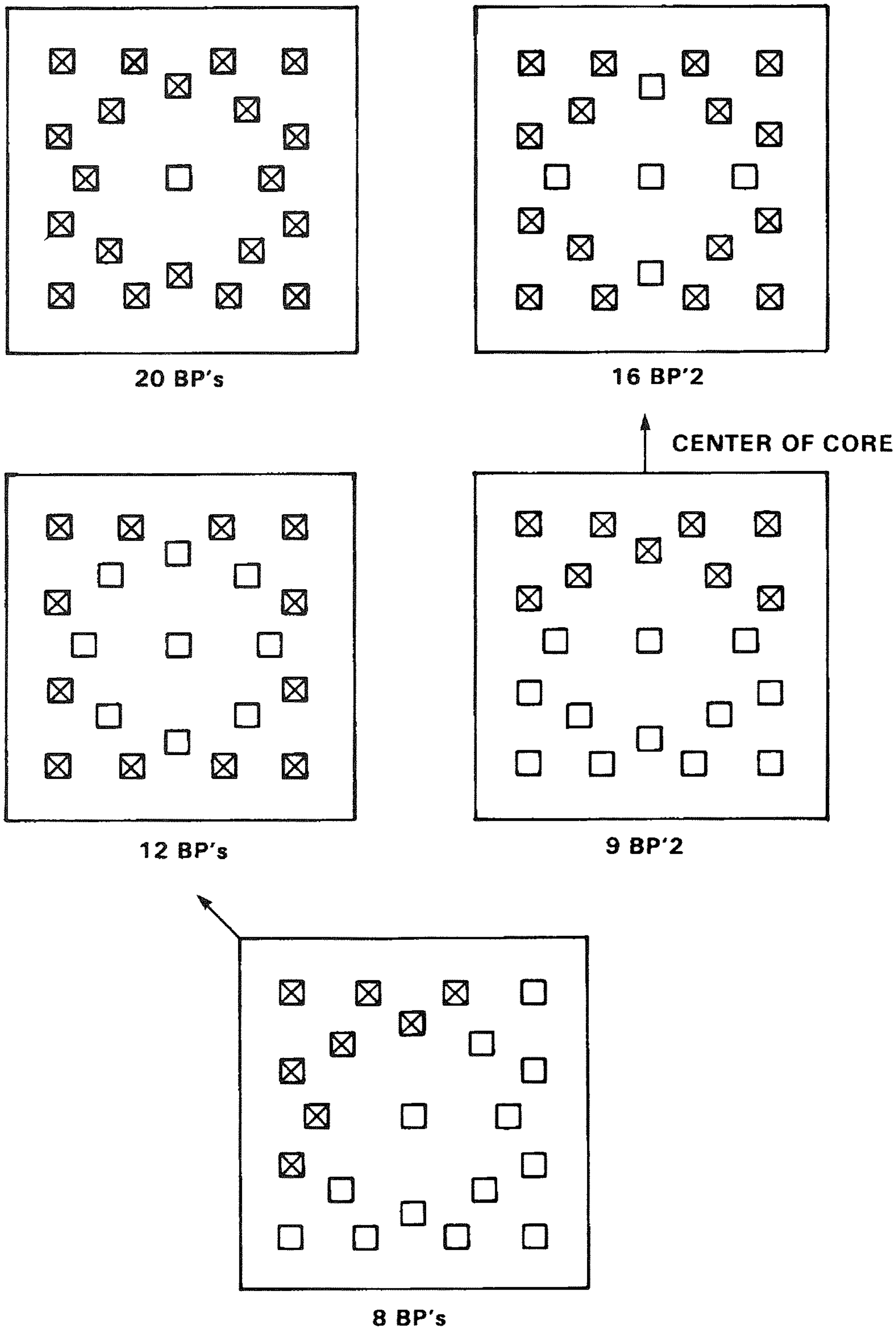
P - PRIMARY/SECONDARY SOURCE COMBINATION
S - SECONDARY SOURCE;

INDIAN POINT UNIT No. 2

UFSAR FIGURE 3.2-8

BURNABLE POISON & SOURCE ASSEMBLY LOCATIONS – CYCLE 1

MIC. No. 1999MC3591	REV. No. 17A

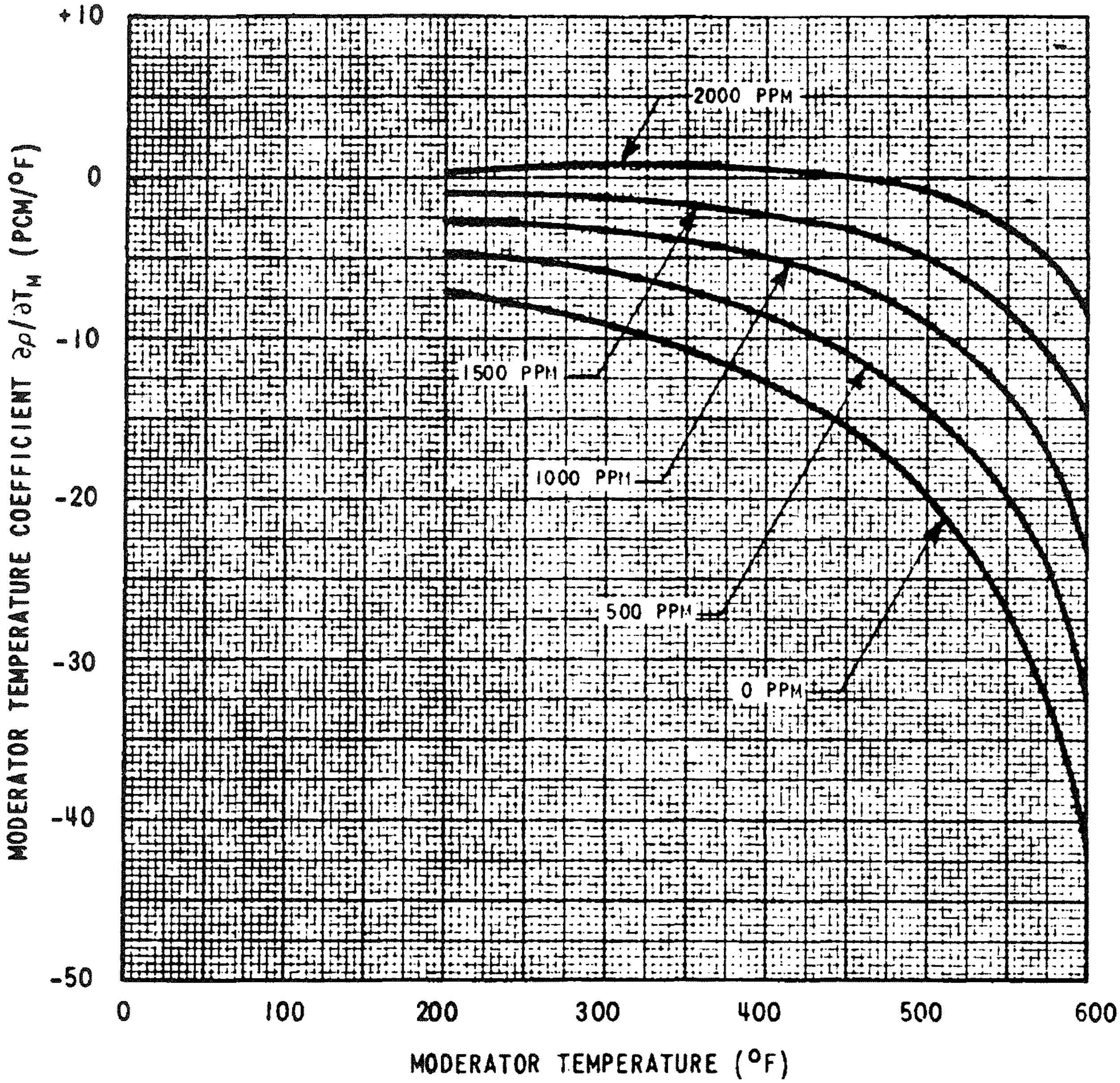


INDIAN POINT UNIT No. 2

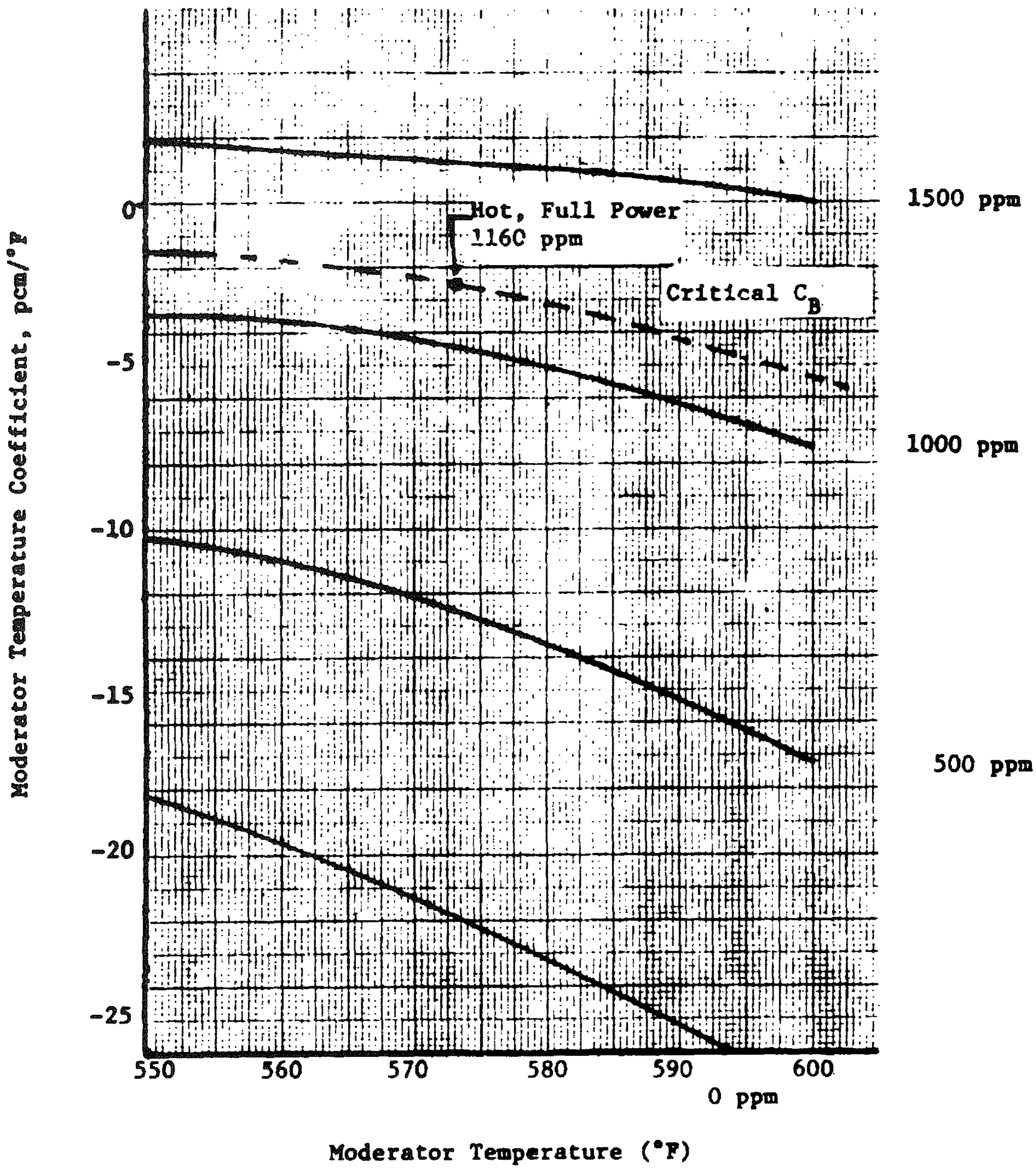
UFSAR FIGURE 3.2-9

BURNABLE POISON ROD LOCATIONS
CYCLE 1

MIC. No. 1999MC3592	REV. No. 17A

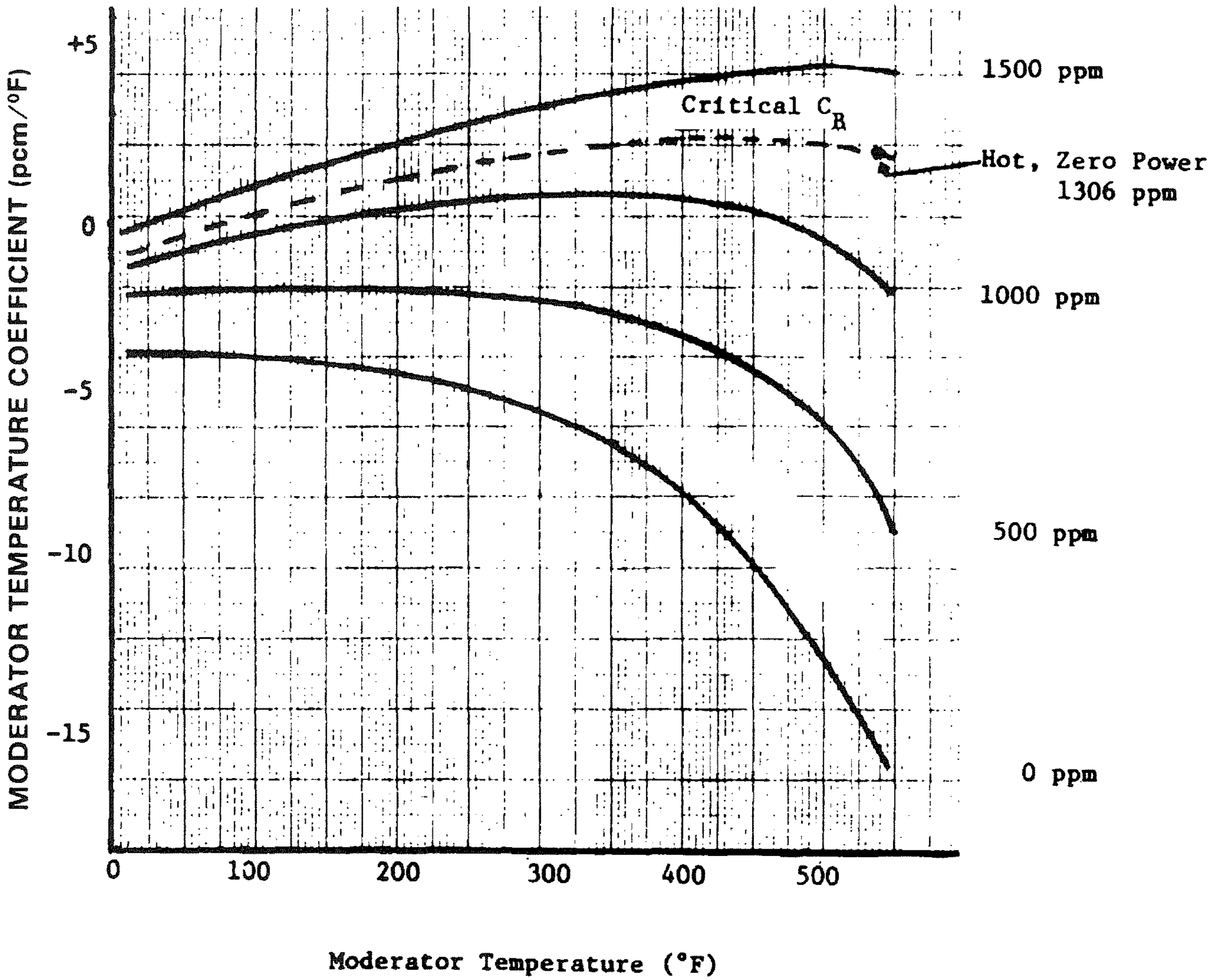


INDIAN POINT UNIT No. 2	
UFSAR FIGURE 3.2–10 MODERATOR TEMPERATURE COEFFICIENT vs MODERATOR TEMPERATURE – EOL, CYCLE 1	
MIC. No. 1999MC3593	REV. No. 17A



INDIAN POINT UNIT No. 2
UFSAR FIGURE 3.2-11
MODERATOR TEMPERATURE COEFFICIENT vs MODERATOR TEMPERATURE – BOL, CYCLE 1 FULL POWER

MIC. No. 1999MC3594	REV. No. 17A



INDIAN POINT UNIT No. 2

UFSAR FIGURE 3.2-12

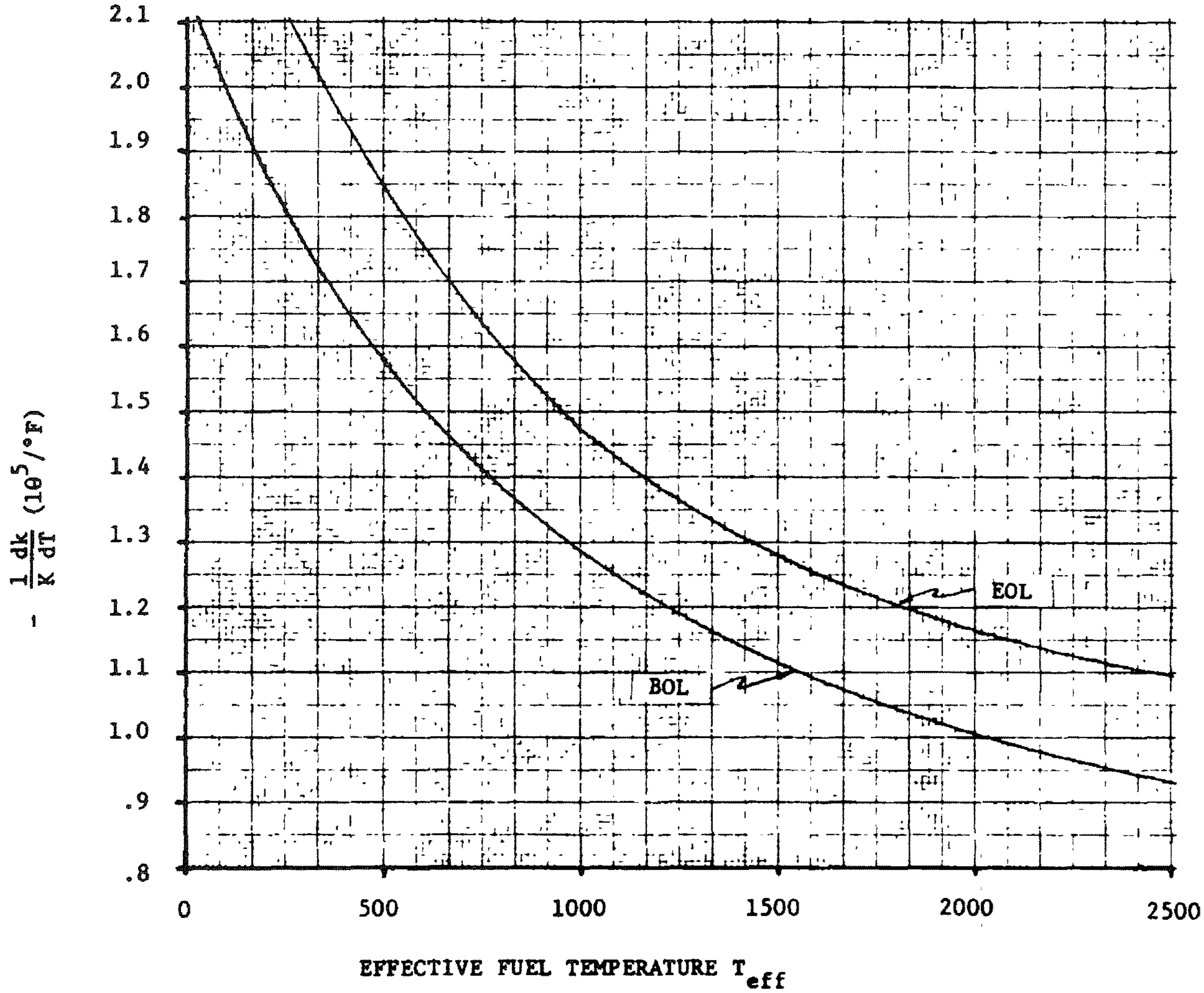
MODERATOR TEMPERATURE COEFFICIENT vs MODERATOR TEMPERATURE – BOL, CYCLE 1 ZERO POWER

MIC. No. 1999MC3595	REV. No. 17A

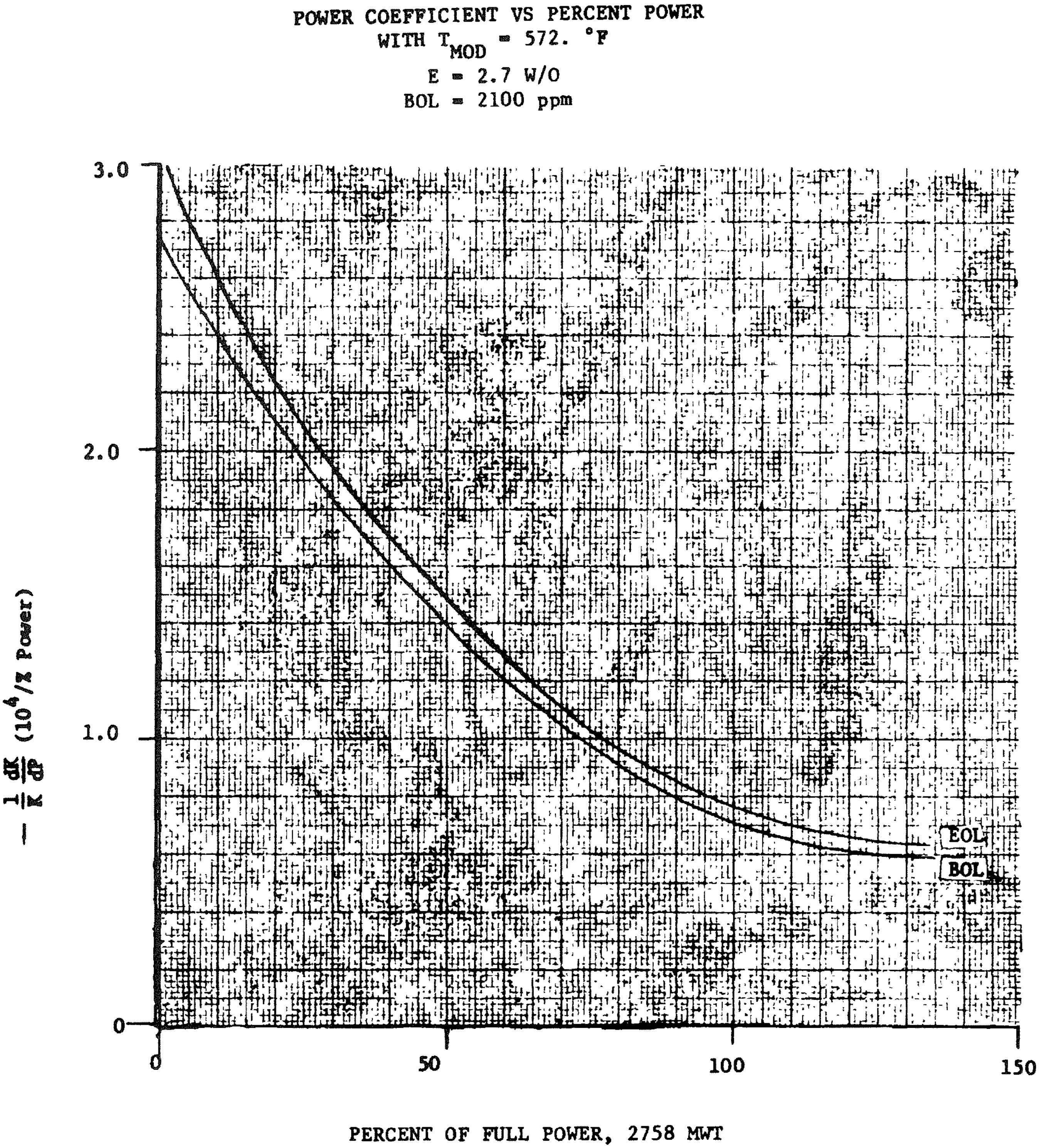
DOPPLER COEFFICIENT

vs

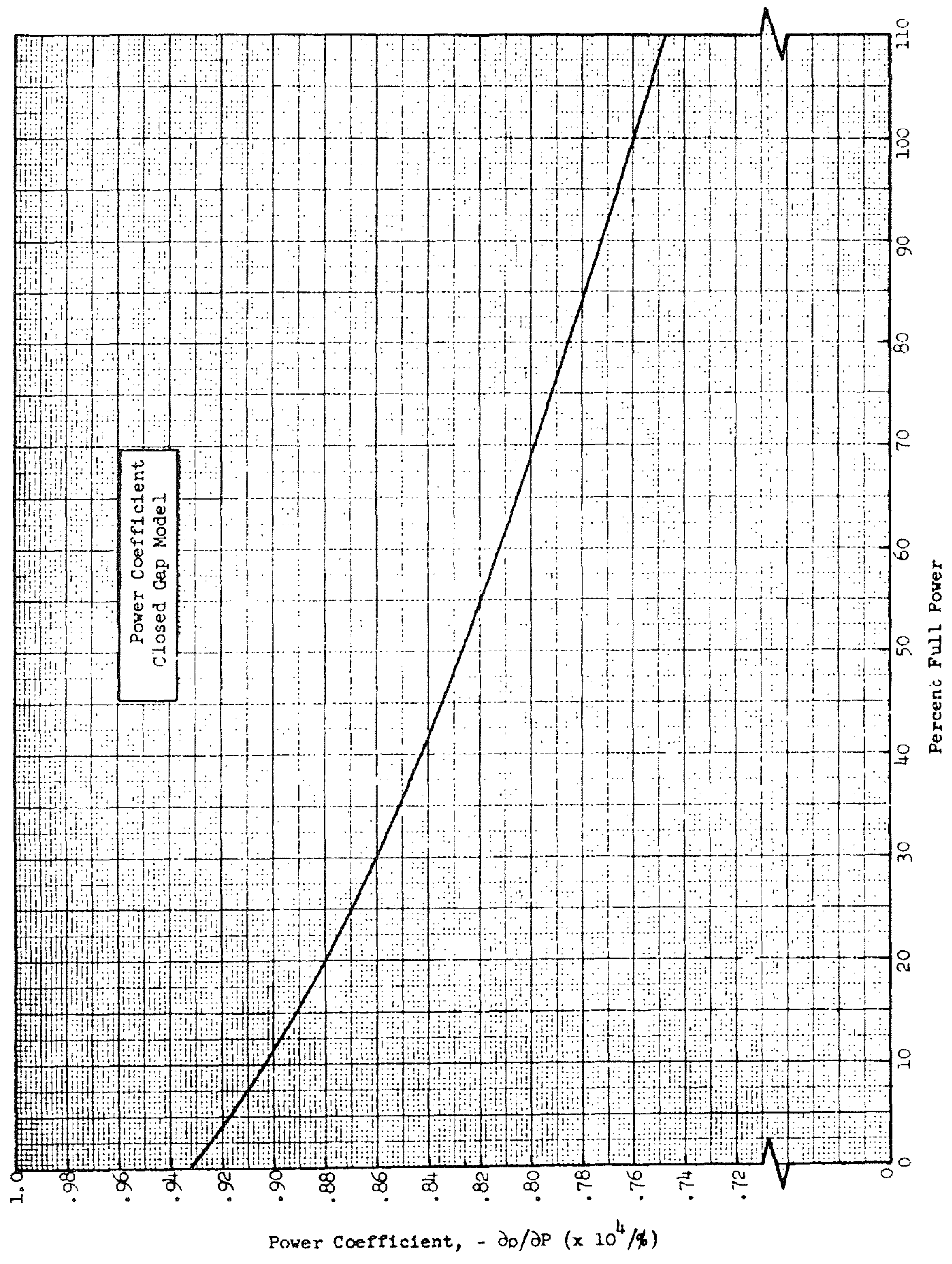
EFFECTIVE FUEL TEMPERATURE



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 3.2–13 DOPPLER COEFFICIENT vs EFFECTIVE FUEL TEMPERATURE – CYCLE 1	
MIC. No. 1999MC3596	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 3.2–14 POWER COEFFICIENT vs PERCENT POWER CYCLE 1	
MIC. No. 1999MC3597	REV. No. 17A

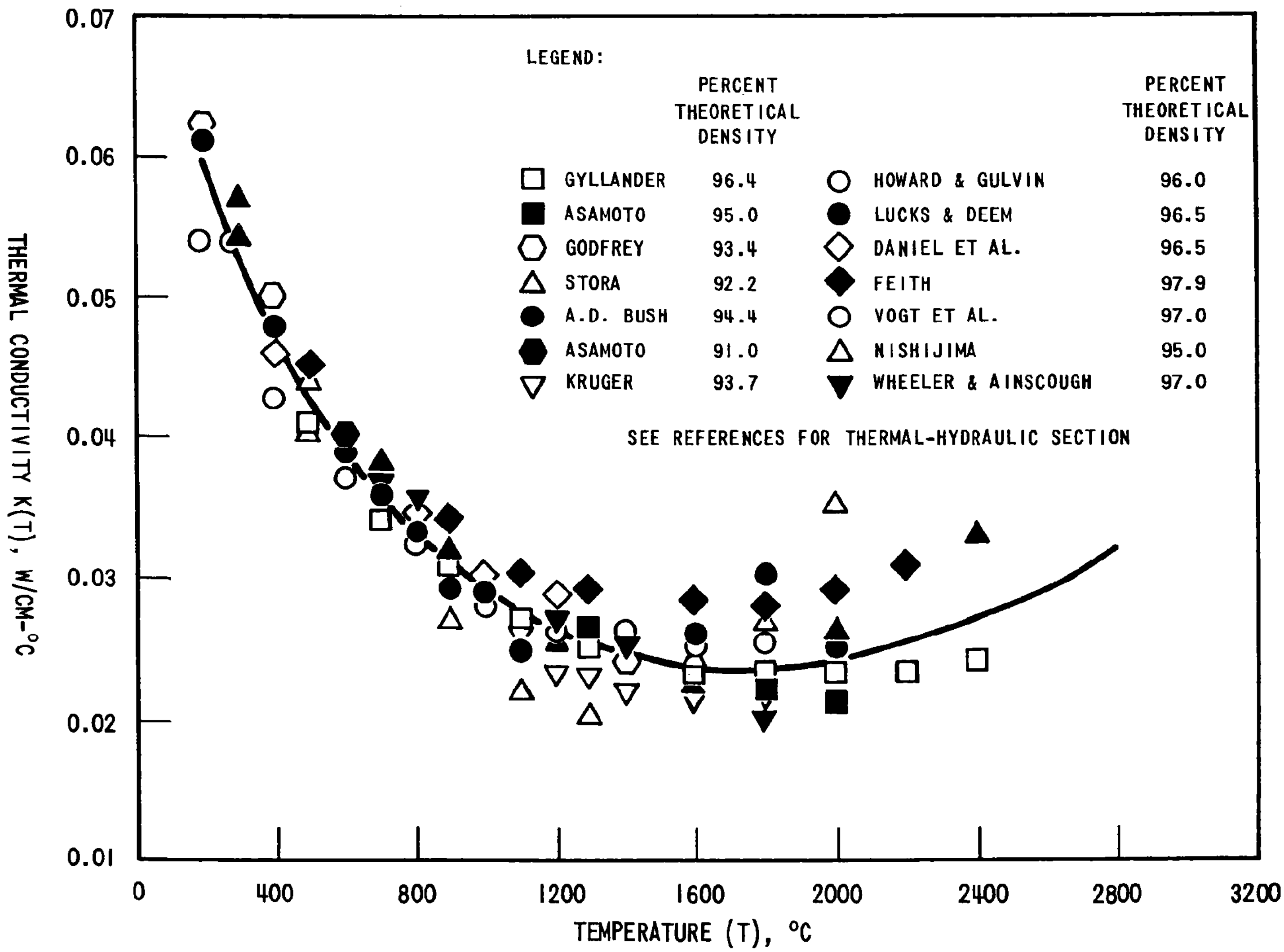


INDIAN POINT UNIT No. 2

UFSAR FIGURE 3.2–15

POWER COEFFICIENT – CLOSED GAP MODEL

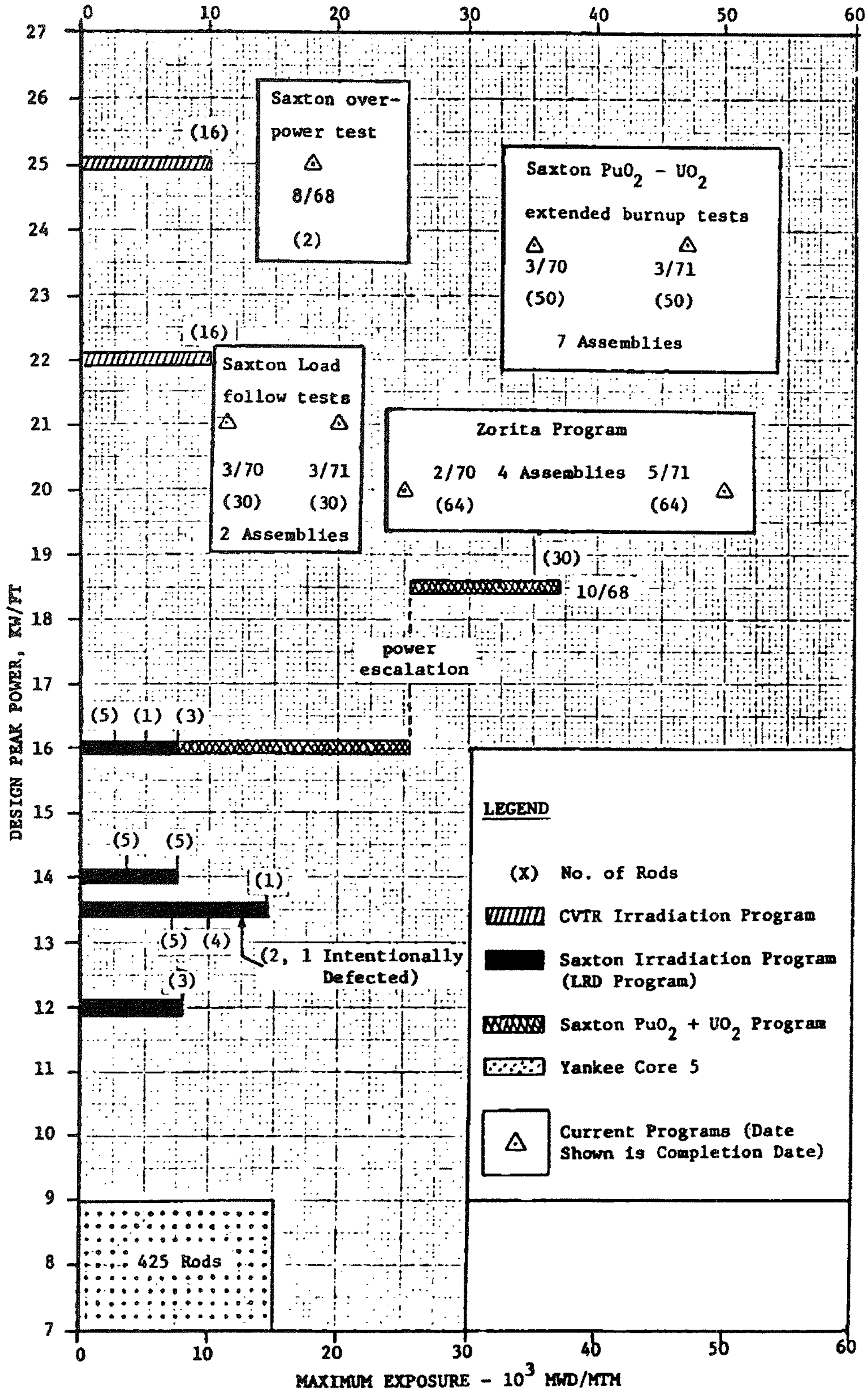
MIC. No. 1999MC3598	REV. No. 17A



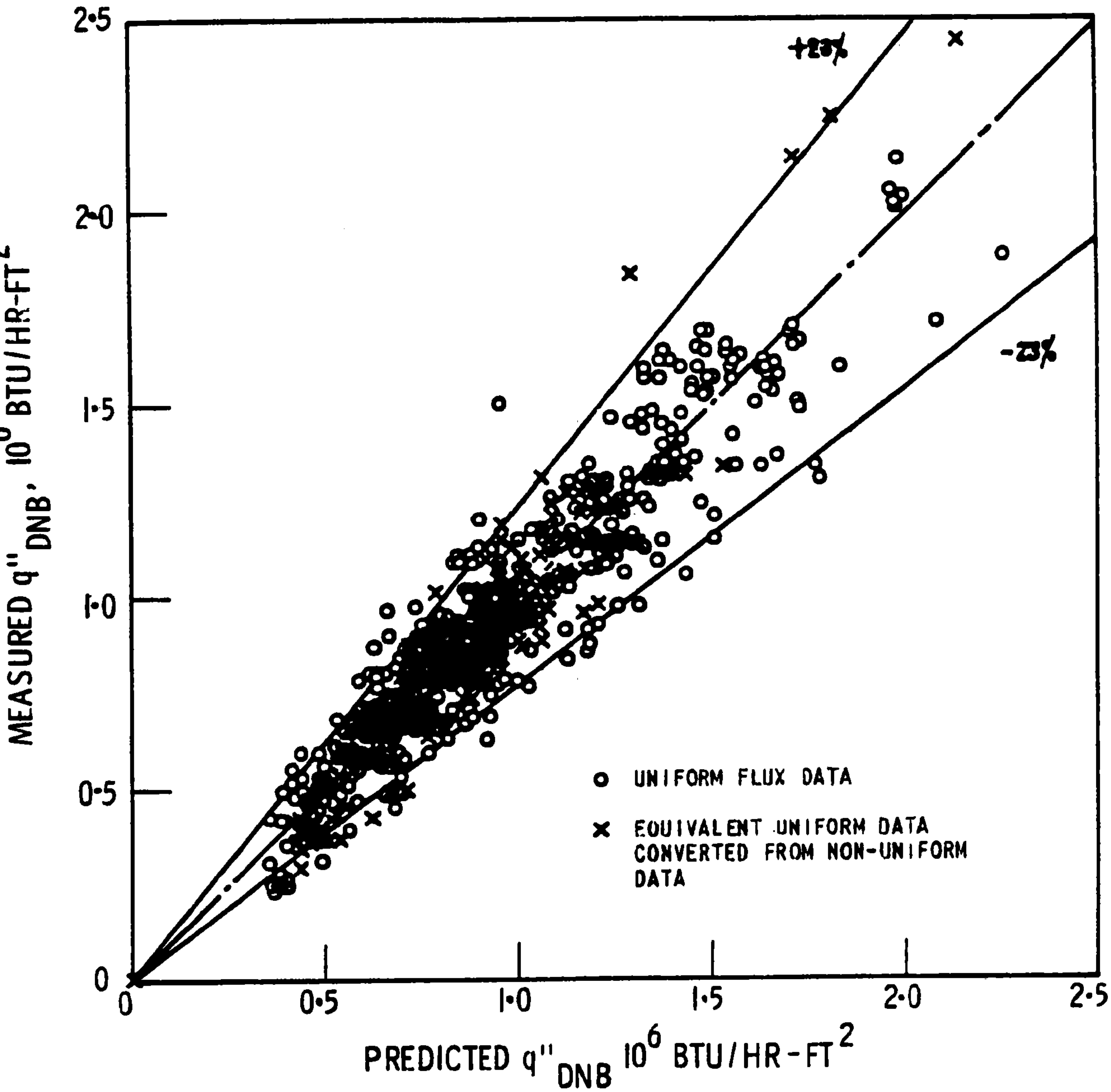
INDIAN POINT UNIT No. 2

UFSAR FIGURE 3.2-38
TYPICAL
THERMAL CONDUCTIVITY
OF UO_2

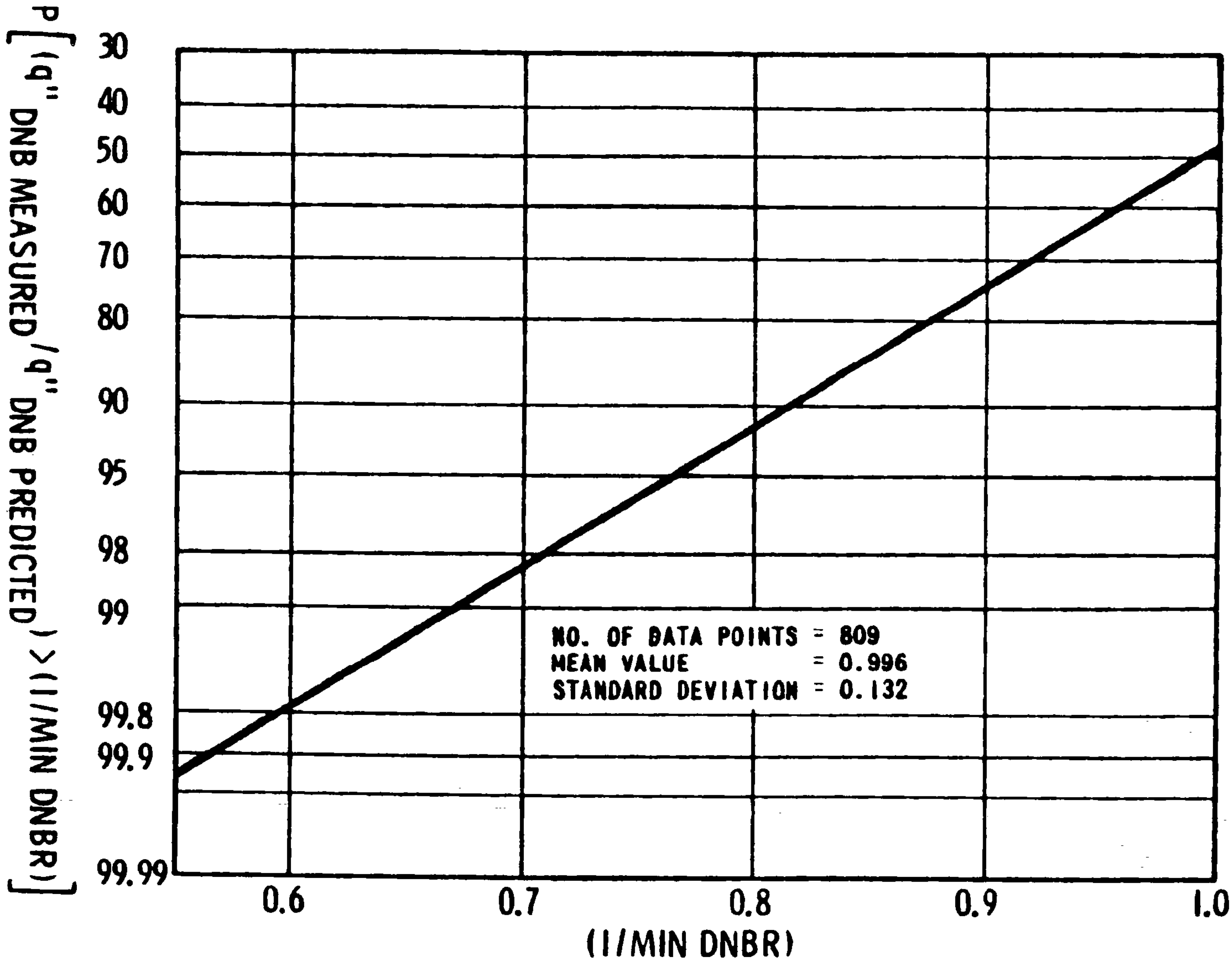
MIC. No. 1999MC3643	REV. No. 17B



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 3.2-39	
HIGH POWER FUEL ROD EXPERIMENTAL PROGRAM	
MIC. No. 1999MC3644	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 3.2-40 TYPICAL COMPARISON OF W-3 PREDICTION AND UNIFORM FLUX DATA	
MIC. No. 1999MC3645	REV. No. 17B

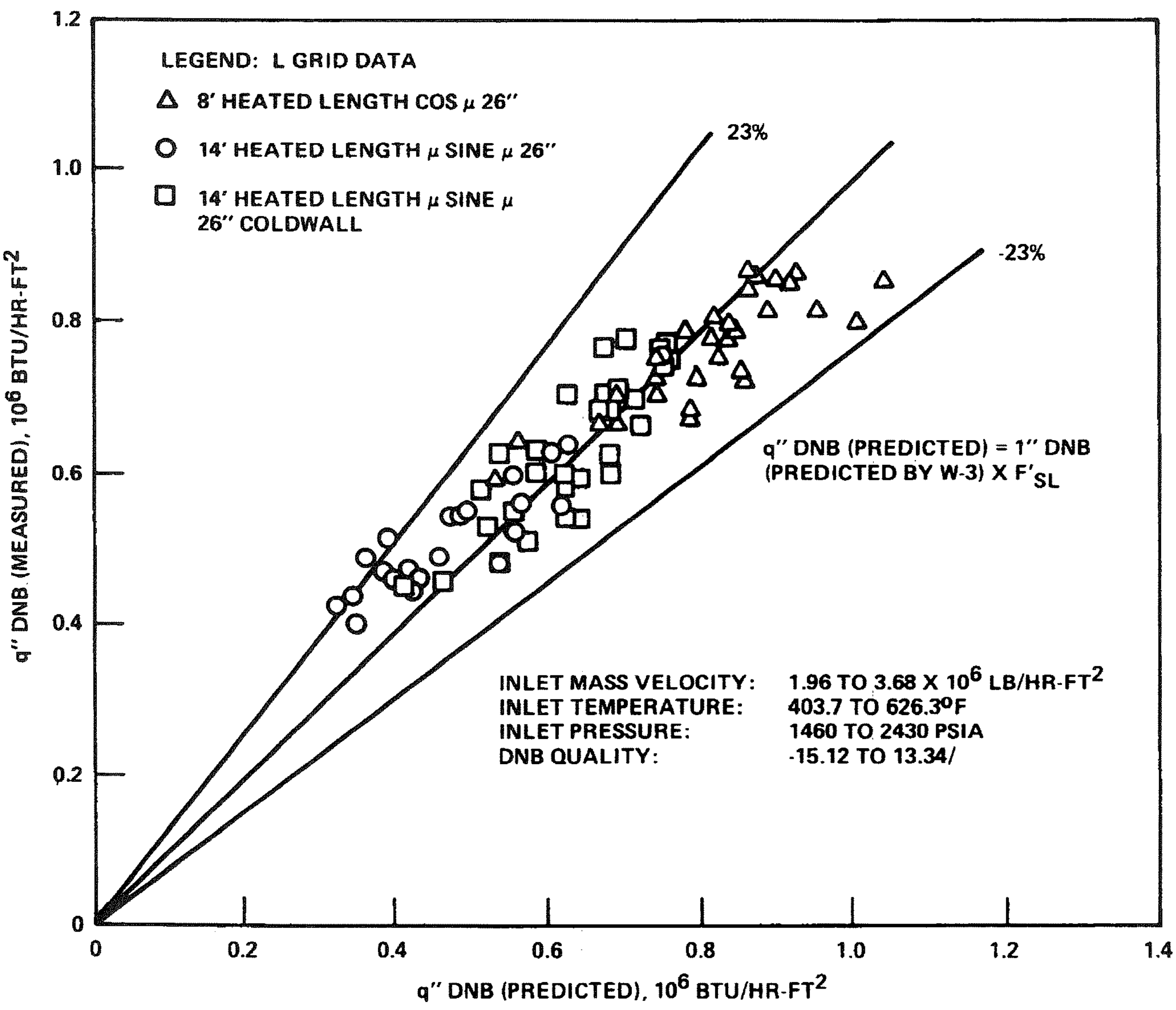


W-3 CORRELATION PROBABILITY DISTRIBUTION CURVE

INDIAN POINT UNIT No. 2

UFSAR FIGURE 3.2-41
TYPICAL
W-3 CORRELATION PROBABILITY DISTRIBUTION CURVE

MIC. No. 1999MC3678 | REV. No. 17B

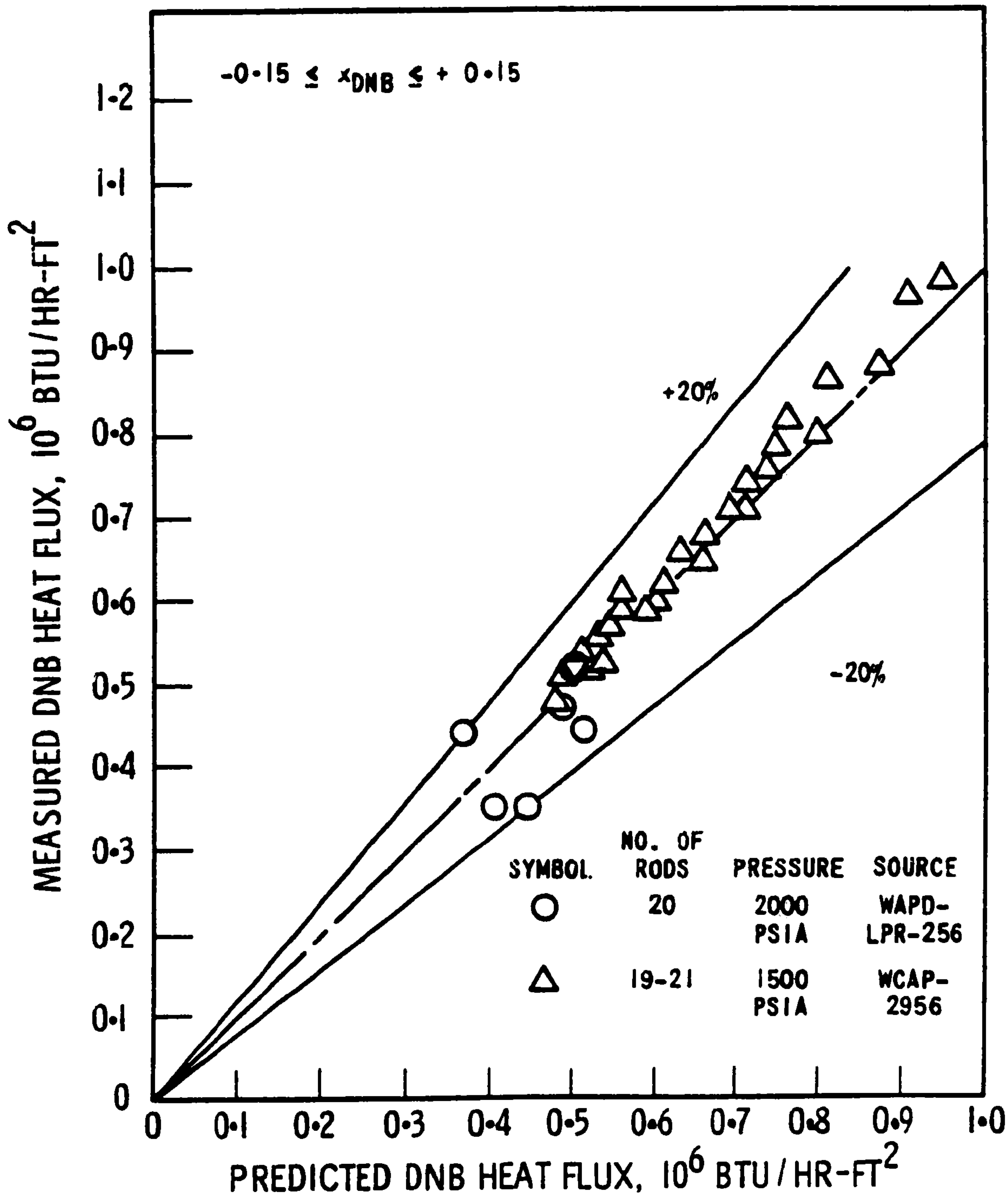


INDIAN POINT UNIT No. 2

UFSAR FIGURE 3.2-42

COMPARISON OF "L" GRID TYPICAL AND THIMBLE COLD WALL CELL ROD BUNDLE DNB DATA FOR NON-UNIFORM AXIAL HEAT FLUX WITH PREDICTIONS OF W-3 x F'_{SL}

MIC. No. 1999MC3679	REV. No. 17A

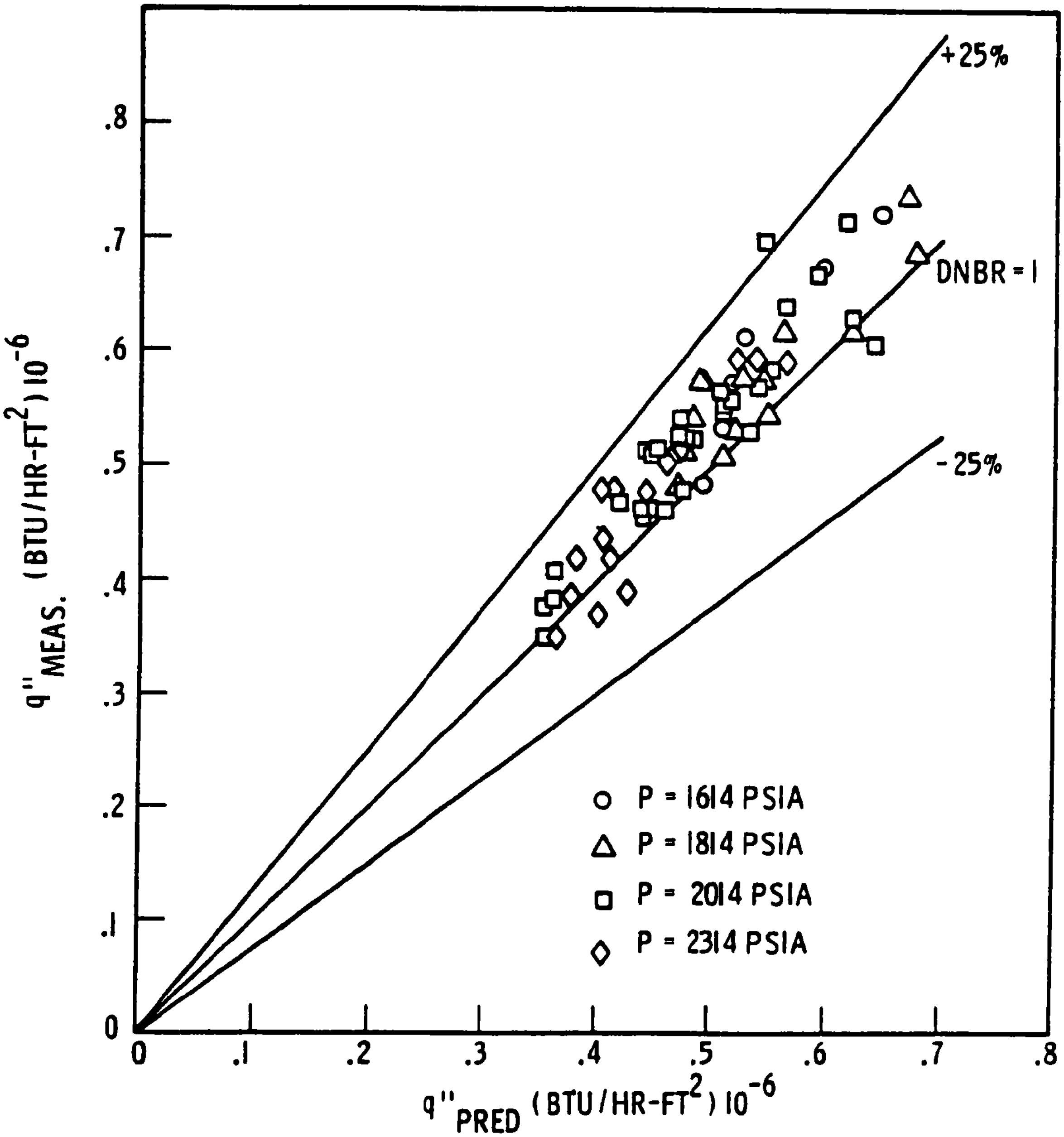


INDIAN POINT UNIT No. 2

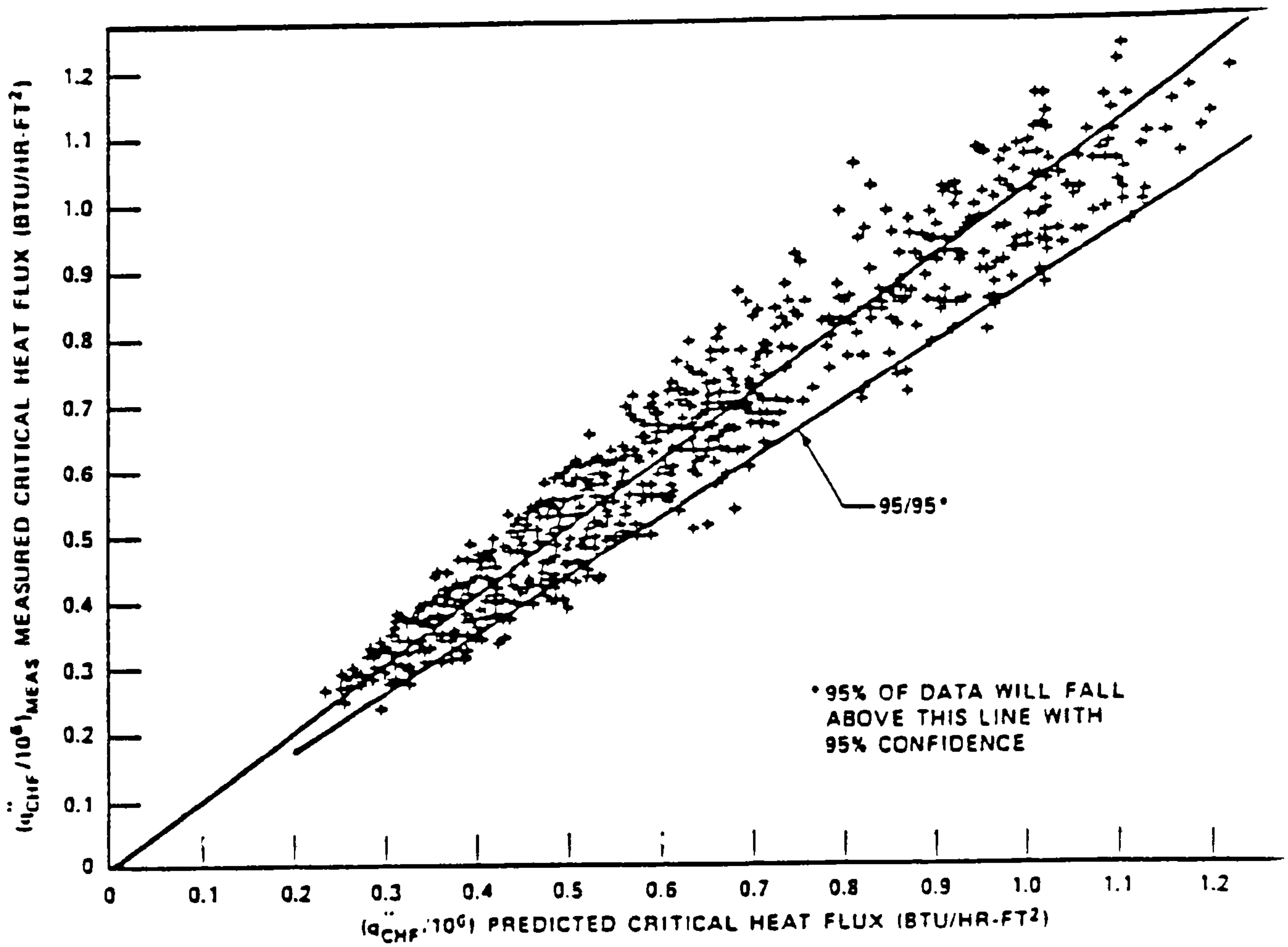
UFSAR FIGURE 3.2-43
TYPICAL
COMPARISON OF W-3 CORRELATION
WITH ROD BUNDLE DNB DATA
(SIMPLE GRID WITHOUT MIXING VANE)

MIC. No. 1999MC3680	REV. No. 17B

2/23/2004 11:52:07 AM vnd1999MC3681-0-17B.DWG



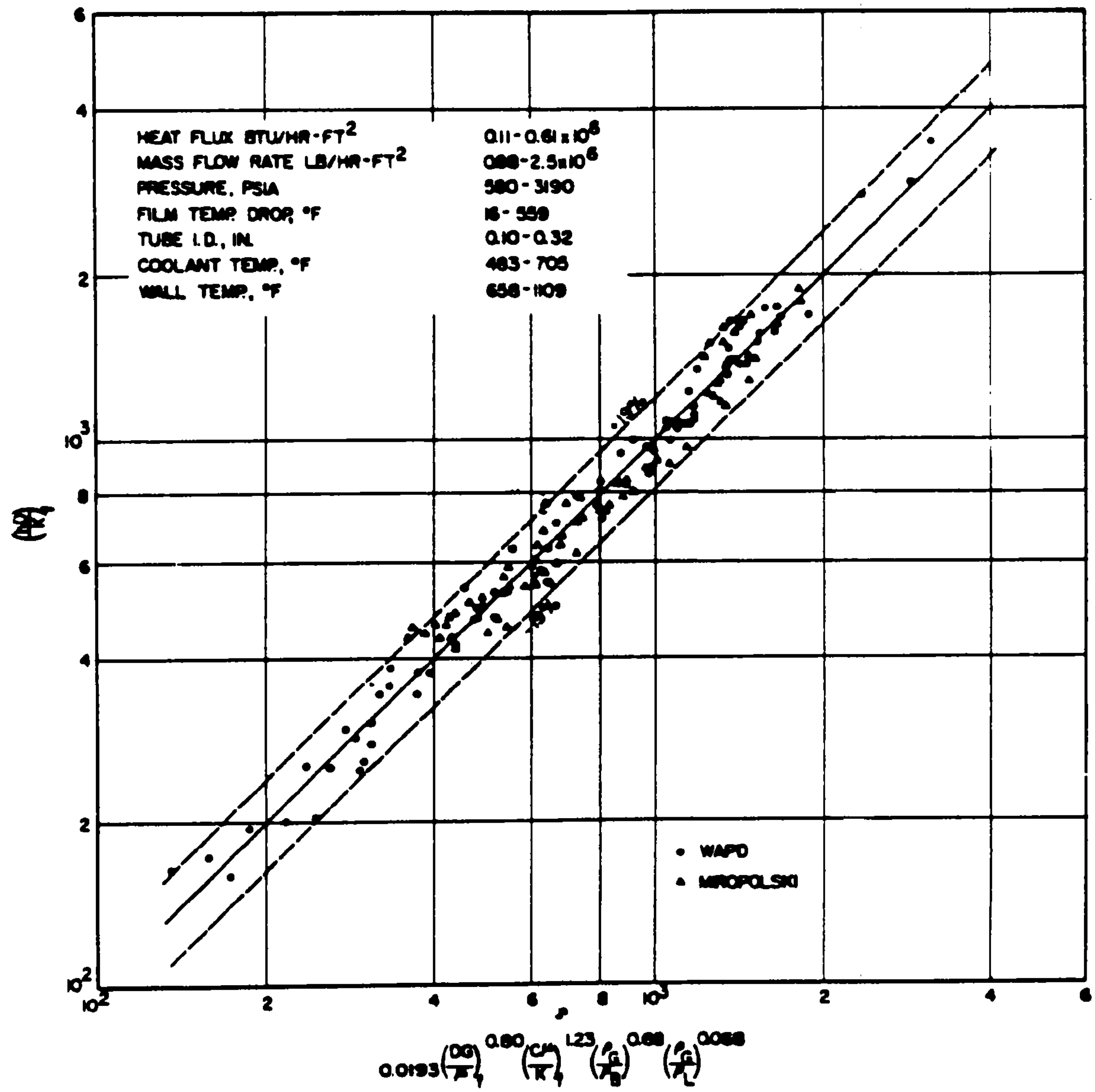
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 3.2-44 TYPICAL COMPARISON OF W-3 CORRELATION WITH ROD BUNDLE DNB DATA (SIMPLE GRID WITH MIXING VANE)	
MIC. No. 1999MC3681	REV. No. 17B



INDIAN POINT UNIT No. 2

UFSAR FIGURE 3.2-44A
TYPICAL
MEASURED VERSUS PREDICTED
CRITICAL HEAT FLUX-WRB-1 CORRELATION

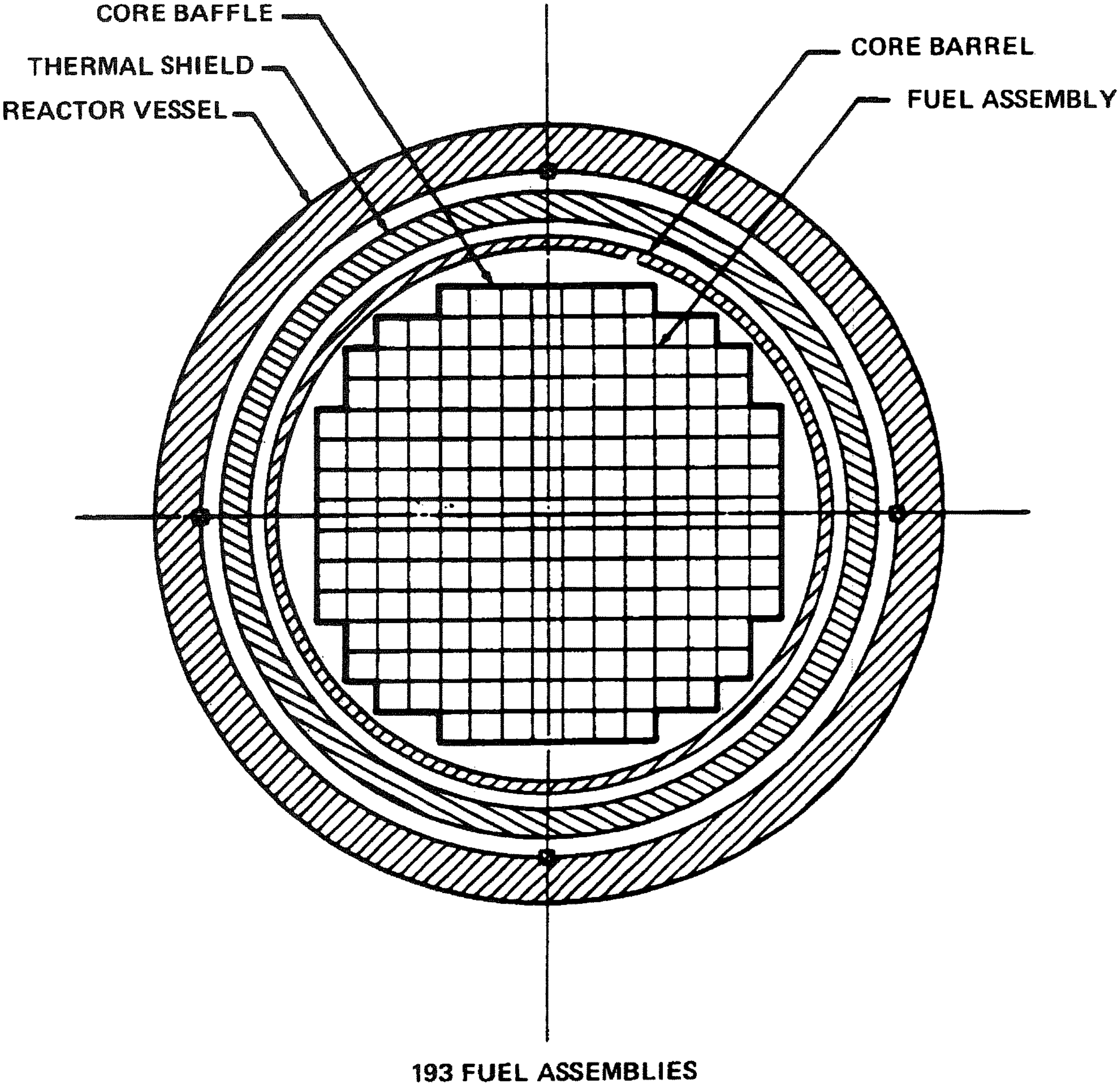
MIC. No. 1999MC3682	REV. No. 17B



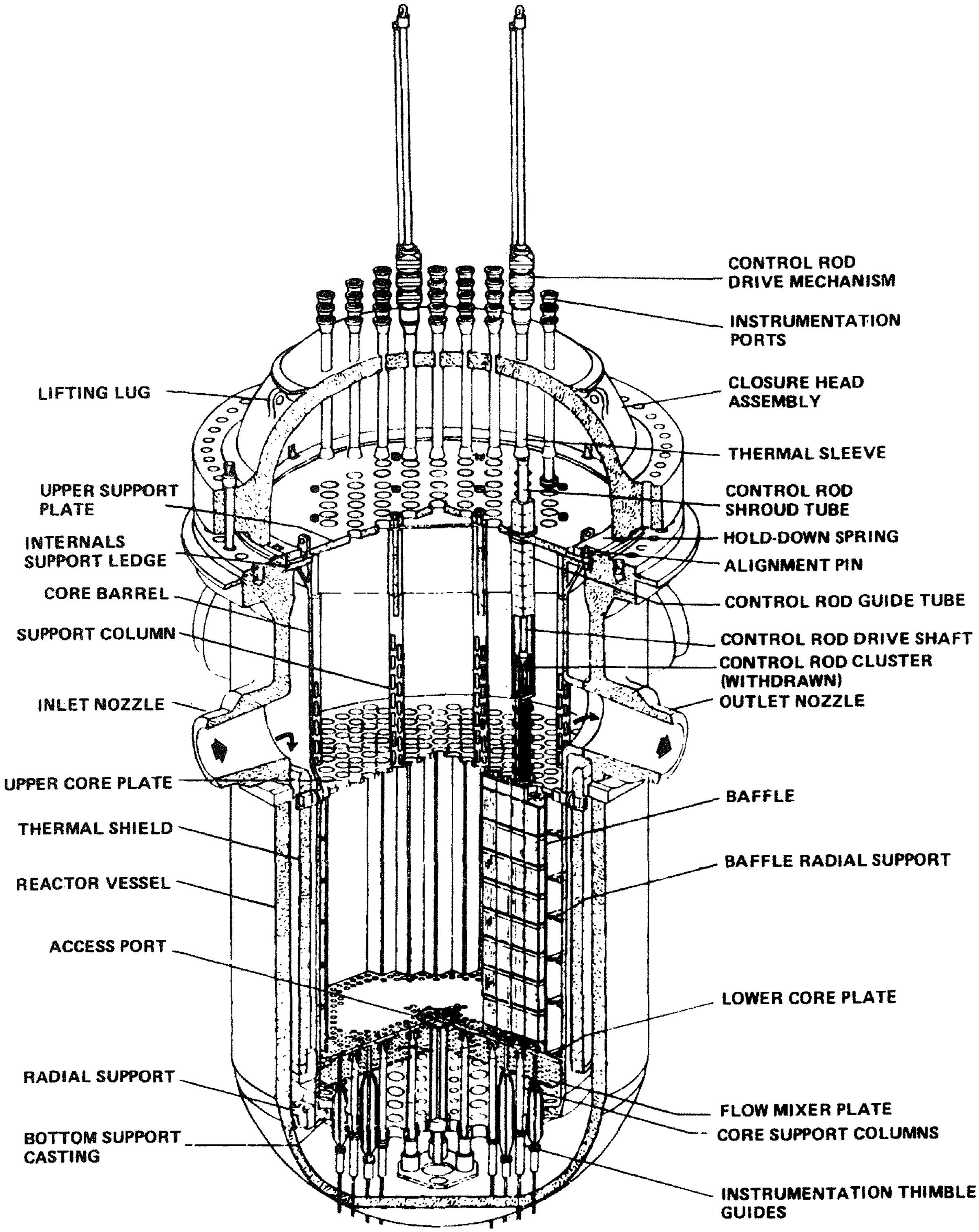
INDIAN POINT UNIT No. 2

UFSAR FIGURE 3.2-45
TYPICAL
STABLE FILM BOILING HEAT TRANSFER
DATA AND CORRELATION

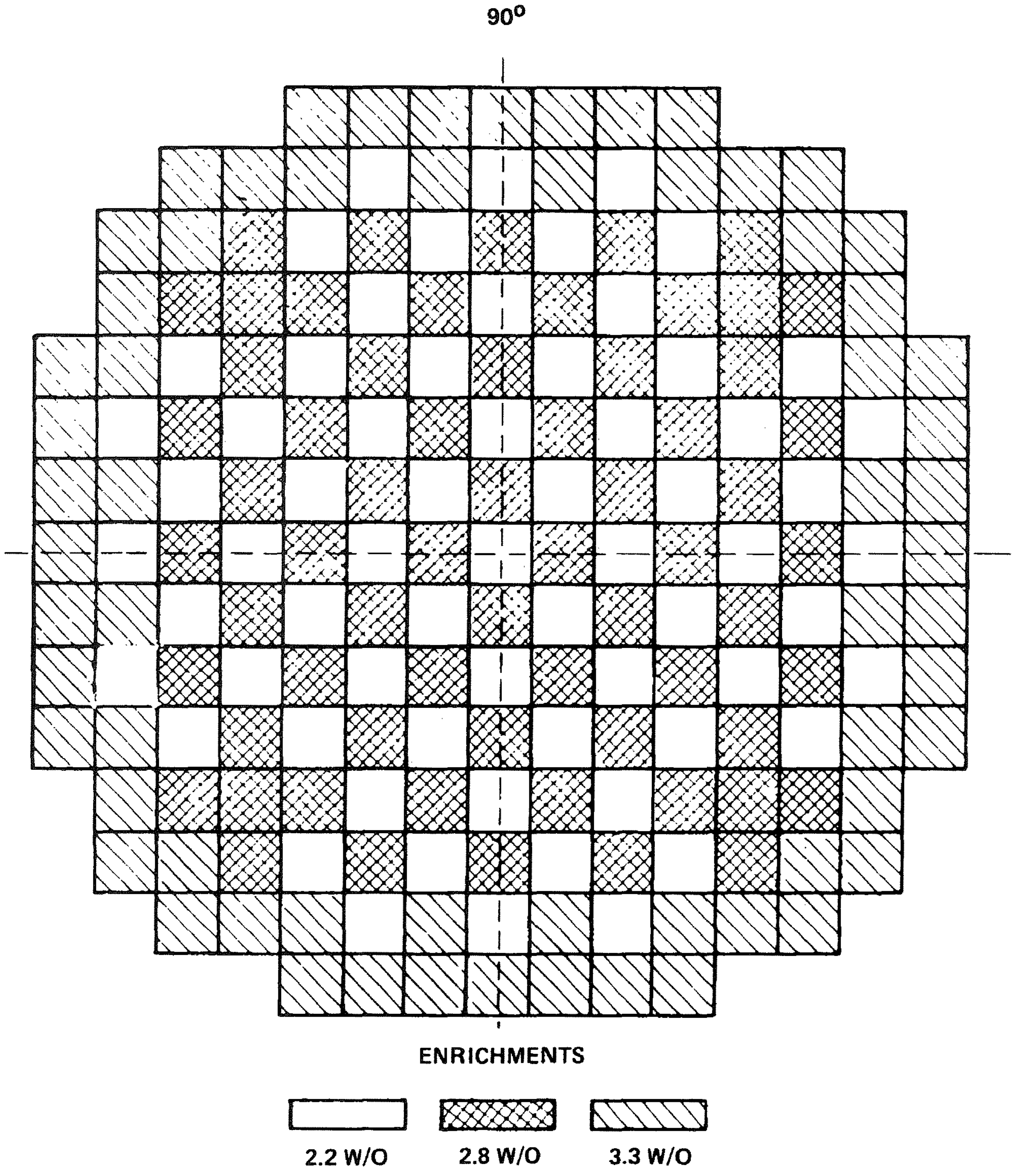
MIC. No. 1999MC3683 | REV. No. 17B



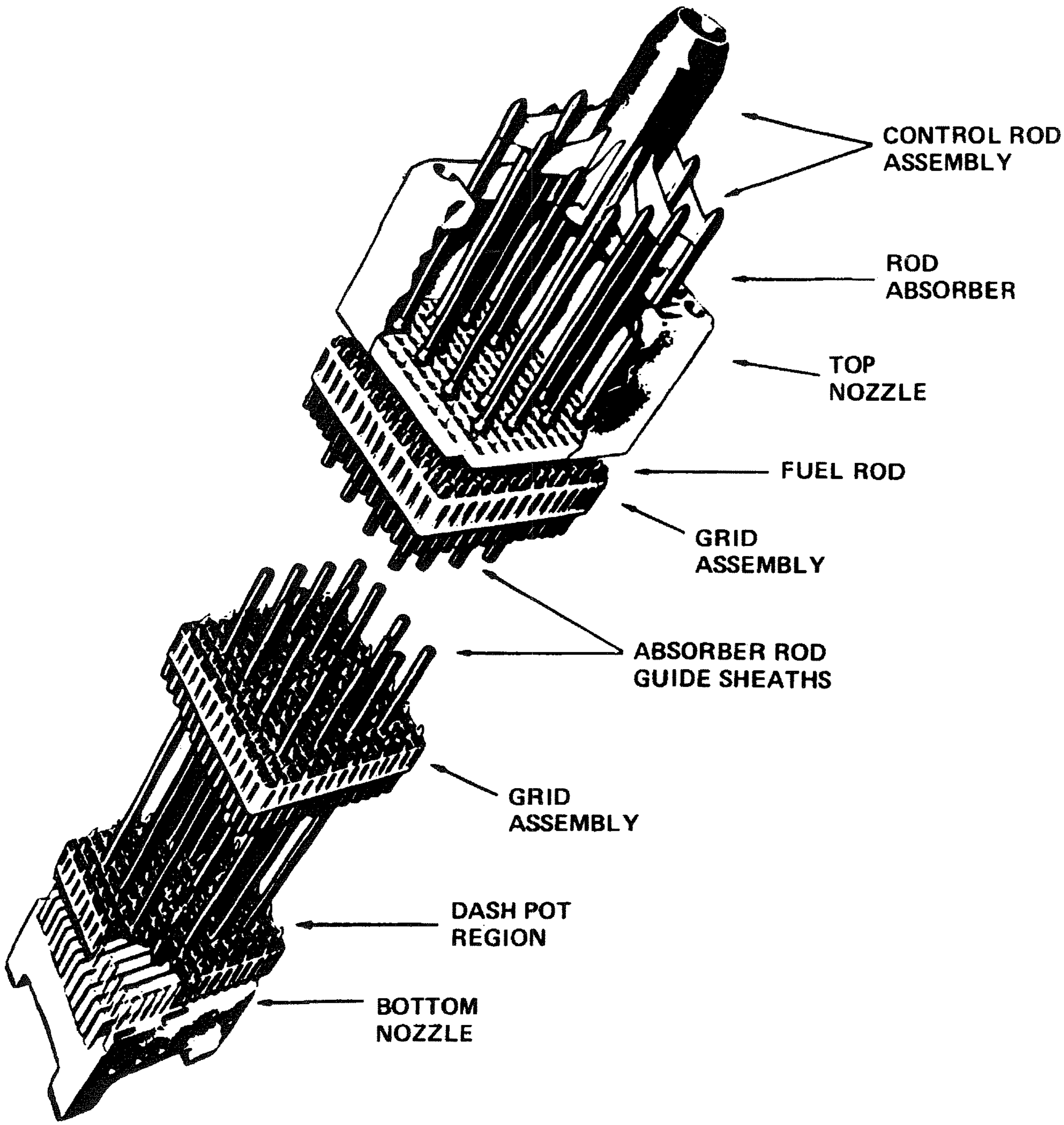
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 3.2-46 CORE CROSS SECTION	
MIC. No. 1999MC3684	REV. No. 17A



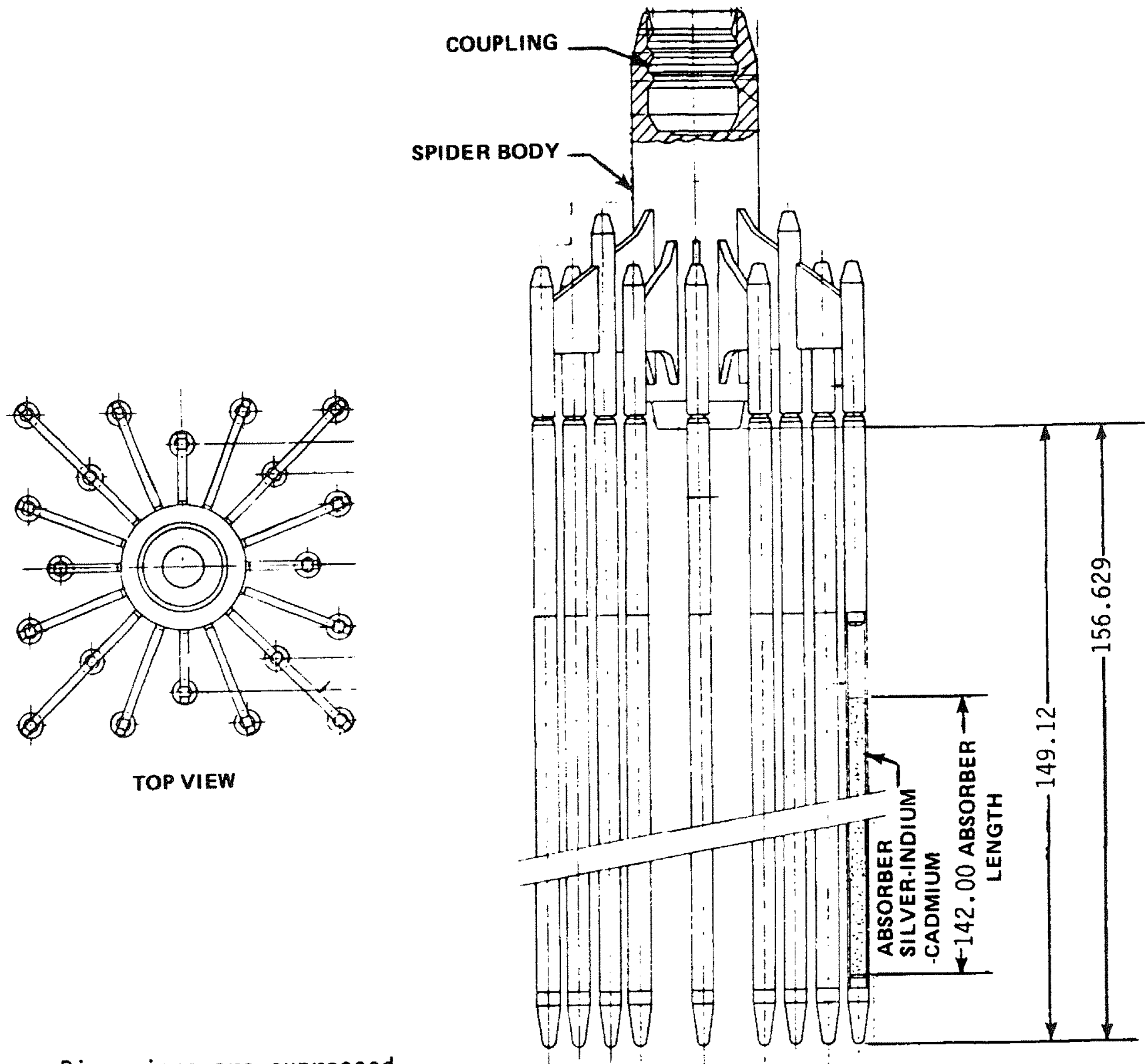
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 3.2-47	
REACTOR VESSEL INTERNALS	
MIC. No. 1999MC3685	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 3.2-48	
CORE LOADING ARRANGEMENT - CYCLE 1	
MIC. No. 1999MC3686	REV. No. 17A

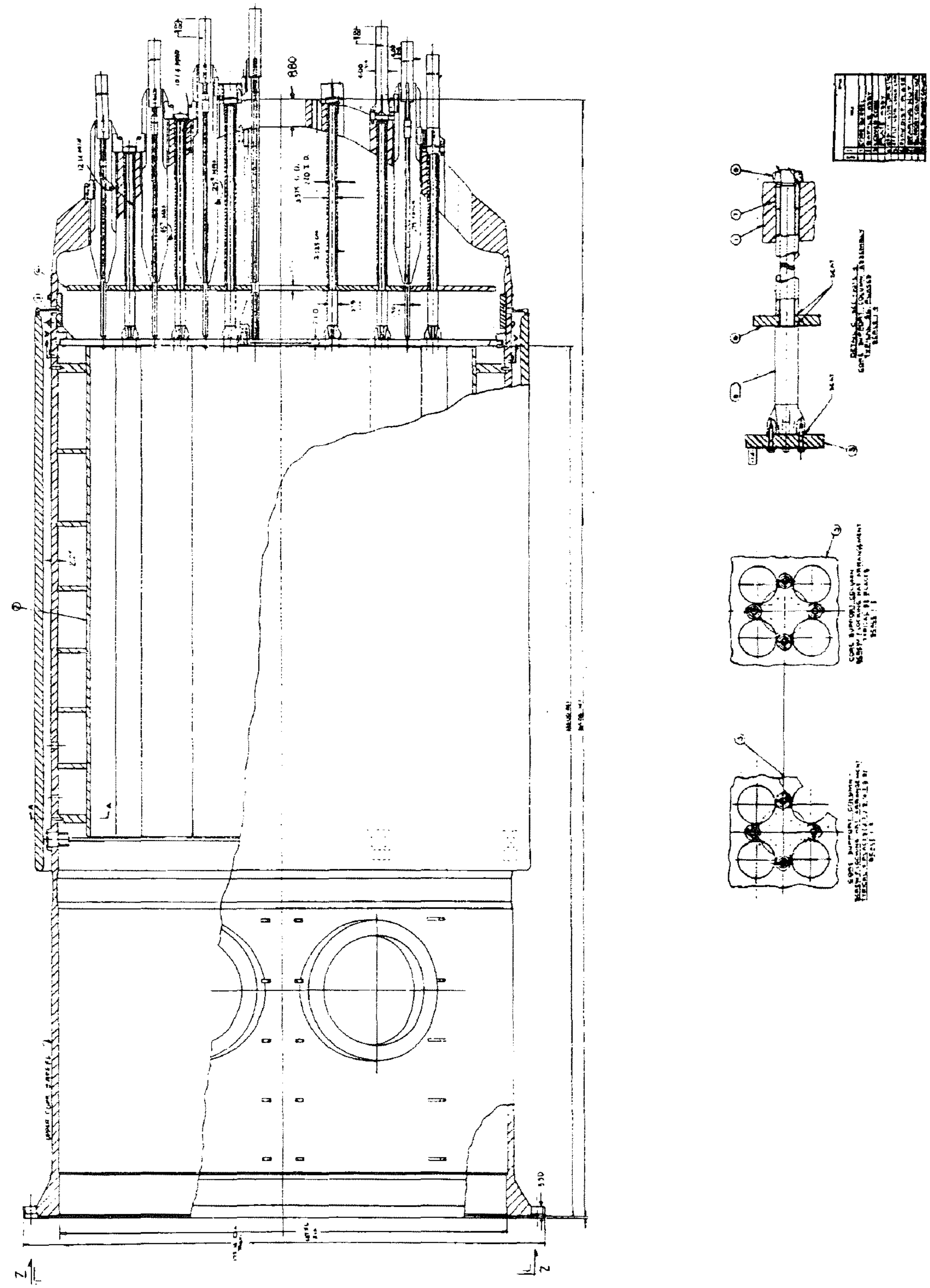


INDIAN POINT UNIT No. 2	
UFSAR FIGURE 3.2-49 TYPICAL ROD CLUSTER CONTROL ASSEMBLY	
MIC. No. 1999MC3687	REV. No. 17A



Note: Dimensions are expressed in inches.

INDIAN POINT UNIT No. 2
UFSAR FIGURE 3.2-50 ROD CONTROL CLUSTER ASSEMBLY OUTLINE
MIC. No. 1999MC3688 | REV. No. 17A

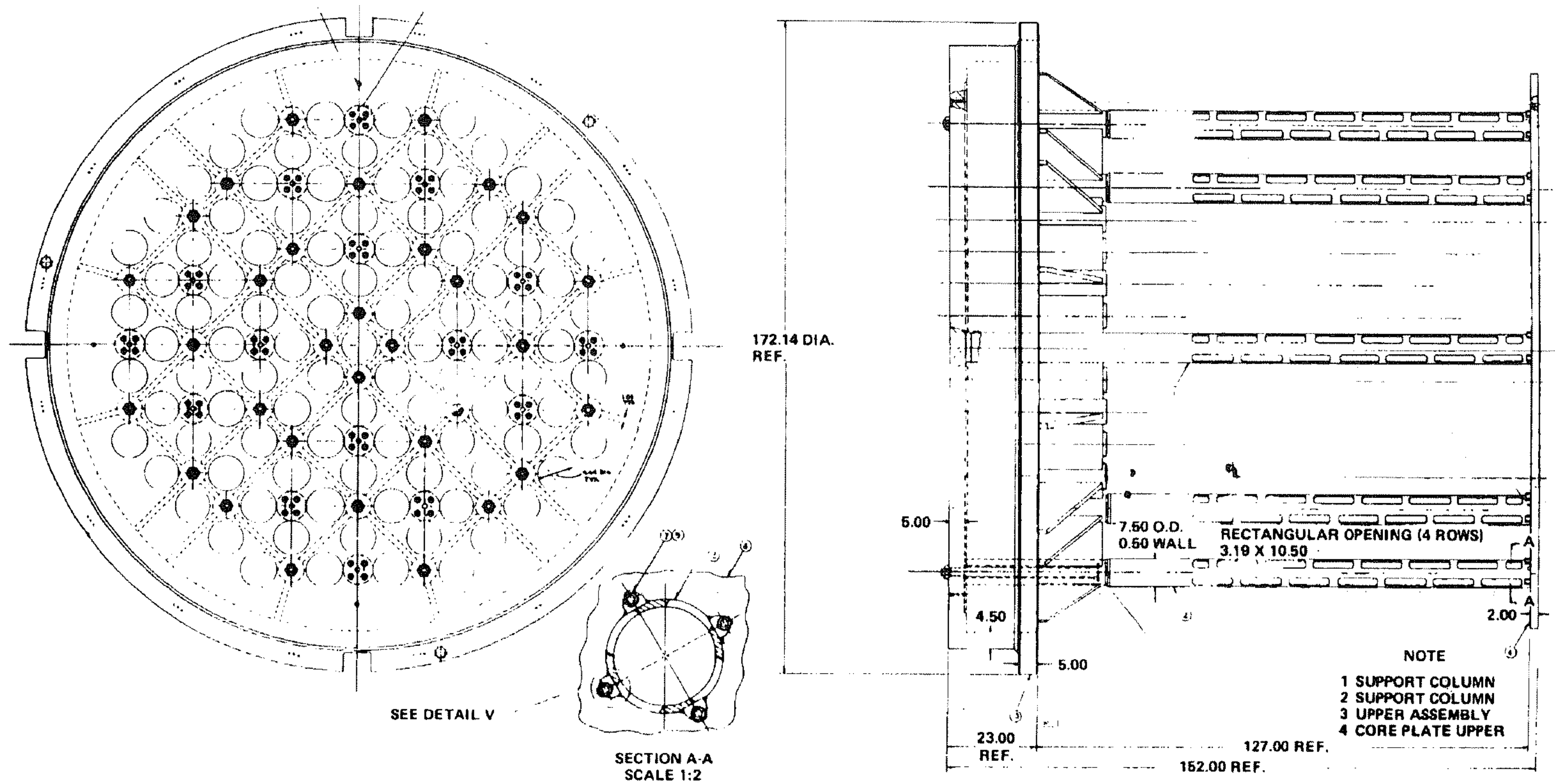


INDIAN POINT UNIT No. 2

UFSAR FIGURE 3.2-51

CORE BARREL ASSEMBLY

MIC. No. 1999MC3705	REV. No. 17A



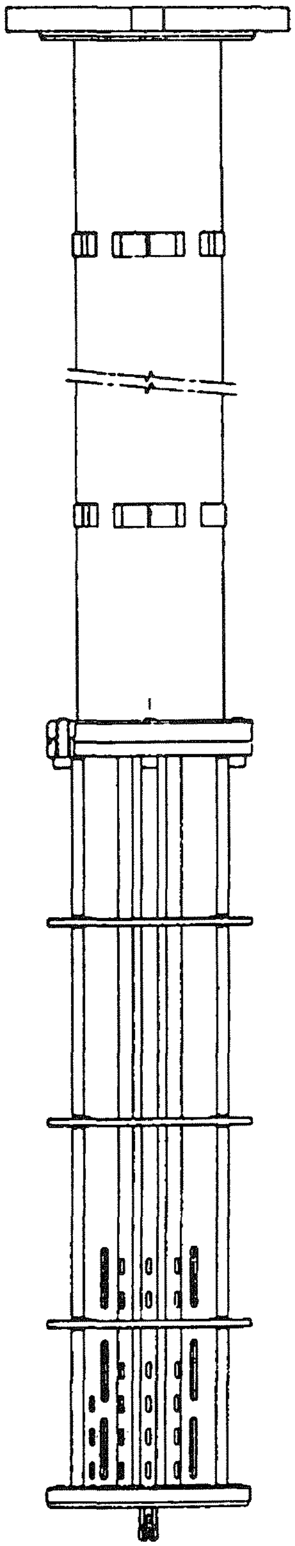
Note: Dimensions are expressed in inches.

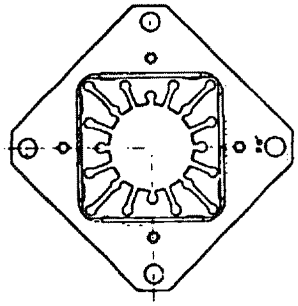
INDIAN POINT UNIT No. 2

UFSAR FIGURE 3.2-52

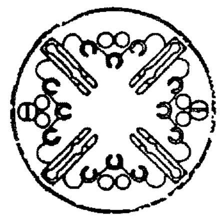
UPPER CORE
SUPPORT STRUCTURE

MIC. No. 1999MC3706 | REV. No. 17A



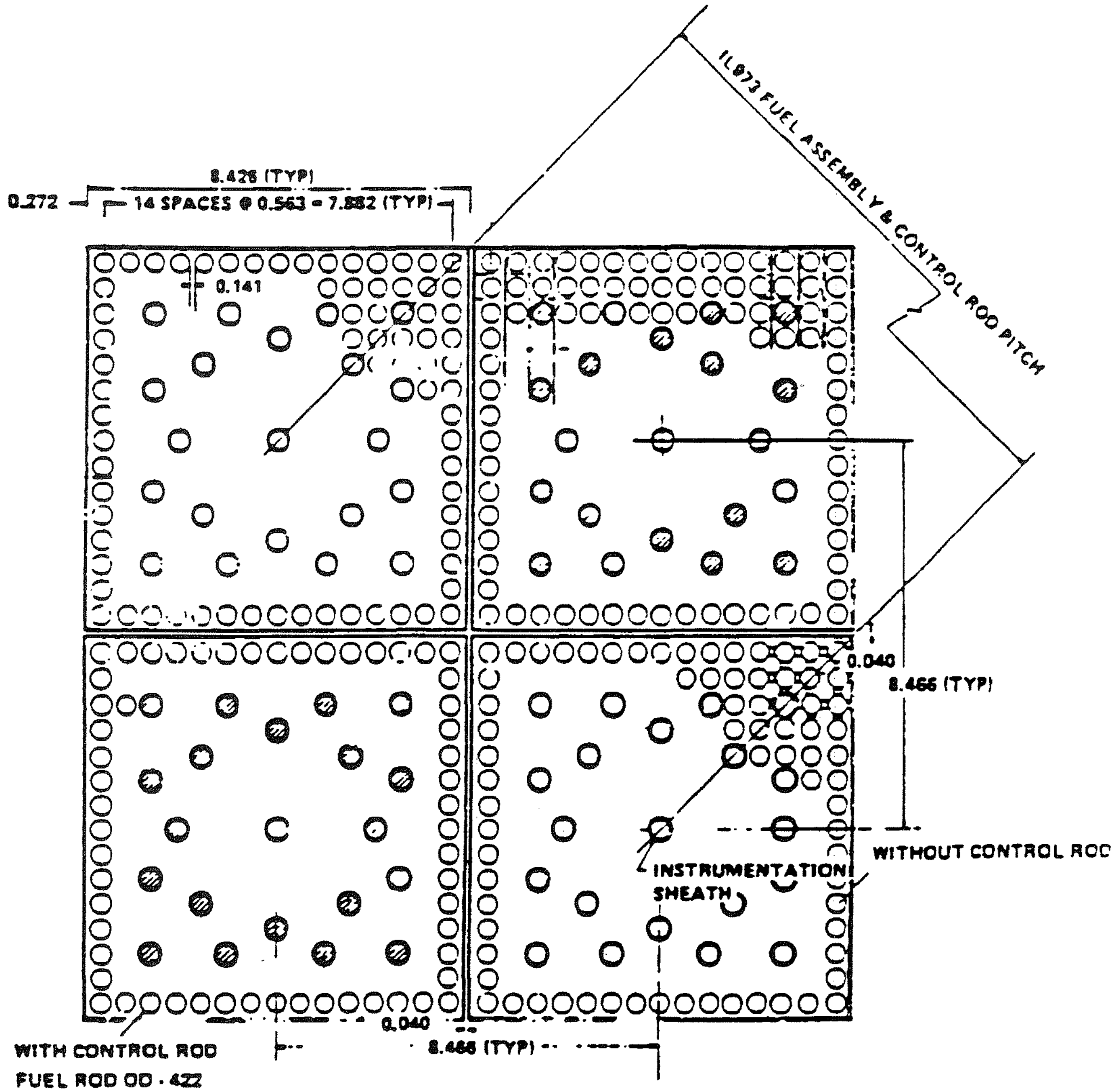


TOP VIEW



BOTTOM VIEW

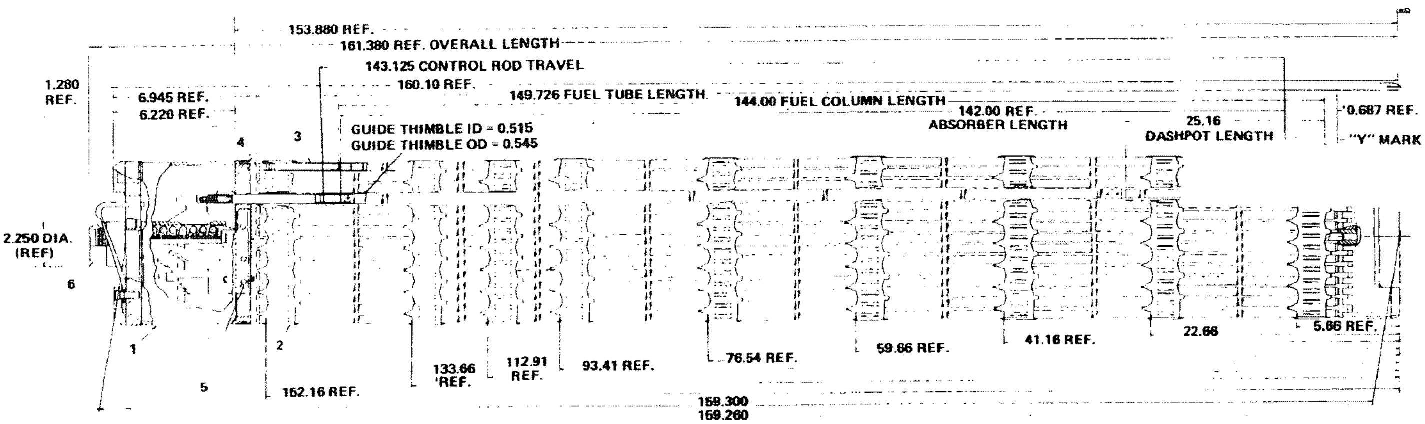
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 3.2-53 GUIDE TUBE ASSEMBLY	
MIC. No. 1999MC3707	REV. No. 17A



FUEL ROD OD - .422
CLAD THICKNESS - .0243
CLAD MATERIAL - ZIRC (ZIRLO™ for VANTAGE+)
FUEL RODS/ASSY - 204

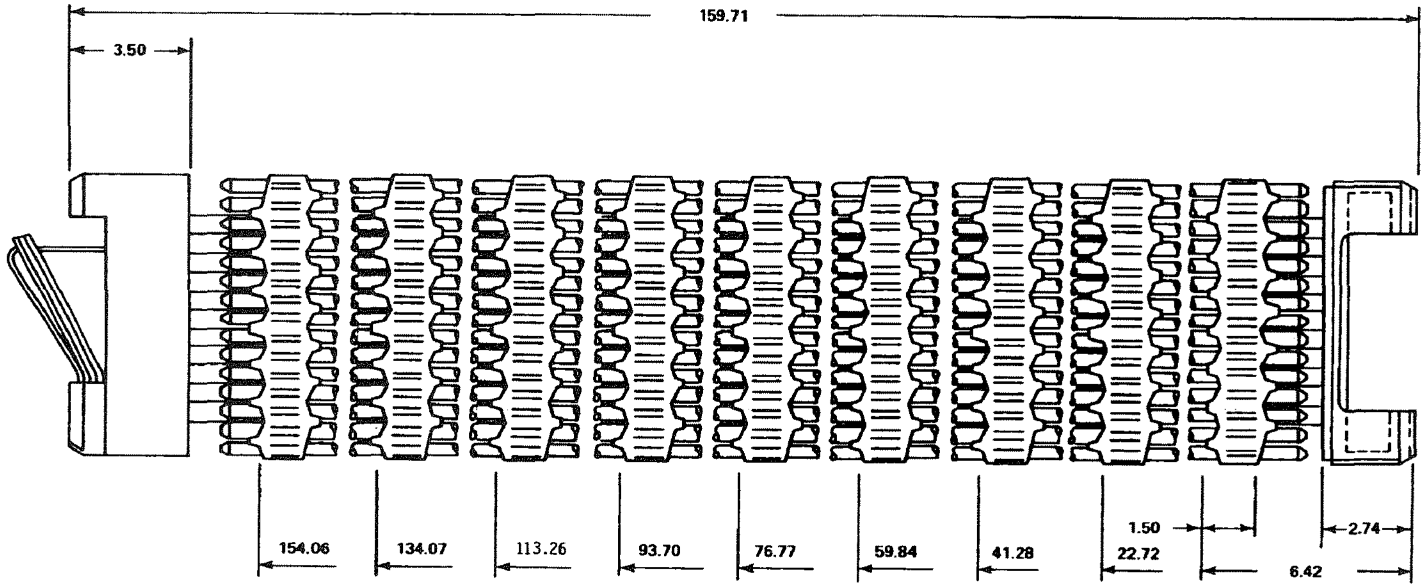
Note: (1) All dim. corrected to 68°F ± 2°
(2) Dimensions are expressed in inches

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 3.2-54 FUEL ASSEMBLY AND CONTROL CLUSTER CROSS SECTION – HIPAR, LOPAR, OFA AND VANTAGE+	
MIC. No. 1999MC3708	REV. No. 17A



Note: Dimensions are expressed in inches.

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 3.2-55	
HIPAR FUEL ASSEMBLY	
MIC. No. 1999MC3709	REV. No. 17A



Note: Dimensions are expressed in inches.

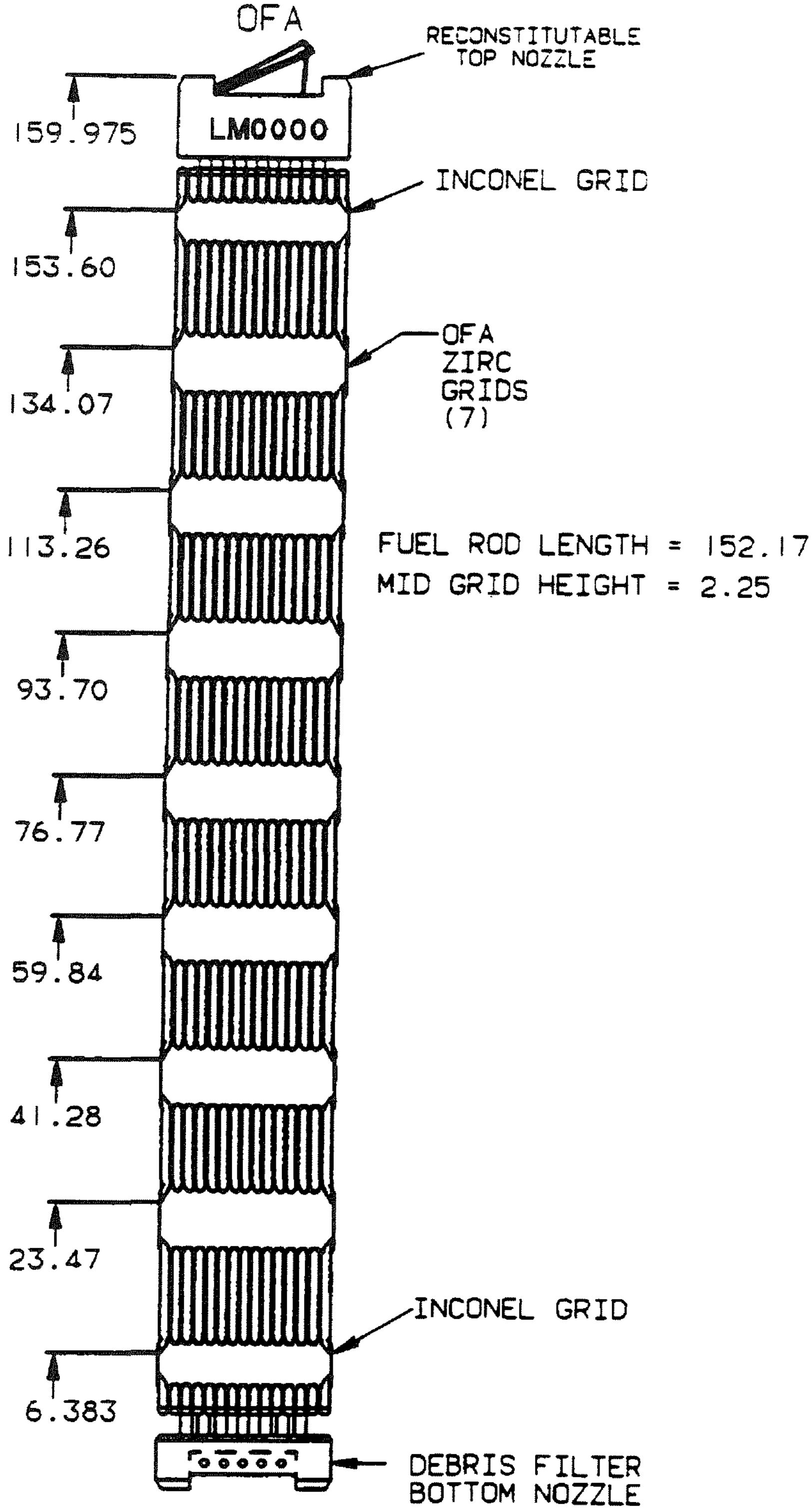
INDIAN POINT UNIT No. 2

UFSAR FIGURE 3.2-56

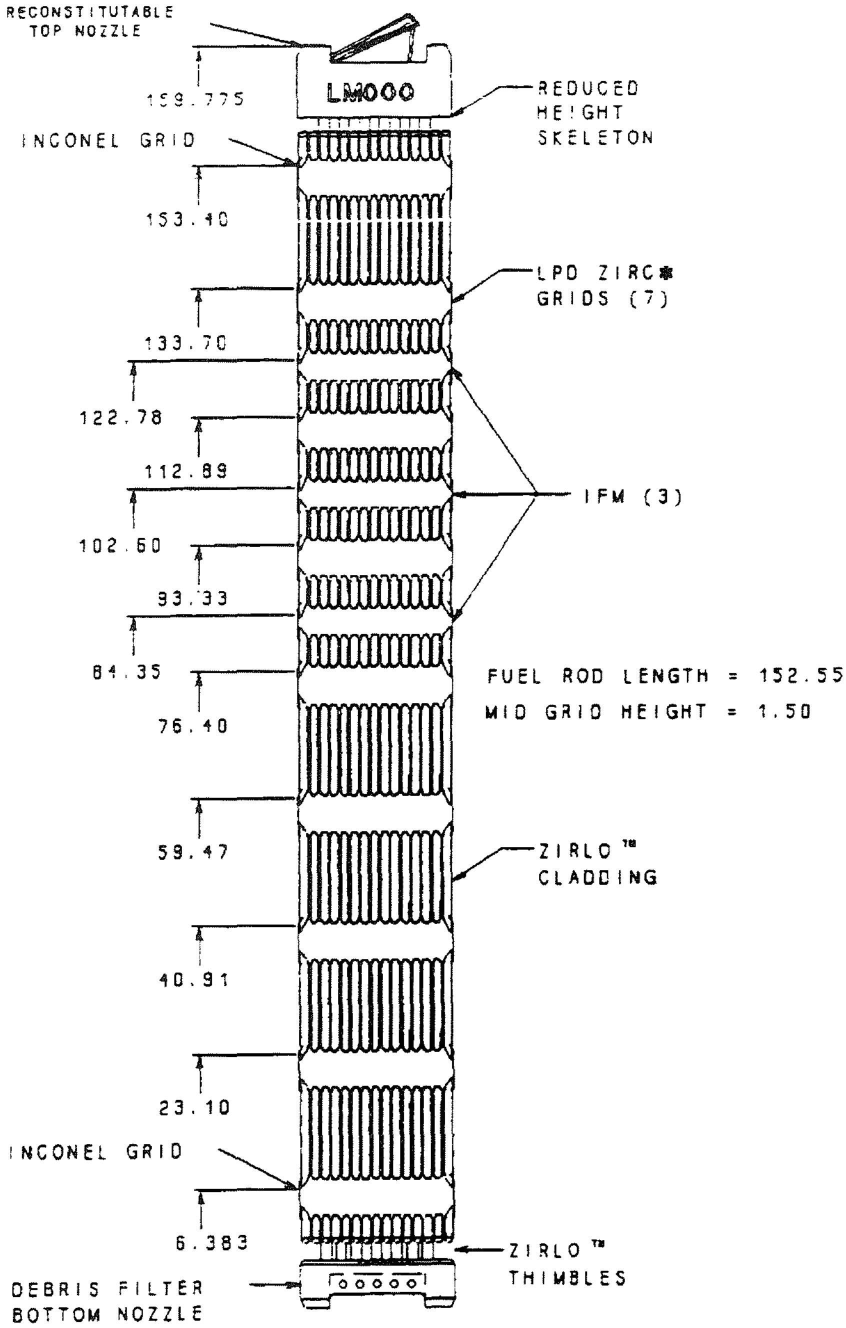
LOPAR FUEL ASSEMBLY

MIC. No. 1999MC3710	REV. No. 17A

OPTIMIZED (OFA) FUEL ASSEMBLY

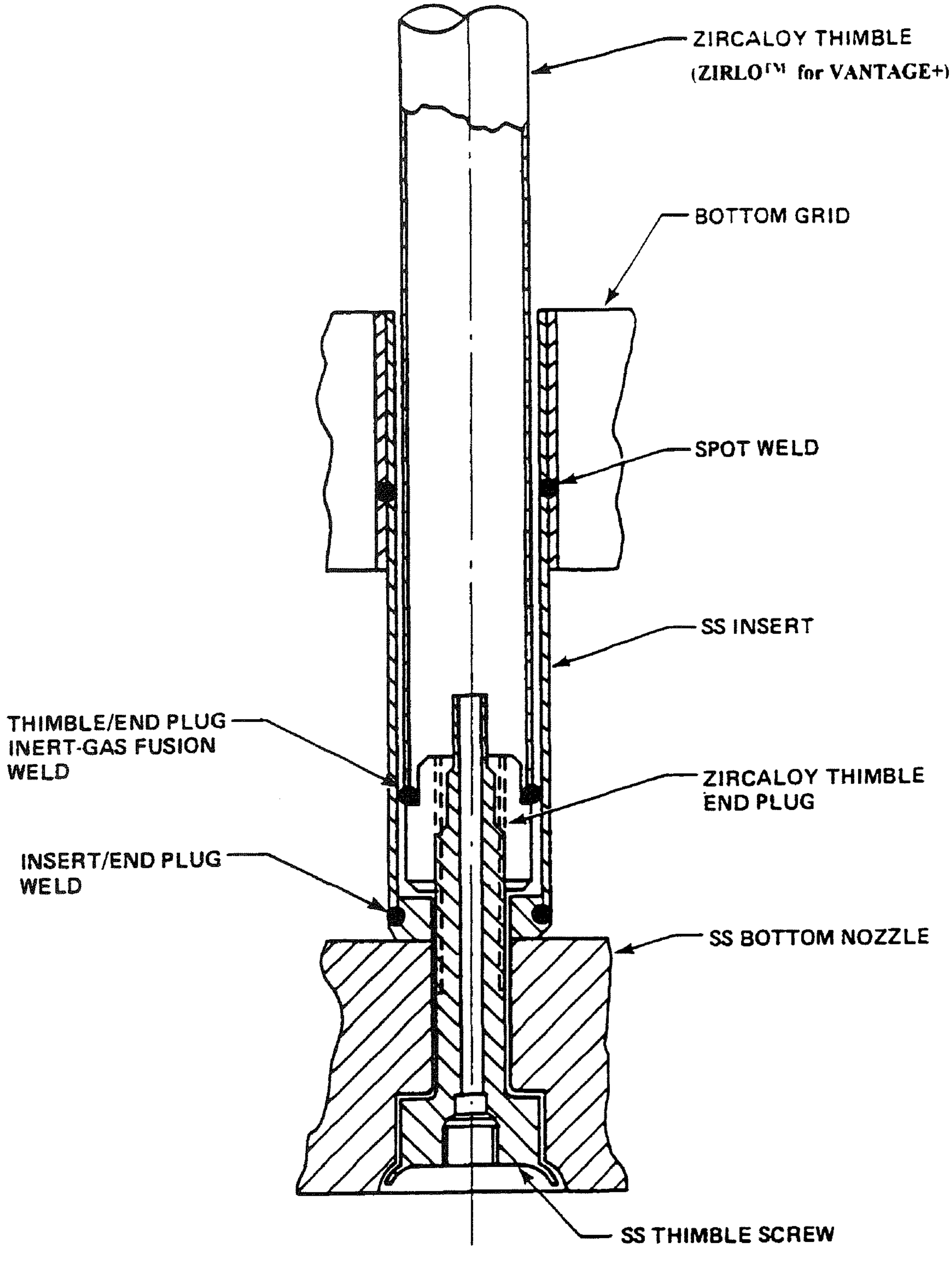


INDIAN POINT UNIT No. 2	
UFSAR FIGURE 3.2–56A	
OFA FUEL ASSEMBLY	
MIC. No. 1999MC3711	REV. No. 17A

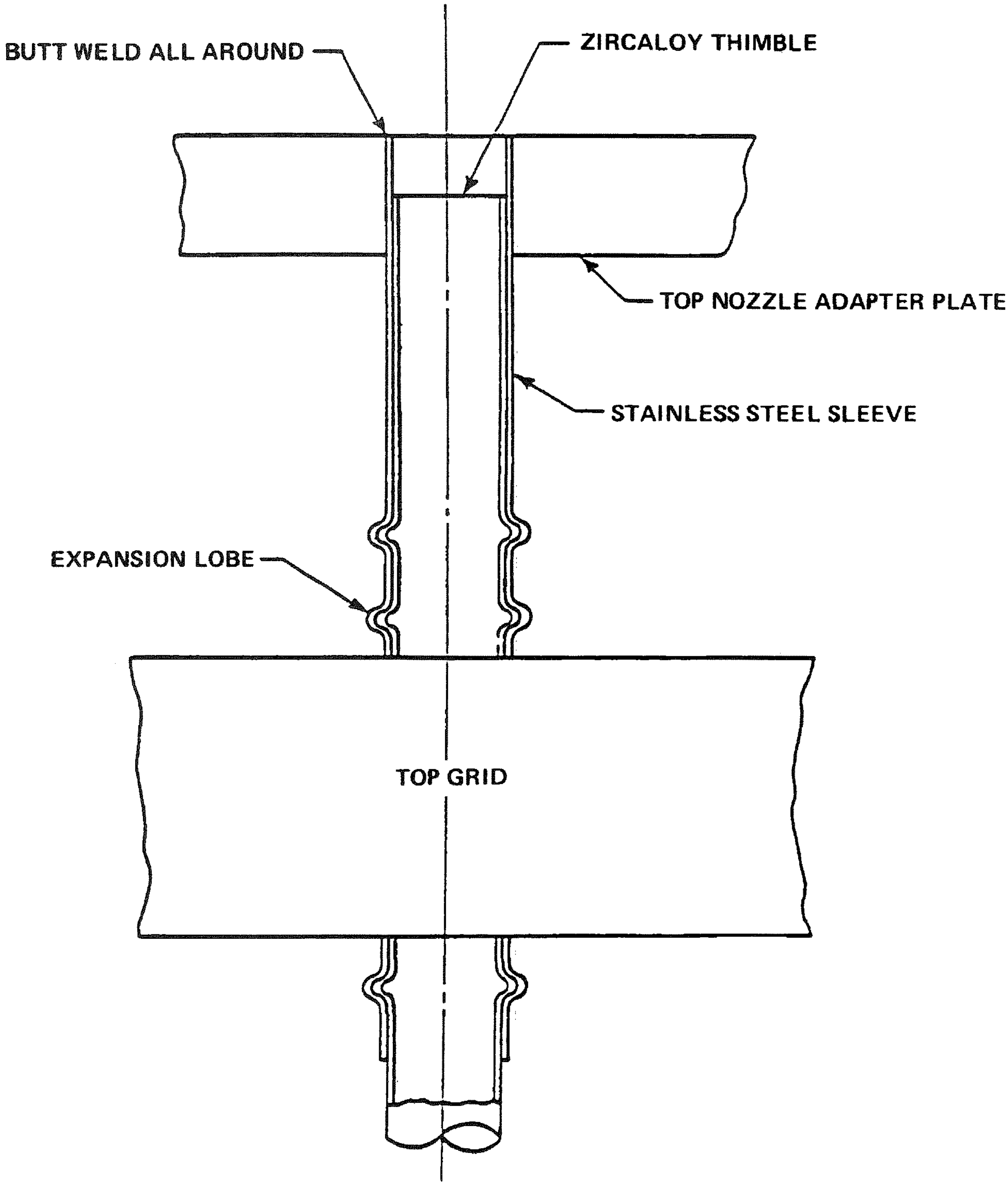


* VANTAGE+ FUEL WITH PERFORMANCE + FEATURES WILL HAVE ZIRLO™ MIDGRIDS.

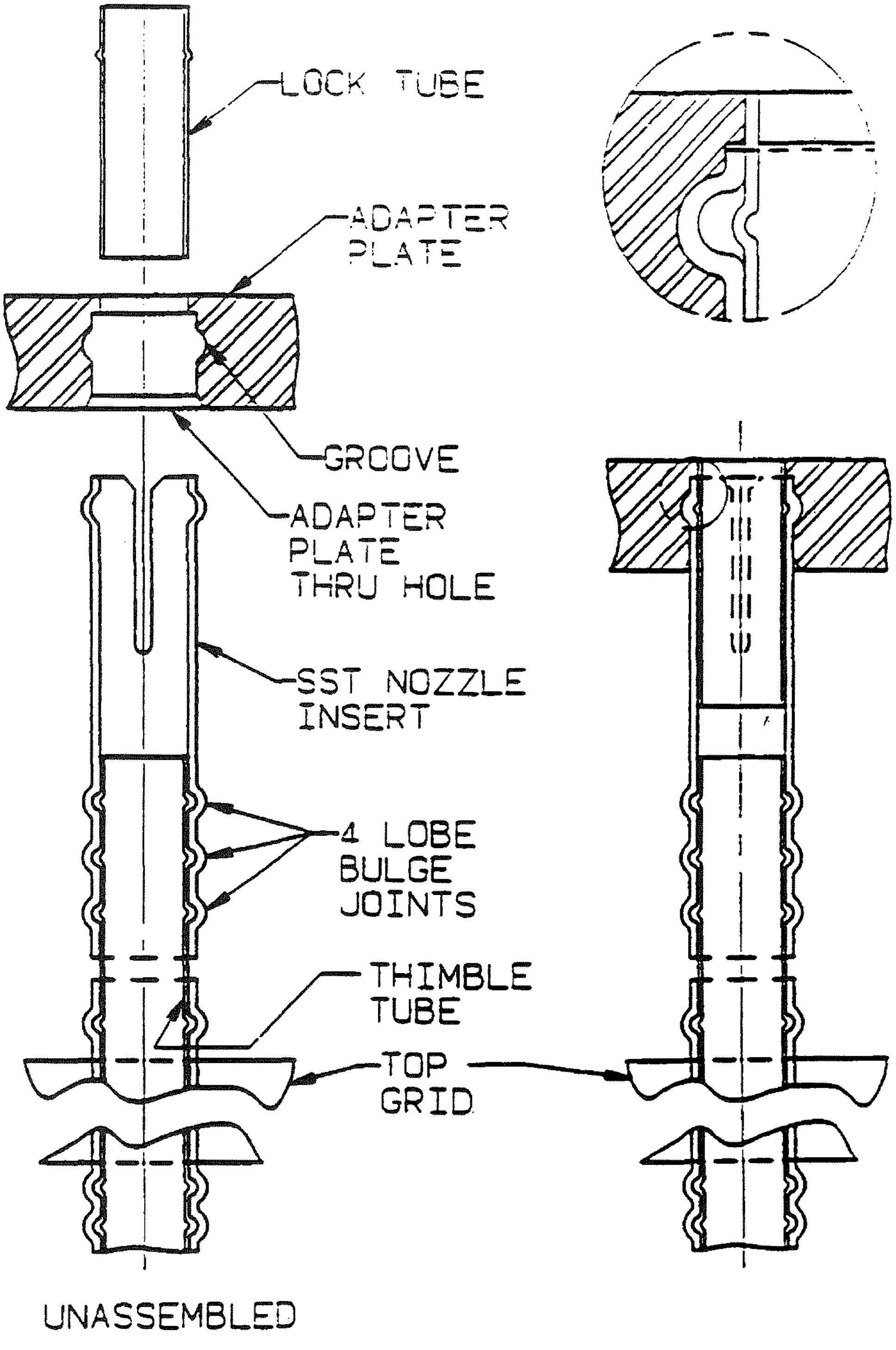
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 3.2-56B	
VANTAGE+ FUEL ASSEMBLY	
MIC. No. 1999MC3712	REV. No. 17A



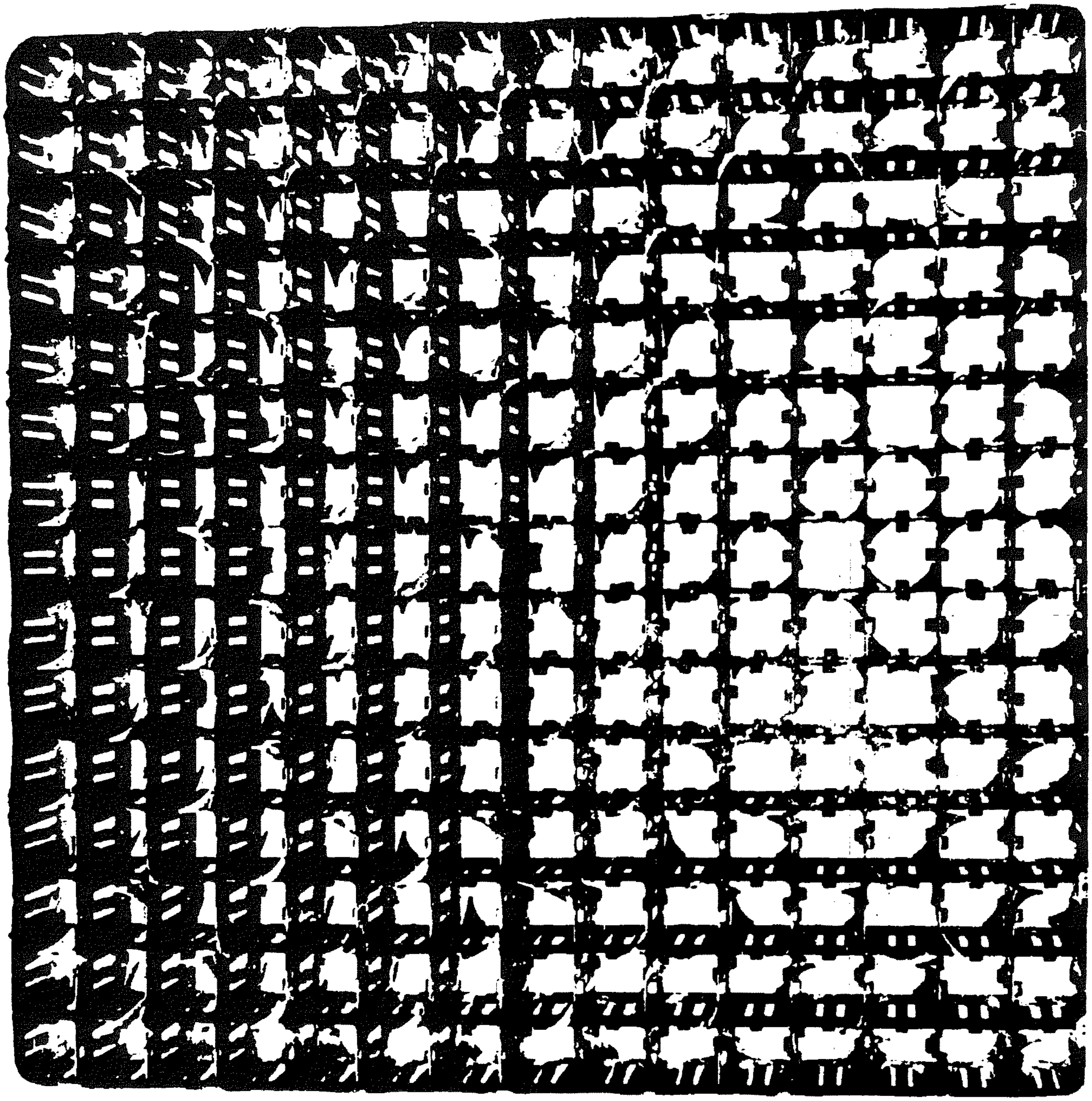
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 3.2-57 GUIDE THIMBLE TO BOTTOM NOZZLE JOINT	
MIC. No. 1999MC3713	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 3.2-58 LOPAR TOP GRID TO NOZZLE ATTACHMENT	
MIC. No. 1999MC3714	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 3.2–58A	
OFA and VANTAGE+ TOP GRID TO NOZZLE ATTACHMENT	
MIC. No. 1999MC3715	REV. No. 17A

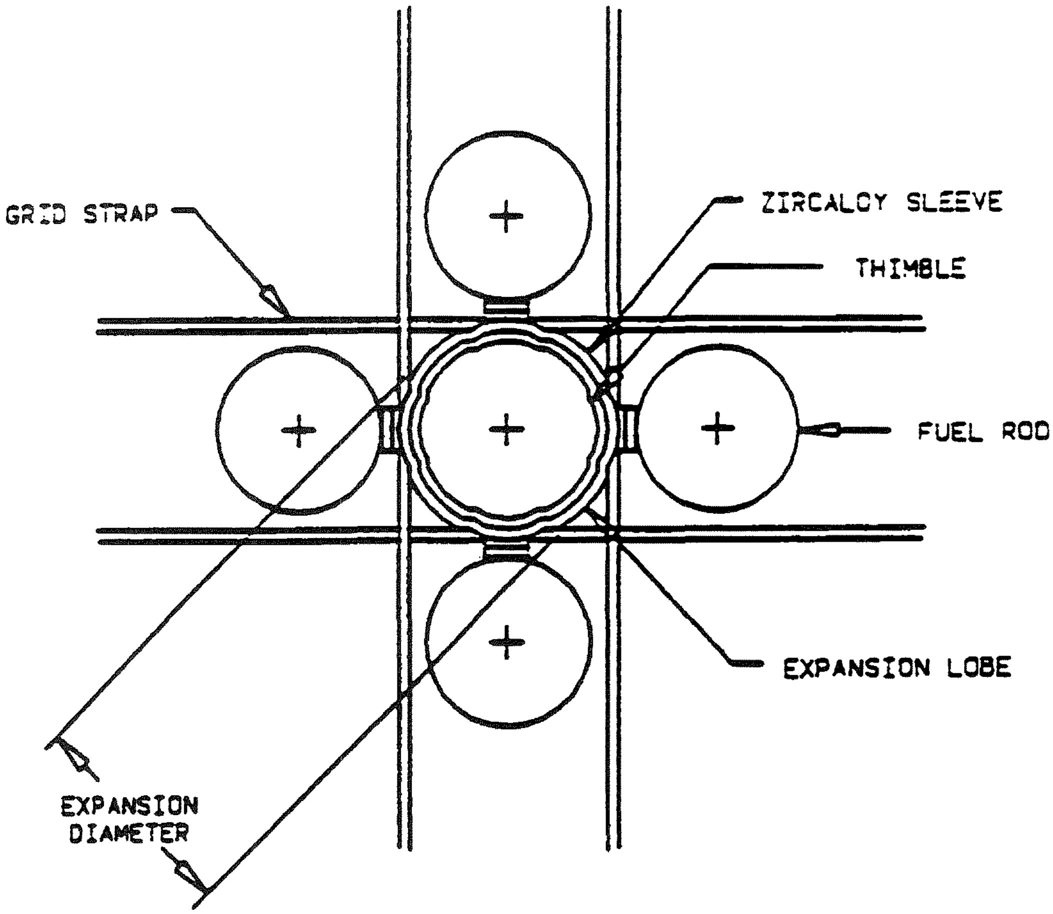


INDIAN POINT UNIT No. 2

UFSAR FIGURE 3.2-59

SPRING CLIP
GRID ASSEMBLY

MIC. No. 1999MC3716	REV. No. 17A

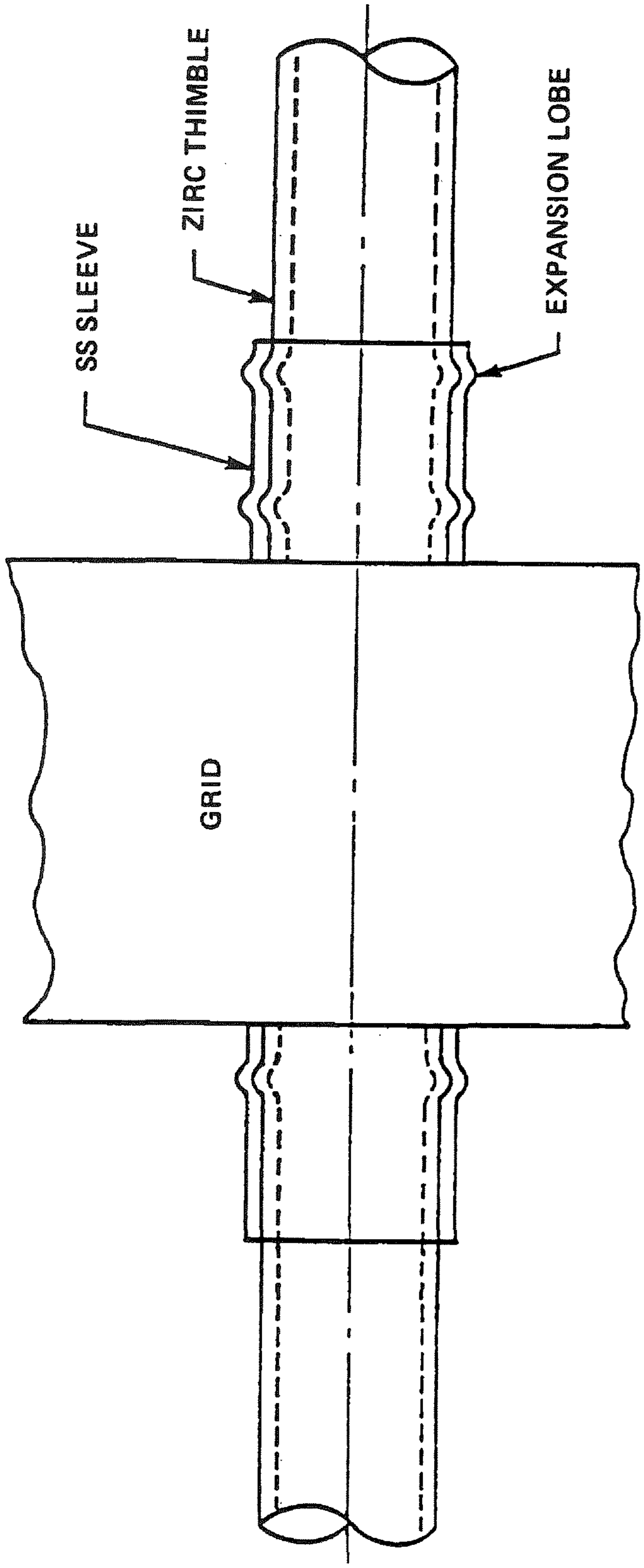


INDIAN POINT UNIT No. 2

UFSAR FIGURE 3.2–60

MID–GRID EXPANSION JOINT DESIGN
PLAN VIEW

MIC. No. 1999MC3717	REV. No. 17A

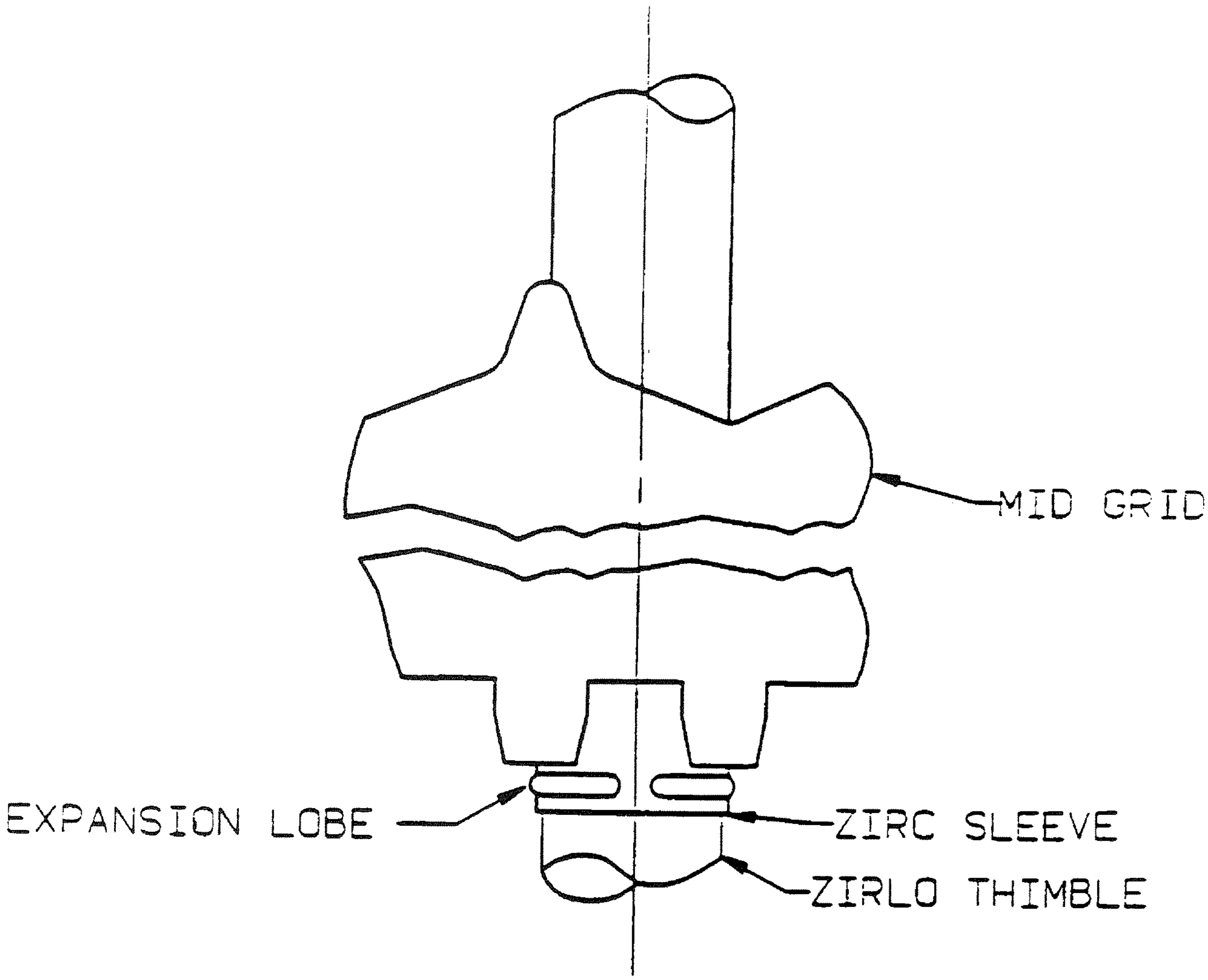


INDIAN POINT UNIT No. 2

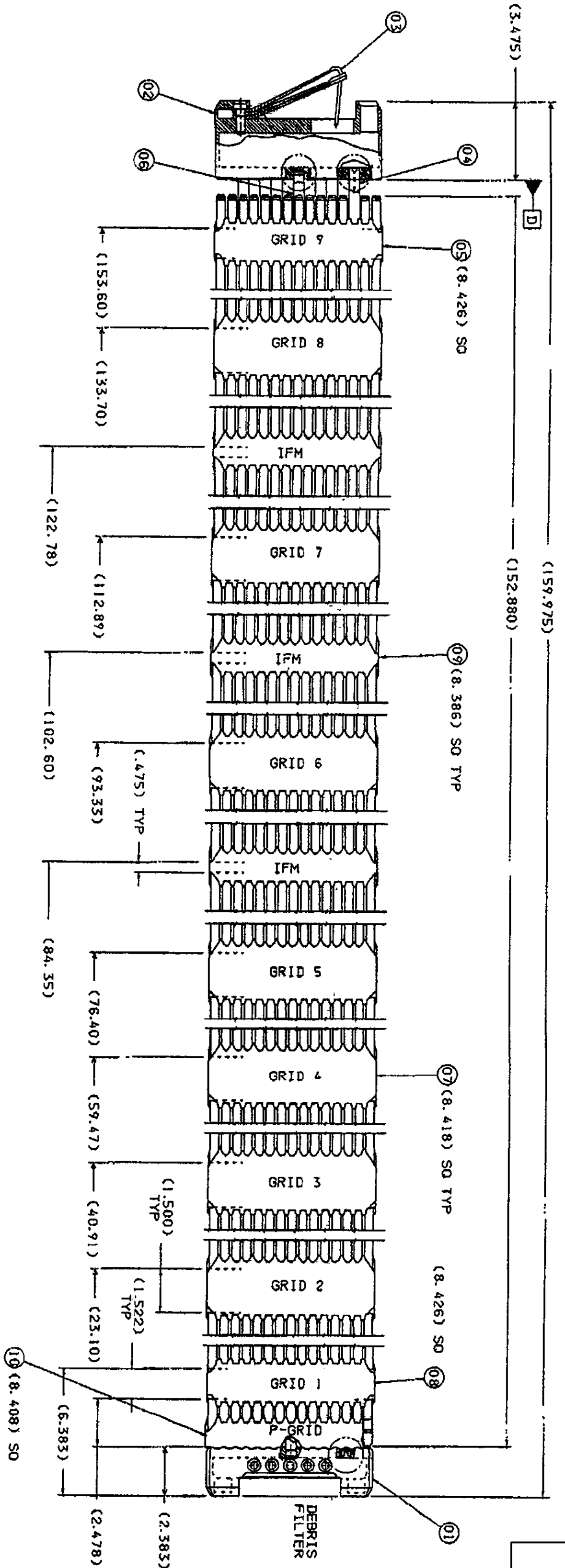
UFSAR FIGURE 3.2–61

ELEVATION VIEW – LOPAR GRID
TO THIMBLE ATTACHMENT

MIC. No. 1999MC3718	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 3.2-61A ELEVATION VIEW – VANTAGE+ GRID TO THIMBLE ATTACHMENT	
MIC. No. 1999MC3719	REV. No. 17A

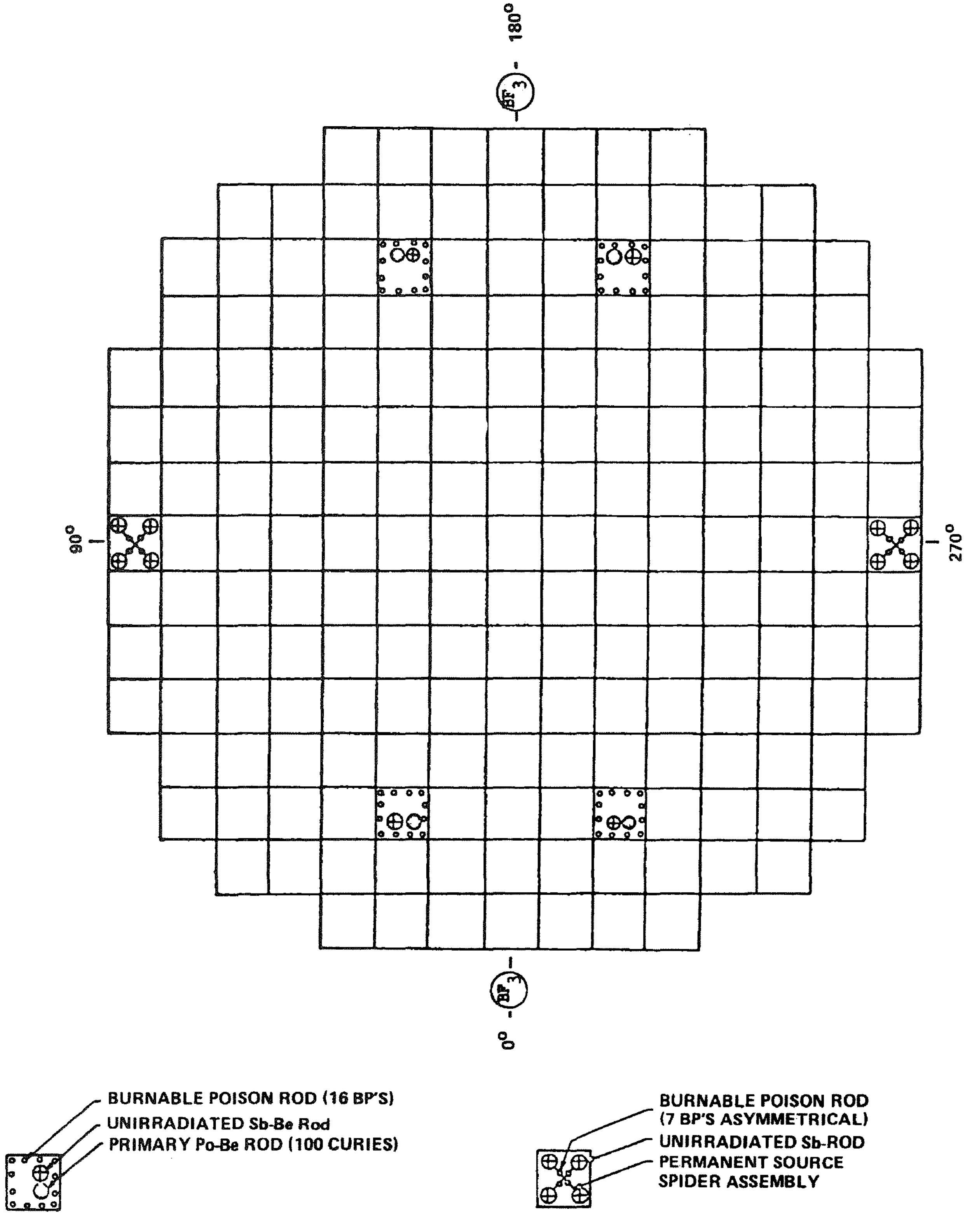


INDIAN POINT UNIT No. 2

UFSAR FIGURE 3.2-61B

VANTAGE+ FUEL ASSEMBLY
WITH PERFORMANCE+ ENHANCEMENTS

MIC. No. 1999MC3720	REV. No. 17B

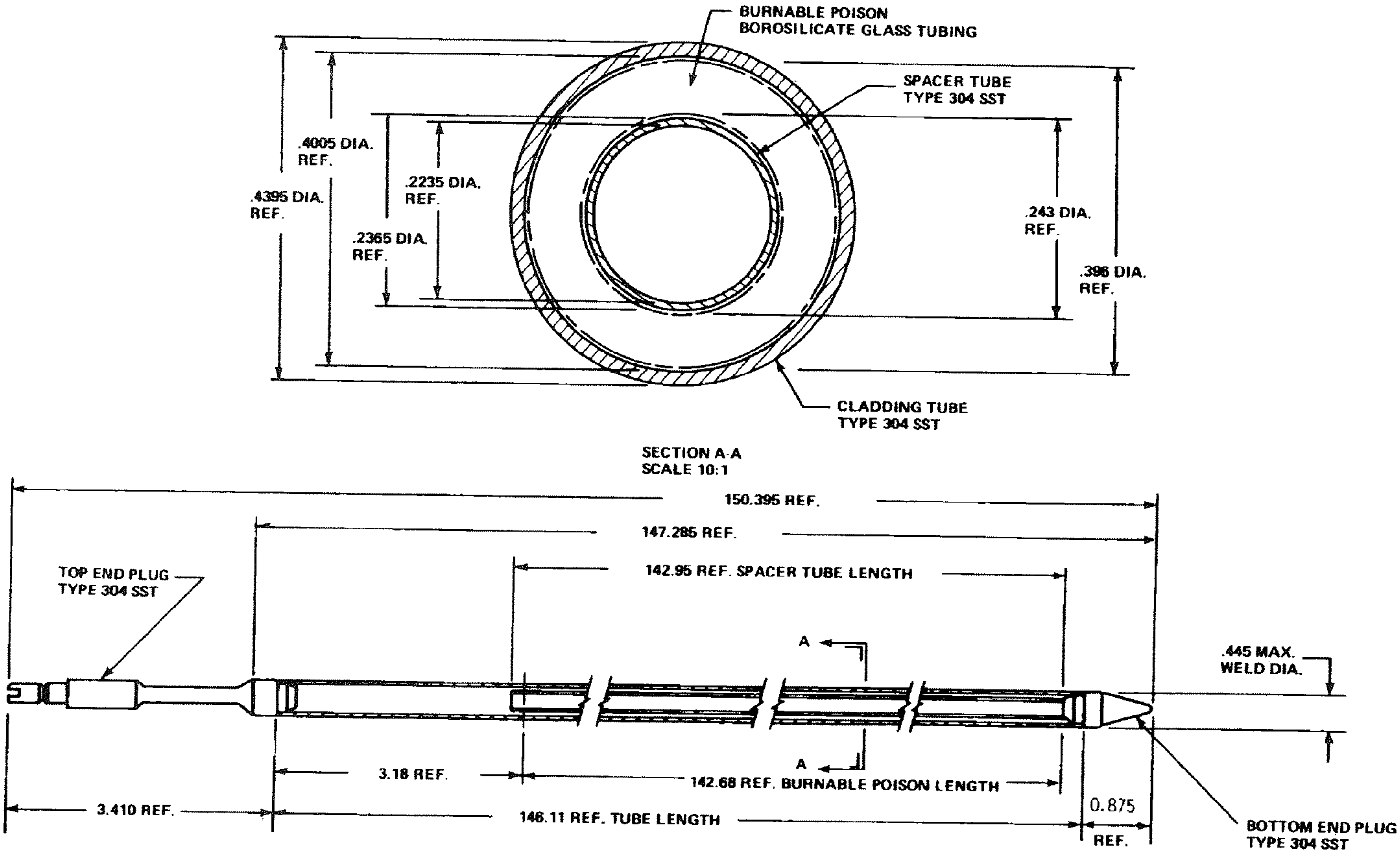


INDIAN POINT UNIT No. 2

UFSAR FIGURE 3.2-62

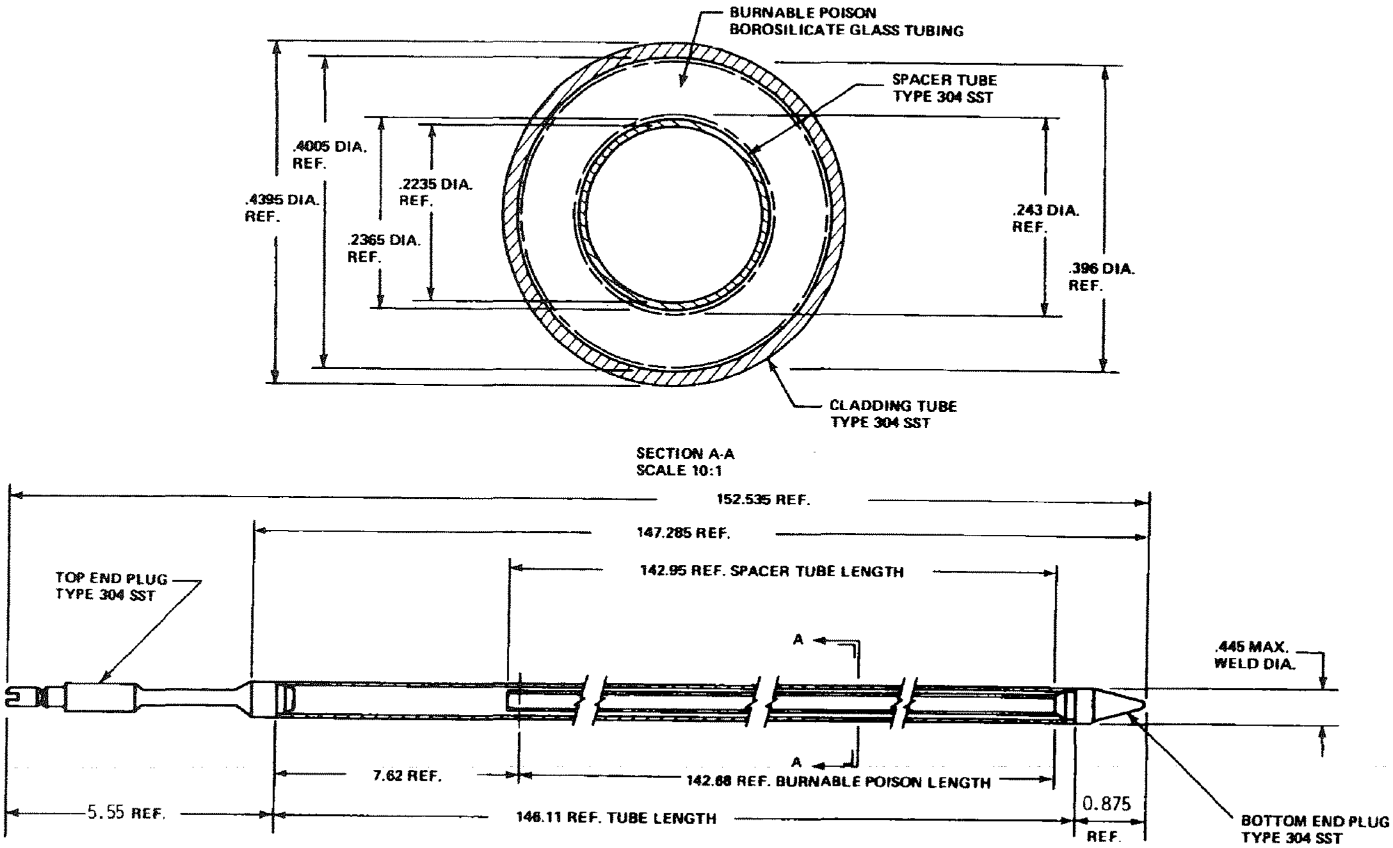
CYCLE 1 –
NEUTRON SOURCE LOCATIONS

MIC. No. 1999MC3721 | REV. No. 17A



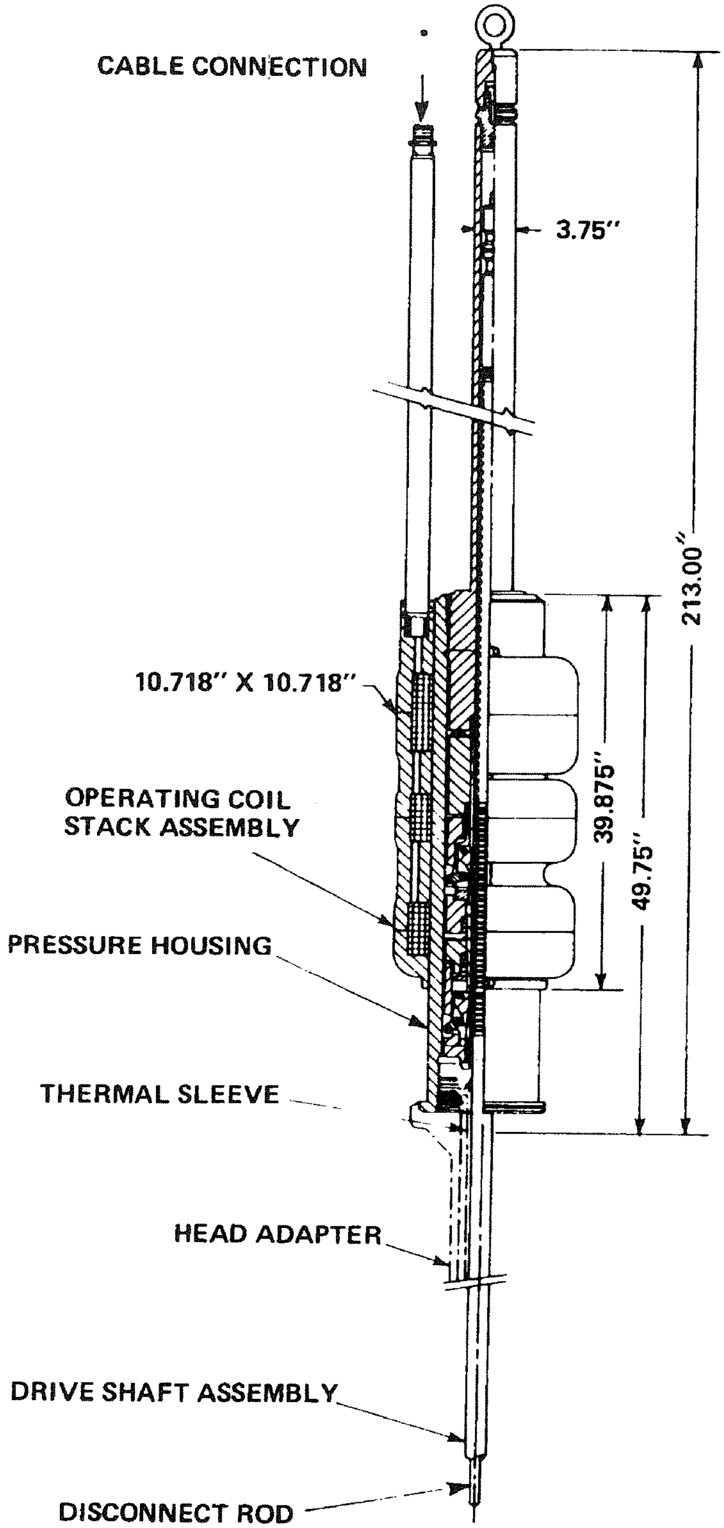
Note: Dimensions are expressed in inches.

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 3.2-63 HIPAR BURNABLE POISON ROD	
MIC. No. 1999MC3722	REV. No. 17A

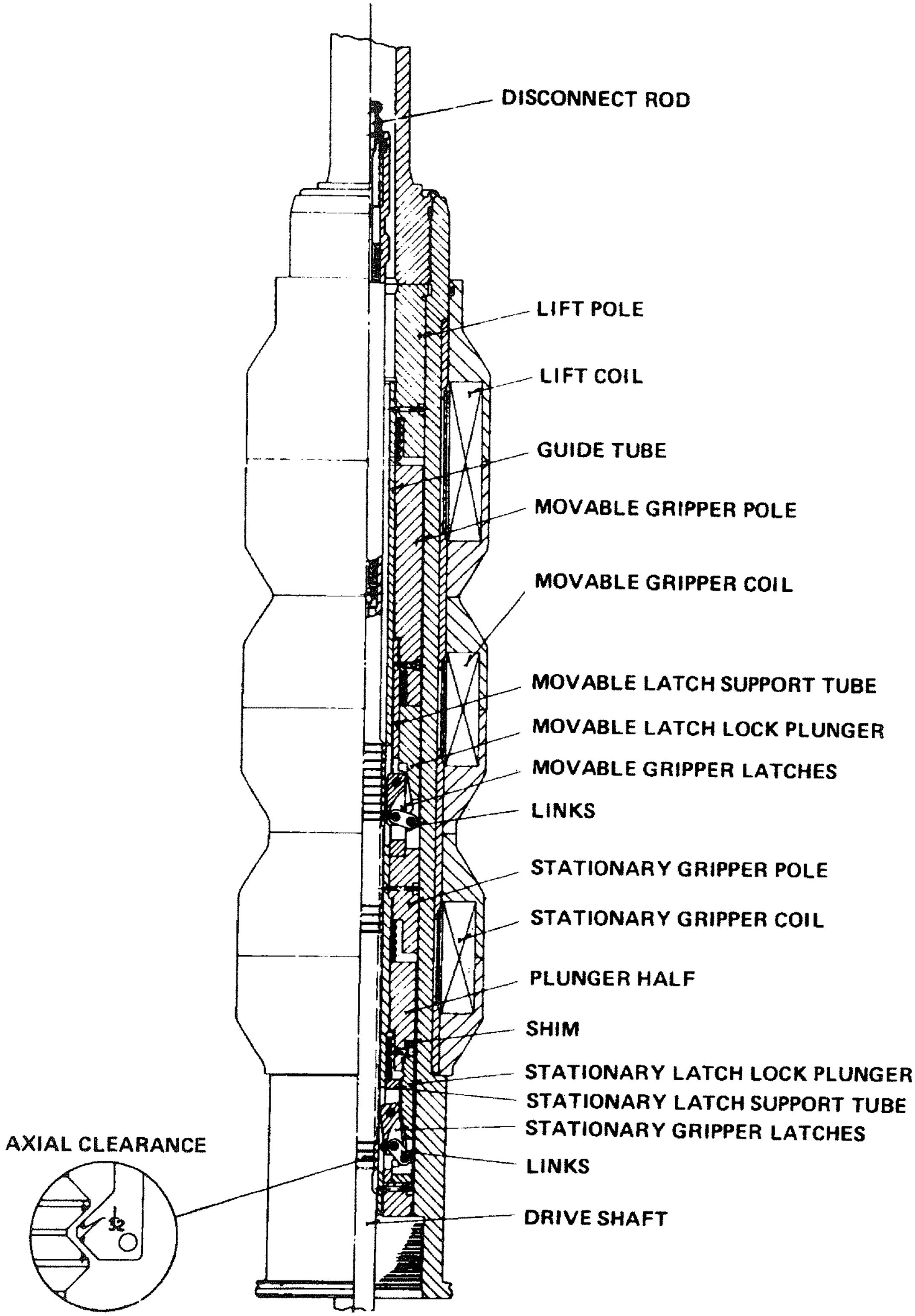


Note: Dimensions are expressed in inches.

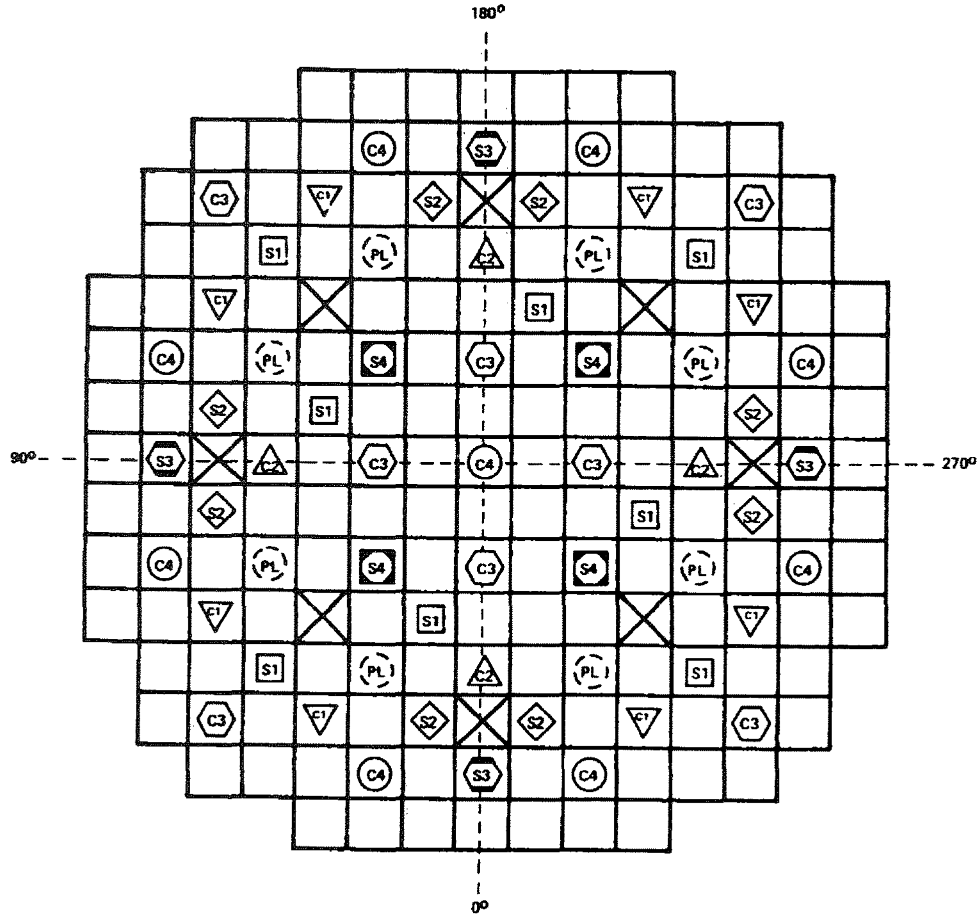
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 3.2-64 LOPAR BURNABLE POISON ROD	
MIC. No. 1999MC3723	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 3.2–65 CONTROL ROD DRIVE MECHANISM ASSEMBLY	
MIC. No. 1999MC3724	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 3.2–66	
CONTROL ROD DRIVE MECHANISM SCHEMATIC	
MIC. No. 1999MC3725	REV. No. 17A

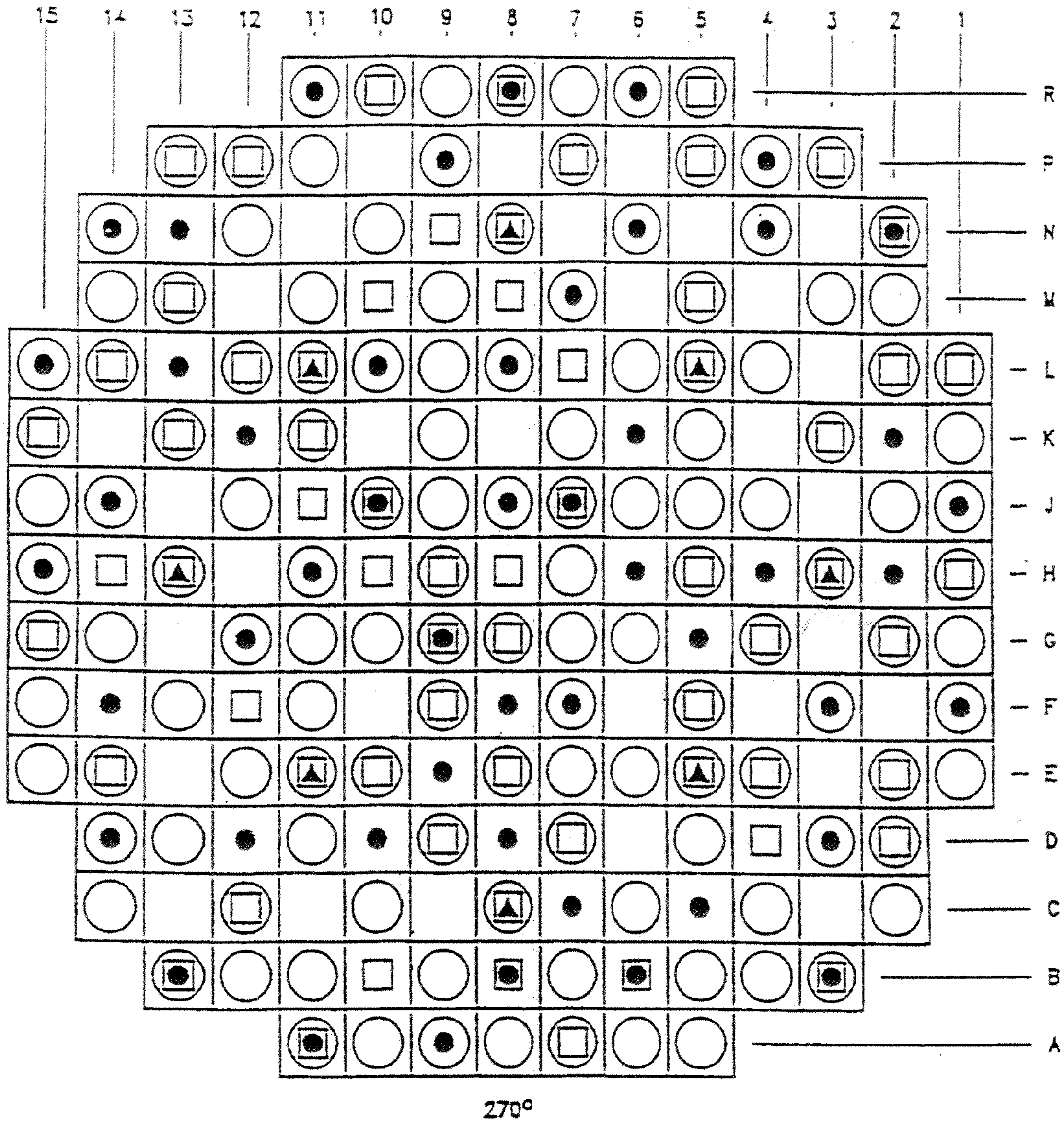


ROD CLUSTER CONTROL BANKS

BANK	SYMBOL
S1	□
S2	◇
S3	⬡
S4	□
C1	▽
C2	△
C3	⬡
C4	○
PL (PART-LENGTH ROD)*	◌
FIXED INCORE	╳

Note: Part-length rods have been removed since the orginal design.

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 3.2-67 THIMBLE LOCATIONS – FIXED INCORE DETECTORS	
MIC. No. 1999MC3726	REV. No. 17A



THERMOCOUPLE LOCATION........ 65

MOVABLE DETECTOR LOCATION.... 50

FLOW MIXING DEVICE LOCATION.. 132

FIXED DETECTOR LOCATION...... 8

INDIAN POINT UNIT No. 2	
INCORE DETECTOR, THERMOCOUPLE AND FLOW MIXING DEVICE LOCATIONS	
UFSAR FIGURE 3.2-68	REV. No. 19

WESTINGHOUSE PROPRIETARY CLASS 2

FIGURE 1

INDIAN POINT UNIT 2, CYCLE 20

REGION AND FUEL ASSEMBLY LOCATIONS

	15	14	13	12	11	10	9	8	7	6	5	4	3	2	1
R					21A W18	21A W13	21B W30	21A W03	21B W28	21A W09	20B V35				
P			21B W50	22B X07	22B X20	22B X23	22B X51	22B X65	22B X68	22B X35	22B X37	22B X06	21B W52		
N		21B W47	21B W62	22B X82	21B W89	22B X45	21B W66	22B X84	21B W92	22B X12	21B W75	22B X50	21B W84	21B W60	
M		22B X30	22B X93	21B W26	22B X56	21B W46	22B X31	21B W42	22B X27	21B W58	22B X77	21B W37	22B X78	22B X28	
L	20B V33	22B X19	21B W93	22B X73	21B W24	22B X75	21B W85	22B X22	21B W74	22B X64	21B W36	22B X67	21B W82	22B X05	21A W19
K	21A W10	22B X34	22B X18	21B W49	22B X49	21A W04	22B X81	21B W25	22B X63	21A W05	22B X58	21B W61	22B X14	22B X42	21A W12
J	21B W45	22B X89	21B W90	22B X40	21B W78	22B X87	21B W69	22A X01	21B W81	22B X80	21B W91	22B X33	21B W88	22B X96	21B W38
H	21A W07	22B X79	22B X76	21B W35	22B X48	21B W43	22A X02	21A W21	22A X03	21B W23	22B X32	21B W39	22B X86	22B X57	21A W06
G	21B W40	22B X91	21B W70	22B X29	21B W63	22B X55	21B W80	22A X04	21B W83	22B X66	21B W79	22B X39	21B W73	22B X85	21B W31
F	21A W14	22B X13	22B X44	21B W57	22B X71	21A W01	22B X88	21B W27	22B X90	21A W08	22B X95	21B W56	22B X11	22B X21	21A W15
E	21A W11	22B X38	21B W77	22B X52	21B W22	22B X94	21B W86	22B X26	21B W65	22B X53	21B W34	22B X61	21B W76	22B X41	20B V47
D		22B X43	22B X54	21B W29	22B X62	21B W48	22B X47	21B W41	22B X08	21B W51	22B X70	21B W44	22B X69	22B X46	
C		21B W55	21B W72	22B X59	21B W71	22B X09	21B W67	22B X83	21B W87	22B X16	21B W64	22B X74	21B W68	21B W54	
B			21B W53	22B X10	22B X17	22B X25	22B X92	22B X72	22B X60	22B X24	22B X15	22B X36	21B W59		
A					20B V41	21A W20	21B W33	21A W02	21B W32	21A W17	21A W16				

0°

270°

LEGEND

R — Region Identifier

ID — Fuel Assembly Identifier

Fuel Assembly Orientation

■ Reference Hole

○ Core Pin Hole

\ Holdown Bar

NOTE: Figures are Top View

INDIAN POINT UNIT No. 2

CYCLE 20 REGION AND FUEL ASSEMBLY LOCATIONS

UFSAR FIGURE 3.2-68A | REV. No. 22

WESTINGHOUSE PROPRIETARY CLASS 2

FIGURE 2

INDIAN POINT UNIT 2, CYCLE 20

CORE COMPONENTS AND FRESH INTEGRAL BA LOCATIONS

	15	14	13	12	11	10	9	8	7	6	5	4	3	2	1
R					DCPD	DCPD	DCPD	DCPD	DCPD	DCPD	DCPD				
P			DCPD	DCPD 116I	DCPD 116I	RCCA 116I	20W 148I	RCCA 148I	20W 148I	RCCA 116I	DCPD 116I	DCPD 116I	DCPD		
N		DCPD	RCCA	DCPD 148I	RCCA	20W 116I	RCCA	16W 148I	RCCA	20W 116I	RCCA	DCPD 148I	RCCA	DCPD	
M		DCPD 116I	DCPD 148I	RCCA	20W 148I	DCPD	20W 116I	RCCA	20W 116I	DCPD	20W 148I	RCCA	68SA 148I	DCPD 116I	
L	DCPD	DCPD 116I	RCCA	20W 148I	DCPD	16W 148I	DCPD	16W 116I	RCCA	16W 148I	DCPD	20W 148I	RCCA	DCPD 116I	DCPD
K	DCPD	RCCA 116I	20W 116I	DCPD	16W 148I	RCCA	20W 148I	RCCA	20W 148I	RCCA	16W 148I	DCPD	20W 116I	RCCA 116I	DCPD
J	DCPD	20W 148I	RCCA	20W 116I	RCCA	20W 148I	DCPD	20W 116I	DCPD	20W 148I	DCPD	20W 116I	RCCA	20W 148I	DCPD
H	DCPD	RCCA 148I	16W 148I	RCCA	16W 116I	RCCA	20W 116I	RCCA	20W 116I	RCCA	16W 116I	RCCA	16W 148I	RCCA 148I	DCPD
G	DCPD	20W 148I	RCCA	20W 116I	DCPD	20W 148I	DCPD	20W 116I	DCPD	20W 148I	RCCA	20W 116I	RCCA	20W 148I	DCPD
F	DCPD	RCCA 116I	20W 116I	DCPD	16W 148I	RCCA	20W 148I	RCCA	20W 148I	RCCA	16W 148I	DCPD	20W 116I	RCCA 116I	DCPD
E	DCPD	DCPD 116I	RCCA	20W 148I	DCPD	16W 148I	RCCA	16W 116I	DCPD	16W 148I	DCPD	20W 148I	RCCA	DCPD 116I	DCPD
D		DCPD 116I	68SA 148I	RCCA	20W 148I	DCPD	20W 116I	RCCA	20W 116I	DCPD	20W 148I	RCCA	DCPD 148I	DCPD 116I	
C		DCPD	RCCA	DCPD 148I	RCCA	20W 116I	RCCA	16W 148I	RCCA	20W 116I	RCCA	DCPD 148I	RCCA	DCPD	
B			DCPD	DCPD 116I	DCPD 116I	RCCA 116I	20W 148I	RCCA 148I	20W 148I	RCCA 116I	DCPD 116I	DCPD 116I	DCPD		
A					DCPD	DCPD	DCPD	DCPD	DCPD	DCPD	DCPD				

0°

270°

LEGEND

TYPE / ##I	
TYPE	COMPONENT TYPE
##I	NUMBER OF FRESH IFBA RODS

CORE COMPONENT TYPES

DCPD - DUALLY COMPATIBLE PLUGGING DEVICE
RCCA - CONTROL OR SHUTDOWN RCCA
##W - NUMBER OF RODLETS ON WABA ASSEMBLY
##SA - NUMBER OF RODLETS ON SECONDARY SOURCE ASSEMBLY

Fuel Assembly Orientation

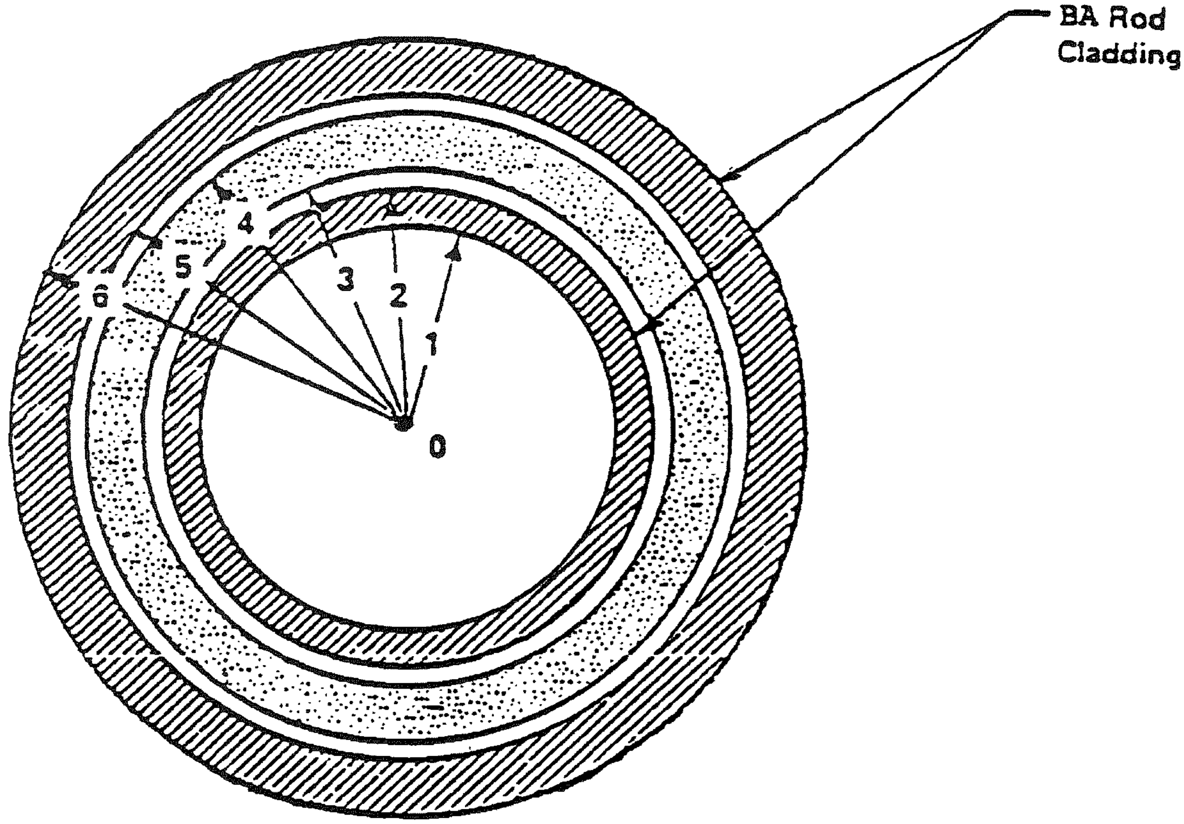
Reference Hole
O Core Pin Hole
\ Holdown Bar

NOTE: Figures are Top View
COMPONENT ORIENTATION SHOWN IN TABLE 2

INDIAN POINT UNIT No. 2

CYCLE 20 CORE COMPONENTS AND FRESH IFBA LOCATIONS

UFSAR FIGURE 3.2–68B | REV. No. 22



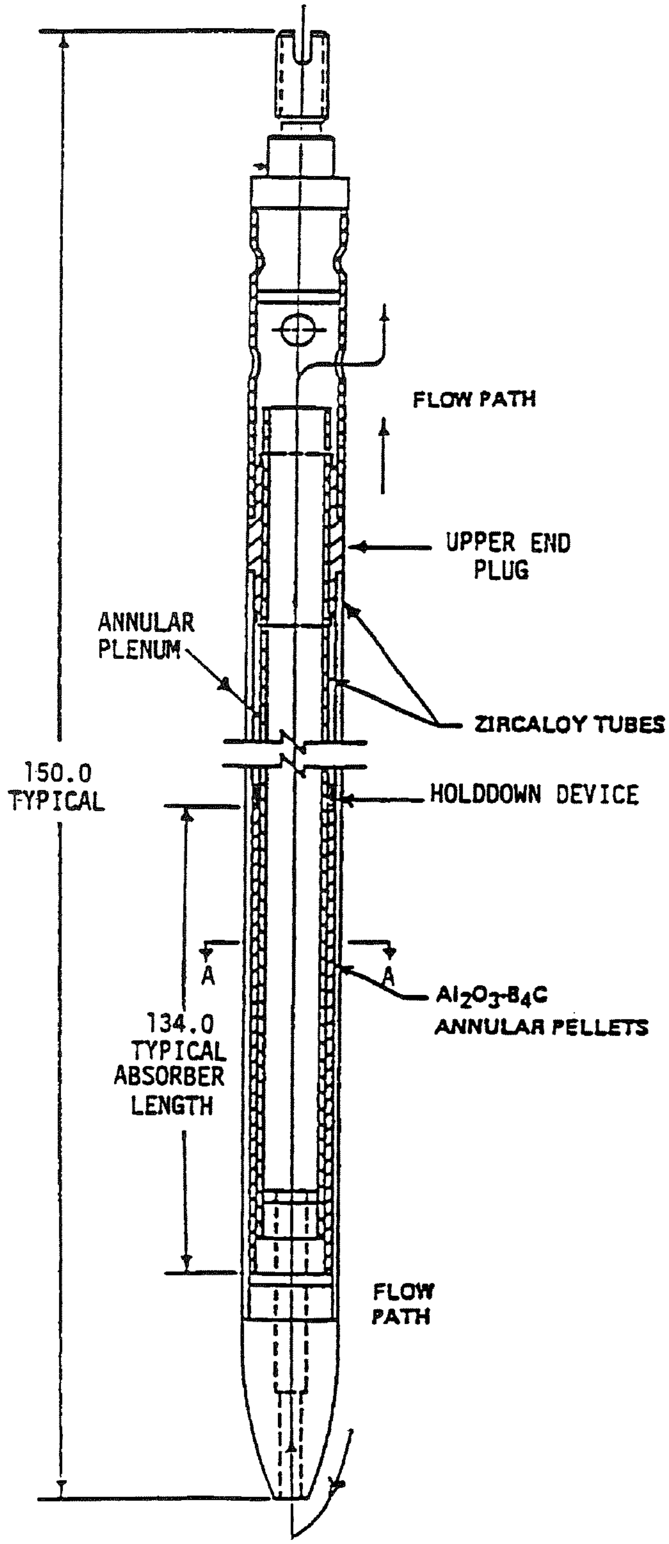
	Zone Number	Previous Design BA	WABA Design
	0 - 1	Air	Water
	1 - 2	Stainless steel	Zircaloy
	2 - 3	Air	Helium
	3 - 4	Borosilicate glass	$Al_2O_3 \cdot B_4C$
	4 - 5	Air	Helium
	5 - 6	Stainless steel	Zircaloy

INDIAN POINT UNIT No. 2

UFSAR FIGURE 3.2-69

COMPARISON OF BOROSILICATE GLASS ABSORBER ROD WITH WABA ROD

MIC. No. 1999MC3730 | REV. No. 17A



INDIAN POINT UNIT No. 2
UFSAR FIGURE 3.2-70 WET ANNULAR BURNABLE ABSORBER ROD

MIC. No. 1999MC3732	REV. No. 17A

CHAPTER 4
REACTOR COOLANT SYSTEM

4.0 GENERAL DESCRIPTION

The reactor coolant system, shown in Plant Drawing 9321-2738 [Formerly UFSAR Figure 4.2-1], consists of four similar heat transfer loops connected in parallel to the reactor vessel. Each loop contains a reactor coolant circulation pump and a steam generator. The system also includes a pressurizer, pressurizer relief tank, connecting piping, and instrumentation necessary for operational control.

4.1 DESIGN BASES

4.1.1 Performance Objectives

The reactor coolant system transfers the heat generated in the core to the steam generators where steam is generated to drive the turbine generator. Demineralized light water is circulated at the flow rate and temperature consistent with achieving the reactor core thermal-hydraulic performance presented in Section 3.2. The water also acts as a neutron moderator and reflector and as a solvent for the neutron absorber used in chemical shim control.

The reactor coolant system provides a boundary for containing the coolant under operating temperature and pressure conditions. It serves to confine radioactive material and limits any uncontrolled release to the secondary system and to other parts of the plant to acceptable values under conditions of either normal or abnormal reactor behavior. During transient operation, the system heat capacity attenuates thermal transients generated by the core or extracted by the steam generators. The reactor coolant system accommodates coolant volume changes within the protection system criteria.

By appropriate selection of the inertia of the reactor coolant pumps, the thermal-hydraulic effects are reduced to a safe level during the pump coastdown that would result from a loss-of-offsite power situation. The layout of the system ensures the natural circulation capability following a loss of offsite power, to permit decay heat removal without overheating the core. Portions of the system piping are used by the safety injection system to deliver cooling water to the core during a loss-of-coolant accident.

4.1.2 General Design Criteria

General design criteria (GDC) that apply to the reactor coolant system are given below.

4.1.2.1 Quality Standards

Criterion: Those systems and components of reactor facilities, which are essential to the prevention, or the mitigation of the consequences, of nuclear accidents, which could cause undue risk to the health and safety of the public shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes and standards pertaining to design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance

programs, test procedures, and inspection acceptance criteria to be used shall be identified. An indication of the applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance criteria used is required. Where such items are not covered by applicable codes and standards, a showing of adequacy is required. (GDC 1)

The reactor coolant system is of primary importance with respect to its safety function in protecting the health and safety of the public.

Quality standards of material selection, design, fabrication, and inspection conform to the applicable provisions of recognized codes and good nuclear practice (Section 4.1.7). Details of the quality assurance programs, test procedures, and inspection acceptance levels are given in Sections 4.3.1 and 4.5. Particular emphasis is placed on the assurance of quality of the reactor vessel to obtain material whose properties are uniformly within tolerances appropriate to the application of the design methods of the code.

4.1.2.2 <u>Performance Standards</u>

Criterion: Those systems and components of reactor facilities, which are essential to the prevention or to the mitigation of the consequences of nuclear accidents, which could cause undue risk to the health and safety of the public shall be designed, fabricated, and erected to performance standards that will enable such systems and components to withstand, without undue risk to the health and safety of the public, the forces that might reasonably be imposed by the occurrence of an extraordinary natural phenomenon such as earthquake, tornado, flooding condition, high wind, or heavy ice. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been officially recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design. (GDC 2)

All piping, components, and supporting structures of the reactor coolant system are designed to Class I requirements, as discussed in Sections 1.11 and 4.1.4.3.

The reactor coolant system is located in the containment whose design, in addition to being a Class I structure, also considers accidents or other applicable natural phenomena. Details of the containment design are given in Chapter 5.

4.1.2.3 <u>Records Requirements</u>

Criterion: The reactor licensee shall be responsible for assuring the maintenance throughout the life of the reactor of records of the design, fabrication, and construction of major components of the plant essential to avoid undue risk to the health and safety of the public. (GDC 5)

Records of the design of the major reactor coolant system components and the related engineered safety features components are maintained for the life of the plant.

Records of fabrication are maintained in the manufacturers' plants as required by the appropriate code or other requirements pending submittal to Westinghouse or ENIP2. They are

available at any time throughout the life of the plant. Records of changes made to the plant as described in the FSAR are maintained for the life of the plant.

4.1.2.4 Missile Protection

Criterion: Adequate protection for those engineered safety features, the failures of which could cause an undue risk to the health and safety of the public, shall be provided against dynamic effects and missiles that might result from plant equipment failures. (GDC 40)

4.1.2.4.1 Original Design Basis

The dynamic effects during blowdown following a loss-of-coolant accident are evaluated in the detailed layout and design of the high-pressure equipment and barriers that afford missile protection. Support structures are designed with consideration given to fluid and mechanical thrust loadings.

Original plant design basis required that the steam generators be supported, guided, and restrained in a manner that prevents rupture of the steam side of a generator, the steam lines, and the feedwater piping as a result of forces created by a reactor coolant system pipe rupture. These supports, guides, and restraints also prevent rupture of the primary side of a steam generator as a result of forces created by a steam or feedwater line rupture.

Original plant design basis also required that the mechanical consequences of a pipe rupture as a result of forces created by a reactor coolant system pipe rupture be restricted by design such that the functional capability of the engineered safety features would not be impaired.

4.1.2.4.2 Revised Design Basis

In 1989, the NRC approved elimination of the necessity for considering and protecting against dynamic effects of postulated primary loop pipe ruptures from the design basis of Indian Point Unit 2. "Leak before break" technology was applied as permitted by revised General Design Criterion 4 of 10CFR50, Appendix A. References 1, 2, 3 and 4 contain further information.

With the elimination of the necessity for considering and protecting against the dynamic effects of postulated primary loop ruptures from the design basis of Indian Point 2, breaks have been postulated in the following branch lines: the Accumulator branch line in the cold leg, the Pressurizer Surge line and the Residual Heat Removal (RHR) line in the hot leg. This is discussed in Reference 5. These breaks are the new design basis breaks with respect to considering and protecting against the dynamic effects of postulated ruptures.

In general, these new breaks are significantly less severe than the original design basis breaks. The following is a description of how the dynamic effects of these new breaks have been considered and protected against. The dynamic effects have been grouped into three categories as follows:

1. Containment subcompartment pressurization
2. Break reaction forces (i.e. forcing function analysis, asymmetric blowdown loading) used in structural support design and for confirming the structural integrity of systems and components
3. Missile Protection (pipe whip, jet impingement and missiles)

4.1.2.4.2.1 Containment Subcompartment Pressurization

The dynamic effects of these new breaks with respect to containment subcompartment pressurization is discussed in Section 14.3.5.4.3.2.

4.1.2.4.2.2 Break Reaction Forces

As discussed in Reference 5, the new break locations were used to develop hydraulic forcing functions for revised structural analyses. These analyses demonstrate that for the new break locations the structural integrity of the reactor vessel internals, core components including fuel assemblies, and the reactor coolant loop will be maintained and will preserve the ability to maintain a coolable geometry for the rated stretch power conditions. The dynamic effects of postulated breaks with respect to break reaction forces is discussed further in Sections 1.3.7, 1.11.3, 4.2.4, 4.3.1.3, Appendix 4B.7, 5.1.1.1.5, 5.1.5, 6.1.1.5, and 6.2.2.5.

4.1.2.4.2.3 Missile Protection

Since the new break locations remain inside the missile barrier, engineered safety features and associated systems remain protected from loss of function due to dynamic effects and missiles, which might result from these breaks. This protection is discussed further in Sections 1.3.7, 4.2.4, 5.1.2.5, 6.1.1.3, 6.2.2.5, and 8.2.2.6.

4.1.3 Principal Design Criteria

The criteria that apply solely to the reactor coolant system are given below.

4.1.3.1 Reactor Coolant Pressure Boundary

Criterion: The reactor coolant pressure boundary shall be designed, fabricated and constructed so as to have an exceedingly low probability of gross rupture of significant uncontrolled leakage throughout its design lifetime. (GDC 9)

The reactor coolant system in conjunction with its control and protective provisions is designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions and to maintain the stresses within applicable code stress limits.

Fabrication of the components that constitute the pressure retaining boundary of the reactor coolant system is carried out in strict accordance with the applicable codes. In addition, there are areas where equipment specifications for reactor coolant system components go beyond the applicable codes. Details are given in Section 4.5.1.

The materials of construction of the pressure retaining boundary of the reactor coolant system are protected by control of coolant chemistry from corrosion phenomena that might otherwise reduce the system's structural integrity during its service lifetime.

System conditions resulting from anticipated transients or malfunctions are monitored, and appropriate action is automatically initiated to maintain the required cooling capability and to limit system conditions so that continued safe operation is possible.

The system is protected from overpressure by means of pressure-relieving devices as required by Section III of the ASME Boiler and Pressure Vessel Code. Isolated sections of the system are provided with overpressure-relieving devices discharging to closed systems such that the system code allowable relief pressure within the protected section is not exceeded.

4.1.3.2 Monitoring Reactor Coolant Leakage

Criterion: Means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary. (GDC 16)

Positive indications in the control room of leakage of coolant from the reactor coolant system to the containment are provided by equipment that permits continuous monitoring of containment air activity and humidity and of runoff from the condensate-collecting pans under the cooling coils of the containment air recirculation units. This equipment provides indication of normal environmental conditions within the containment. Any increase in the observed parameters could be an indication of change within the containment, and the equipment provided is capable of monitoring this change. The basic design criterion is the detection of deviations from normal containment environmental conditions including air particulate activity, radiogas activity, humidity, condensate runoff, and in the case of significant leakage, the liquid inventory in the process systems and containment sump.

Further details are supplied in Sections 4.2.7 and 6.7.

4.1.3.3 Reactor Coolant Pressure Boundary Capability

Criterion: The reactor coolant pressure boundary shall be capable of accommodating without rupture the static and dynamic loads imposed on any boundary component as a result of an inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition. (GDC 33)

The reactor coolant boundary is shown to be capable of accommodating without further rupture the static and dynamic loads imposed as a result of a sudden reactivity insertion such as a rod ejection. Details of this analysis are provided in Section 14.2.6.10.

The operation of the reactor is such that the severity of an ejection accident is inherently limited. Since control rod clusters are used to control load variations only and core depletion is followed with boron dilution, only the rod cluster control assemblies in the controlling groups are inserted in the core at power, and at full power these rods are only partially inserted. A rod insertion limit monitor is provided as an administrative aid to the operator to ensure that this condition is met.

By using the flexibility in the selection of control rod groupings, radial locations and positions as a function of load, the design limits the maximum fuel temperature for the highest worth ejected rod to a value that precludes any resultant damage to the primary system pressure boundary, i.e., gross fuel dispersion in the coolant and possible excessive pressure surges.

The failure of a rod mechanism housing causing a rod cluster to be rapidly ejected from the core is evaluated as a theoretical, though not a credible, accident. While limited fuel damage could result from this hypothetical event, the fission products are confined to the reactor coolant

system and the reactor containment. The environmental consequences of rod ejection are less severe than from the hypothetical loss of coolant, for which public health and safety are shown to be adequately protected. Refer to Section 14.2.6.

4.1.3.4 Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention

Criterion: The reactor coolant pressure boundary shall be designed and operated to reduce to an acceptable level the probability of rapidly propagating type failure. Consideration is given (a) to the provisions for control over service temperature and irradiation effects, which may require operational restrictions, (b) to the design and construction of the reactor pressure vessel in accordance with applicable codes, including those, which establish requirements for absorption of energy within the elastic strain energy range and for absorption of energy by plastic deformation and (c) to the design and construction of reactor coolant pressure boundary piping and equipment in accordance with applicable codes. (GDC 34)

The reactor coolant pressure boundary is designed to reduce to an acceptable level the probability of a rapidly propagating type failure. In the core region of the reactor vessel it is expected that the notch toughness of the material will change as a result of fast neutron exposure. This change is evidenced as a shift in the nil-ductility transition temperature (NDTT), which is factored into the operating procedures in such a manner that full operating pressure is not obtained until the affected vessel material is above the design transition temperature (DTT) in the ductile material region. The pressure during startup and shutdown at the temperature below NDTT is maintained below the threshold of concern for safe operation.

The DTT is a minimum of NDTT plus 60°F and dictates the procedures to be followed in the hydrostatic test and in station operations to avoid excessive cold stress. The value of the DTT is increased during the life of the plant as required by the expected shift in NDTT and as confirmed by the experimental data obtained from irradiated specimens of reactor vessel materials during the plant lifetime. Further details are given in Section 4.1.6.

All pressure-containing components of the reactor coolant system are designed, fabricated, inspected, and tested in conformance with the applicable codes. Further details are given in Section 4.1.7.

4.1.3.5 Reactor Coolant Pressure Boundary Surveillance

Criterion: Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance of critical areas by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with current applicable codes shall be provided. (GDC 36)

The design of the reactor vessel and its arrangement in the system provides the capability for accessibility during service life to the entire internal surfaces of the vessel including the nozzle to reactor coolant piping welds and the top and bottom heads. The reactor arrangement within the containment provides sufficient space for inspection of the external surfaces of the reactor coolant piping, except for the area of pipe within the primary shielding concrete. Monitoring of the nil-ductility transition temperature properties of the core region plates, forgings, weldments, and associated heat-treated zones is performed in accordance with ASTM E185

(Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors). Samples of reactor vessel plate materials are retained and cataloged in case further engineering development shows the need for further testing.

The material properties surveillance program includes not only the conventional tensile and impact tests, but also fracture mechanics specimens. The fracture mechanics specimens are the wedge opening loading type specimens. The observed shifts in nil ductility transition temperature of the core region materials with irradiation are used to confirm the calculated limits to startup and shutdown transients.

To define permissible operating conditions below the design transition temperature, a pressure range is established, which is bounded by a lower limit for pump operation and an upper limit, which satisfies reactor vessel stress criteria. To allow for thermal stresses during heatup or cooldown of the reactor vessel, an equivalent pressure limit is defined to compensate for thermal stress as a function of rate of change of coolant temperature. Since the normal operating temperature of the reactor vessel is well above the maximum expected design transition temperature, brittle fracture during normal operation is not considered to be a credible mode of failure.

4.1.4 Design Characteristics

4.1.4.1 Design Pressure

The reactor coolant system design and operating pressure together with the safety, power relief, and pressurizer spray valves setpoints and the protection system setpoint pressures are listed in Table 4.1-1. The design pressure allows for operating transient pressure changes. The selected design margin considers core thermal lag, coolant transport times and pressure drops, instrumentation and control response characteristics, and system relief valve characteristics. The design pressures and data for the respective system components are listed in Tables 4.1-2 through 4.1-6. Table 4.1-7 gives the design pressure drop of the system components.

4.1.4.2 Design Temperature

The design temperature for each component is selected to be above the maximum coolant temperature in that component under all normal and anticipated transient load conditions. The design and operating temperatures of the respective system components are listed in Tables 4.1-2 through 4.1-6.

4.1.4.3 Seismic Loads

The seismic loading conditions are established by the "design earthquake" and "maximum potential earthquake." The former is selected to be typical of the largest probable ground motion based on the site seismic history. The latter is selected to be the largest potential ground motion at the site according to seismic and geological factors and their uncertainties.

For the design earthquake loading condition, the nuclear steam supply system is designed to be capable of continued safe operation. Therefore, for this loading condition critical structures and equipment needed for this purpose are required to operate within normal design limits. The seismic design for the maximum potential earthquake is intended to provide a margin in design that ensures capability to shut down and maintain the nuclear facility in a safe condition. In this case, it is only necessary to ensure that the reactor coolant system components do not lose

their capability to perform their safety function. This has come to be referred to as the "no-loss-of function" criteria and the loading condition as the "no-loss-of-function earthquake" loading condition.

The criteria adopted for allowable stresses and stress intensities in vessels and piping subjected to normal loads plus seismic loads are defined in Section 1.11. These criteria ensure the integrity of the reactor coolant system under seismic loading.

For the combination of normal and design earthquake loadings, the stresses in the support structures are kept within the limits of the applicable codes.

For the combination of normal and no-loss-of-function earthquake loadings, the deflections and stresses in the support structures are limited to values as necessary to ensure their integrity and to maintain supported equipment within their stress limits as stated in Table 1.11-2.

4.1.5 Cyclic Loads

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to reactor system temperature and pressure changes. These cyclic loads are introduced by normal unit load transients, reactor trip, and startup and shutdown operations. The number of thermal and loading cycles used for design purposes and the bases thereof are given in Table 4.1-8. There is a station program, which tracks these thermal and loading cycles. During unit startup and shutdown, the rates of temperature and pressure changes are limited as indicated in Section 4.4.1. The cycles are estimated for equipment design purposes (40-year life) and are not intended to be an accurate representation of actual transients or actual operating experience. For example, the number of cycles for plant heatup and cooldown at 100^{o}F per hr was selected as a conservative estimate based on an evaluation of the expected requirements. The resulting number, which averages five heatup and cooldown cycles per year, could be increased significantly; however, it is the intent to represent a conservative realistic number rather than the maximum allowed by the design.

Although loss-of-flow and loss-of-load transients are not included in the tabulation because the tabulation is only intended to represent normal design transients, the effects of these transients have been analytically evaluated and are included in the fatigue analysis for primary system components.

Over the range from 15-percent full power up to and including but not exceeding 100-percent of full power, for the purpose of cyclic load definition, the reactor coolant system and its components are designed to accommodate 10-percent of full power step changes in plant load and 5-percent of full power per minute ramp changes without reactor trip. The reactor coolant system will accept a complete loss of load from full power with reactor trip. In addition, the turbine bypass and steam dump system makes it possible to accept a step load decrease of 50-percent of full power without reactor trip. These transient capability definitions bracket the transient design bases used for the Regulating Systems as discussed in Section 7.3.

4.1.6 Service Life

The service life of reactor coolant system pressure components depends upon the end-of-life material radiation damage, unit operational thermal cycles, quality manufacturing standards, environmental protection, and adherence to established operating procedures.

The reactor vessel is the only component of the reactor coolant system that is exposed to a significant level of neutron irradiation and it is therefore the only component that is subject to material radiation damage effects.

The nil-ductility transition temperature shift of the vessel material and welds, due to radiation damage effects, is monitored by a radiation damage surveillance program, which conforms with ASTM E185 standards.

Reactor vessel design is based on the transition temperature method of evaluating the possibility of brittle fracture of the vessel material, as a result of operations such as leak testing and plant heatup and cooldown.

To establish the service life of the reactor coolant system components as required by the ASME (Section III) Boiler and Pressure Vessel Code for Class "A" vessels, the unit operating conditions have been established for the 40-year design life. These operating conditions include the cyclic application of pressure loadings and thermal transients.

The number of thermal and loading cycles used for design purposes are listed in Table 4.1-8.

4.1.7 Codes And Classifications

The quality assurance criteria specified below apply to all nuclear Class I piping and fittings.

All pressure-containing components of the reactor coolant system are designed, fabricated, inspected, and tested in conformance with the applicable codes listed in Table 4.1-9.

Shop and field fabrication requirements, documentation, and quality assurance examinations all comply with those found in USAS B31.7 for Class I nuclear piping.

Quality control techniques used in the fabrication of the reactor coolant system are equivalent to those used in the manufacture of the reactor vessel, which conforms to Section III of the ASME Boiler and Pressure Vessel Code.

The piping is designed to the USAS B31.1 (1955 and Summer 1973) Code for Power Piping using the allowable stresses found in the Nuclear Code Cases N-7 and N-10 for pipe and fittings, respectively.

The quality assurance requirements required by Westinghouse in the purchase and examination of the reactor coolant piping ensures that the quality level of a Westinghouse plant is comparable to that delineated by USAS B31.7, Class I, Code for Nuclear Piping.

1. All materials conform to ASTM specifications listed for B31.7 Class I, Nuclear Piping. In addition, all materials are certified, identified, and marked to facilitate traceability thus complying with the requirements of USAS B31.7, Class I, Code for Nuclear Piping.

2. Piping base materials are examined by quality assurance methods having acceptance criteria that meet the requirements set forth in USAS B31.7, Class I, Code for Nuclear Piping.

3. All welding procedures, welders, and welding operators are qualified to the requirements of ASME Section IX, Welding Qualifications, which is in compliance with the requirements of USAS B31.7, Class I, Code for Nuclear Piping.

4. All welds are examined by nondestructive testing methods and to the extent prescribed in USAS B31.7 for Class I nuclear piping.

5. All branch connection nozzle welds of nominal sizes of 3-in. and larger are 100-percent radiographed. This exceeds the requirements of USAS B31.7 for Class I piping, since it includes nominal sizes of 6-in. and larger for 100-percent radiography.

6. All finished welds are liquid penetrant examined on both the outside and inside (if accessible) surfaces as required by USAS B31.7, Class I. In addition, nozzle welds in nominal sizes 2-in. and smaller are progressively examined after each 0.25-in. increment of weld deposit in lieu of radiography.

7. Hydrostatic testing is performed on the erected and installed piping. This requirement is the same as in USAS B31.7, Class I.

Hence, the Westinghouse quality assurance requirements implemented in the procurement of Indian Point Unit 2 piping and fittings are equal to and in some instances exceed the requirements of USAS B31.7.

The reactor coolant system is classified as Class I for seismic design, requiring that there will be no loss of function of such equipment in the event of the assumed maximum potential ground acceleration acting in the horizontal and vertical directions simultaneously, when combined with the primary steady state stresses.

The design and stress criteria specified in USAS B31.7 are not directly comparable to that of USAS B31.1 (1955 and Summer 1973). The following describes how USAS B31.1 (1955 and Summer 1973) was used in the design of the primary coolant piping and the ASME B&PV Code Section III, Subsection NB, 1986 Edition for the presurizer surge line including the effects of Thermal Stratification on Indian Point Unit 2. A thermal expansion flexibility stress analysis was performed on the main primary coolant piping and pressurizer surge line (including the effects of Thermal Stratification) in accordance with the criteria set forth in USAS B31.1 (1955 and Summer 1973) for the reactor coolant piping and the ASME B&PV Code Section III 1986 Edition for the pressurizer surge line including the effects of Thermal Stratification. For the reactor coolant piping the analysis was performed to ensure that the stress range is within the limits prescribed in B31.1. As per the requirements of USAS B31.1, no fatigue analysis is required and hence, no fatigue analysis of the reactor coolant loop piping is performed. For the pressurizer surge line including the effects of Thermal Stratification, the analysis was performed to ensure that the stress range and number of thermal cycles (usage factor) are safely within the limits prescribed in ASME B&PV Code Section III, Subsection NB, 1986 Edition. In addition, seismic analysis were performed on the composite piping, which included the combined stress effects of all the sustained (pressure and weight) loadings plus seismic vertical / horizontal loading components. The resultant reactions of the piping due to the separate and combined effects of thermal, sustained, and seismic loadings were factored into the piping as interconnected. In turn, the equipment supporting structures were checked for adequate design including the added effects of these same loadings. Thus the total design analyses including

pipe, equipment, and structures considered the effects of thermal expansion, sustained, and seismic loadings.

Thermally induced stresses arising from temperature gradients are limited to a safe and low order of magnitude in assigning a maximum permissible time rate of temperature change on plant heatup, cooldown, and incremental loadings in the plant operation procedure.

An added margin of conservatism is obtained through the use of thermal sleeves in nozzles wherein a cold fluid is introduced into a pipe conveying a significantly hotter fluid or vice versa. Typical examples are the charging line, pressurizer surge line, and residual heat return nozzle connections to the primary coolant loop piping. The thermal sleeve is no longer in place on the 10" SI line to the 23 Cold Leg. A detailed analysis demonstrated that the fatigue usage factor and stresses for the nozzle in line 353 still meet the requirements of ASME Section III of the Boiler and Pressure Vessel Code for continued operation through the life of the plant with the thermal sleeve not in place.

REFERENCES FOR SECTION 4.1

1. Letter from Stephen B. Bram, Con Edison, to Document Control Desk, NRC, Subject: Leak-Before-Break, dated May 23, 1988.

2. Letter from Stephen B. Bram, Con Edison, to Document Control Desk, NRC, Subject: Leak-Before-Break (LBB) Submittal (TAC 68318), dated November 18, 1988.

3. Letter from Stephen B. Bram, Con Edison, to Document Control Desk, Subject: Leak-Before-Break (LBB) Submittal (TAC 68318), dated January 12, 1989.

4. Letter from Donald S. Brinkman, NRC, to Stephen B. Bram, Con Edison, Subject: Safety Evaluation Report on Elimination of Dynamic Effect of Postulated Primary Loop Pipe Ruptures from Design Basis for Indian Point Unit 2 (TAC No. 68318), dated February 23, 1989.

5. WCAP-12187 – Consolidated Edison Company of New York, Inc., Indian Point Unit 2, NSSS Stretch Rating – 3083.4 MWT, Engineering Report, Volume II. - Westinghouse Proprietary Class 2

TABLE 4.1-1
Reactor Coolant System Pressure Settings

Category	Pressure (psig)
Design pressure	2485
Operating pressure (at pressurizer)	2235_1
Safety valves	2485_1
Power relief valves	2335_1
Pressurizer spray valves (open)	2260_1
High pressure trip	$\leq 2363_1$
High pressure alarm	$2300/2335_{1,2}$
Low pressure trip	$\geq 1928_1$
Low pressure alarm	2185_1
Hydrostatic test pressure	3110

Notes:
1. Nominal values
2. The fixed high alarm PC-456F is a redundant alarm, and will not annunciate unless there is a failure of the 2300 psig alarm/control circuit.

TABLE 4.1-2
Reactor Vessel Design Data

Design/operating pressure, psig	2485/2235
Hydrostatic test pressure, psig	3110
Design temperature, °F	650
Overall height of vessel and closure head, ft-in. (bottom head OD to top of control rod mechanism housing)	43-9 11/16
Water volume, (with core and internals in place), ft^3	4647
Thickness of insulation, min., in.	3
Number of reactor closure head studs	54
Diameter of reactor closure head studs, in.	7
ID of flange, in.	167 1/16
OD of flange, in.	205
ID at shell, in.	173
Inlet nozzle ID, in.	27 1/2
Outlet nozzle ID, in.	29
Clad thickness, min., in.	5/32
Lower head thickness, min., in.	5 5/16
Vessel belt-line thickness, min., in.	8 5/8
Closure head thickness, in.	7
Reactor coolant inlet temperature, °F	514.3[1]
Reactor coolant outlet temperature, °F	605.8[1]
Reactor coolant flow, lb/hr	1.268×10^8

Notes:
1. Reactor Coolant inlet temperature is for the low T_{avg} case and Reactor Coolant outlet temperature is for the high T_{avg} case.

TABLE 4.1-3
Pressurizer and Pressurizer Relief Tank Design Data

Pressurizer

Design/operating pressure, psig	2485/2235
Hydrostatic test pressure (cold), psig	3110
Design/operating temperature, °F	680/653
Water volume, full power, ft^3	1080_1
Steam volume, full power, ft^3	720_1
Surge line nozzle diameter, in./pipe schedule	14/Sch 140
Shell ID, in./calculated minimum shell thickness, in.	84/4.1
Minimum clad thickness, in.	0.188
Electric heaters capacity, kW	1800
Heatup rate of pressurizer using heaters only, °F/hr	55 (approximately)
Power relief valves	
Number	2
Set pressure (open), psig	2335
Capacity, lb/hr saturated steam per valve	179,000
Safety valves	
Number	3
Set pressure, $psig_2$	2485
Capacity, lb/hr saturated steam per valve	408,000

Pressurizer Relief Tank

Design pressure, psig	100
Rupture disc release pressure, psig	100
Design temperature, °F	340
Normal water temperature, °F	Containment ambient
Total volume, ft^3	1800
Rupture disc relief capacity, lb/hr	1.224×10^6

Notes:
1. Present operation is at a T_{avg} of 562°F. In the safety analysis discussed in section 14.1, a reduced flow is assumed to account for a postulated 25% steam generator tube plugging. Actual values will depend on T_{avg} and actual percentage of tube plugging.
2. Allowance for error is specified in the Technical Specifications.

TABLE 4.1-4
Steam Generator Design Data

Number of steam generators	4	
Design pressure, reactor coolant/steam, psig	2485/1085	
Reactor coolant hydrostatic test pressure (tube side-cold), psig	3107	
Design temperature, reactor coolant/steam, °F	650/556	
Reactor coolant flow, Thermal Design, gpm/loop	80,700	
Total heat transfer surface area, ft^2	43,467	
Heat transferred, Btu/hour	2755×10^6	
Steam conditions at full load, outlet nozzle:	Low T_{avg} *	High T_{avg} *
Steam flow, lb/hr	3.50×10^6	3.51×10^6
Steam temperature, °F	488.0	513.3
Steam pressure, psia	610.1	766.3
Feedwater temperature, °F	436.2	436.2
Overall height, ft-in.	63-1.625	
Shell OD, upper/lower, in.	166/127.0	
Shell thickness, upper/lower, in.	3.5/2.63	
Number of U-tubes	3214	
U-tube diameter, in.	0.875	
Tube wall thickness, (average), in.	0.050	
Number of manways/ID, in.	4/16	
Number of handholes/ID, in.	6/6	
Number of Inspection Openings/ID, in.	1/3	

	3230 MWt Low T_{avg}/High T_{avg} *	Zero Power
Reactor coolant water volume (unplugged), ft^3	924	924
Primary side fluid heat content, Btu	23.67×10^6 / 24.11×10^6	23.630×10^6
Secondary side water volume, ft^3	1493/1599	2778.5
Secondary side steam volume, ft^3	32434/3128	1949
Secondary side fluid heat content, Btu	39.96×10^6 / 75.31×10^6	

Note:
*Refers to low (548.9 deg F) and high (571.9 deg F) T_{avg} and 0% tube plugging cases for design.

TABLE 4.1-5
Reactor Coolant Pumps Design Data

Number of pumps	4
Design pressure/operating pressure, psig	2485/2235
Hydrostatic test pressure (cold), psig	3110
Design temperature (casing), °F	650
RPM at nameplate rating	1189
Suction temperature, °F	555 [1]
Net positive suction head, ft	170 [1]
Developed head, ft	272 [1]
Capacity, gpm	89,700 [1]
Seal water injection, gpm	8
Seal water return, gpm	3
Pump discharge nozzle ID, in.	27 1/2
Pump suction nozzle ID, in.	31
Overall unit height, ft	28.38
Water volume, ft^3	192
Pump-motor moment of inertia, lb/ft^2	82,000
Motor Data:	
Type	AC induction, single speed, air cooled
Voltage	6600
Insulation class	B thermoplastic epoxy
Phase	3
Frequency, cps	60
Starting Current, amp	2950
Input (hot reactor coolant), kW	4221 [1]
Input (cold reactor coolant), kW	5673 [1]
Power, HP (nameplate)	6000

Note:

1. These values represent the pump hydraulic design point. Actual heads, flows, temperatures, currents, and powers are dependant upon system parameters such as reactor internals changes, percentage of steam generator tube plugging, and plant operating T_{avg}. For use in analyses or evaluations, values reflecting the current conditions should be obtained.

TABLE 4.1-6
Reactor Coolant Piping Design Data

Reactor inlet piping ID, in.	27 1/2
Reactor inlet piping nominal thickness, in.	2.375
Reactor outlet piping ID, in.	29
Reactor outlet piping nominal thickness, in.	2.50
Coolant pump suction piping ID, in.	31
Coolant pump suction piping nominal thickness, in.	2.656
Pressurizer surge line piping ID, in.	11.5
Pressurizer surge line piping nominal thickness, in.	1.25
Design/operating pressure, psig	2485/2235
Hydrostatic test pressure, (cold) psig	3110
Design temperature, °F	650
Design temperature, (pressurizer surge line) °F	680
Water volume, (all 4 loops including surge line) ft^3	1156

TABLE 4.1-7
Reactor Coolant System Design Pressure Drop

	Pressure Drop (psi)
Across pump discharge leg	1.2
Across vessel, including nozzles	51.5
Across hot leg	1.1
Across steam generator	31.8
Across pump suction leg	2.8
Total pressure drop	88.4

Notes:
1) DP's based on best estimate flow conditions.

TABLE 4.1-8
Thermal and Loading Cycles

	Transient Condition	Design Cycles[1]
1.	Plant heatup at 100°F per hr	200 (5/yr[2])
2.	Plant cooldown at 100°F per hr	200 (5/yr)
3.	Plant loading at 5-percent of full power per min	14,500 (1/day)
4.	Plant unloading at 5-percent of full power per min	14,500 (1/day)
5.	Step load increase of 10-percent of full power (but not to exceed full power)	2000 (1/wk)
6.	Step load decrease of 10-percent of full power	2000 (1/wk)
7.	Step load decrease of 50-percent of full power	200 (5/yr)
8.	Reactor trip	400 (10/yr)
9.	Hydrostatic test at 3110 psig pressure	5 (preoperational)
10.	Hydrostatic test at 2485 psig pressure and 400°F temperature	5 (postoperational)
11.	Steady state fluctuations — the reactor coolant average temperature for purposes of design is assumed to increase and decrease a maximum of 6°F in one minute. The corresponding reactor coolant pressure variation is less than 100 psig. It is assumed that an infinite number of such fluctuations will occur.	

Notes:

1. Estimated for equipment design purposes (40-yr life) and not intended to be an accurate representation of actual transients, or to reflect actual operating experience.
2. This transient includes pressurizing to 2235 psig.

TABLE 4.1-9
Reactor Coolant System - Design Code Requirements

COMPONENT	CODE	CODE EDITION	APPLICABLE ADDENDA
Reactor vessel	ASME III[1] Class A	1965	Summer 1965 and Code Cases 1332, 1335, 1339, 1359
Control rod drive mechanism	ASME III[1] Class A	1965	Summer 1966
Steam generators			
Tube side	ASME III[1] Class A	1965	Summer 1966
Shell side[4]	ASME III[1] Class C	1965	Summer 1966
Reactor coolant pump volute[5]	ASME III[1] Class A	1965	Winter 1965[2]
Pressurizer	ASME III[1] Class A	1965	Summer 1966
Pressurizer relief tank	ASME III[1] Class C	1964	Winter 1965
Pressurizer safety valves:			
Old Buy	ASME III[1]	1971	Winter 1972
New Buy		1974	Summer 1975
Reactor coolant piping	USAS B31.1[3]	1955	
System valves, fittings, piping	USAS B31.1[3]	1955	

Notes:

1. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

2. Not stamped, but built in accordance with this edition and addenda.

3. USAS B31.1 Code for pressure piping.

4. The shell side of the generator conforms to requirements for Class A vessels and is so stamped as permitted under the rules of Section III.

5. The reactor pump, though not a coded vessel, was designed to Section III of the ASME Boiler and Pressure Vessel Code.

4.2 SYSTEM DESIGN AND OPERATION

4.2.1 General Description

The reactor coolant system consists of four similar heat transfer loops connected in parallel to the reactor vessel. Each loop contains a steam generator, a pump, loop piping, and instrumentation. The pressurizer surge line is connected to one of the loops. Auxiliary system piping connections into the reactor coolant piping are provided as necessary. A flow diagram of the system is shown in Plant Drawing 9321-2738 [Formerly UFSAR Figure 4.2-1], and a schematic flow diagram in Figure 4.2-2. The total design volume of the reactor coolant system, at rated operating conditions, is approximately 12,250-ft^3. The nominal liquid volume of the reactor coolant system, at rated operating conditions and with 0% Steam Generator tube plugging, is 11,350 cubic feet.

The containment boundary shown on the flow diagram indicates those major components, which are to be located inside the containment. The intersection of a process line with this boundary indicates a functional penetration.

Reactor coolant system design data are listed in Tables 4.1-2 through 4.1-6.

Pressure in the system is controlled by the pressurizer, where water and steam pressure is maintained through the use of electrical heaters and sprays. Steam can either be formed by the heaters, or condensed by a pressurizer spray to minimize pressure variations due to contraction and expansion of the coolant. Instrumentation used in the pressure control system is described in Chapter 7. Spring-loaded steam safety valves and power-operated relief valves are connected to the pressurizer and discharge to the pressurizer relief tank, where the discharged steam is condensed and cooled by mixing with water.

4.2.2 Components

4.2.2.1 Reactor Vessel

The reactor vessel is cylindrical with a hemispherical bottom and a flanged and gasketed removable upper head. The vessel is designed in accordance with Section III (Nuclear Vessels) of ASME Boiler and Pressure Vessel Code. Figure 4.2-3 is a schematic of the reactor vessel. The materials of construction of the reactor vessel are given in Table 4.2-1.

Coolant enters the reactor vessel through inlet nozzles in a plane just below the vessel flange and above the core. The coolant flows downward through the annular space between the vessel wall and the core barrel into a plenum at the bottom of the vessel where it reverses direction. Approximately 95-percent of the total coolant flow is effective for heat removal from the core. The remainder of the flow includes the flow through the rod cluster control guide thimbles, the leakage across the outlet nozzles, and the flow deflected into the head of the vessel for cooling the upper flange. All the coolant is united and mixed in the upper plenum and the mixed coolant stream then flows out of the vessel through exit nozzles located on the same plane as the inlet nozzles.

A one-piece thermal shield, concentric with the reactor core, is located between the core barrel and the reactor vessel. It is attached to the core barrel. The shield, which is cooled by the coolant on its downward pass, protects the vessel by attenuating much of the gamma radiation

and some of the fast neutrons, which escape from the core. This shield minimizes thermal stresses in the vessel that result from heat generated by the absorption of gamma energy. This protection is further described in Section 3.2.3.

Fifty-eight core instrumentation nozzles are located on the lower head.

The reactor closure head and the reactor vessel flange are joined by fifty-four 7-in. diameter studs. Two metallic 0-rings seal the reactor vessel when the reactor closure head is bolted in place. A leakoff connection is provided between the two 0-rings to monitor leakage across the inner 0-ring. A leakoff connection is also provided beyond the outer 0-ring seal.

The vessel is insulated with metallic reflective-type insulation supported from the nozzles. Insulation panels are provided for the reactor closure head, which are supported on the refueling seal ledge and vent shroud support rings.

The reactor vessel internals are designed to direct the coolant flow, support the reactor core, and guide the control rods in the withdrawn position. The reactor vessel contains the core support assembly, upper plenum assembly, fuel assemblies, control cluster assemblies, surveillance specimens, and incore instrumentation.

Surveillance specimens made from reactor vessel steel are located between the reactor vessel wall and the thermal shield. These specimens will be examined at selected intervals to evaluate reactor vessel material nil-ductility transition temperature changes as described in Section 4.5.2. The factor by which the maximum specimen exposure exceeds that at the vessel wall (at the location of maximum vessel wall exposure) has a maximum value of 3.5. Four of the eight irradiation specimens will lead the vessel wall maximum exposure by this factor.

Ring forgings have been used for closure flanges; no other forgings have been used in the reactor vessel shell sections. The eight primary inlet and outlet nozzles have been provided with nozzle safe ends (forgings). These safe ends have been overlaid in the field with stainless steel weld metal. The Charpy V-notch and drop weight tests for the reactor vessel plates and forgings are discussed in Section 4.2.5.

The reactor internals are described in detail in Section 3.2.3 and the general arrangement of the reactor vessel and internals is shown in Figure 3.2-47.

Reactor vessel design data are listed in Table 4.1-2.

4.2.2.2 Pressurizer

The general arrangement of the pressurizer is shown in Figure 4.2-4 and the design characteristics are listed in Table 4.1-3.

The pressurizer maintains the required reactor coolant pressure during steady-state operation, limits the pressure changes caused by coolant thermal expansion and contraction during normal load transients, and prevents the pressure in the reactor coolant system from exceeding the design pressure.

The pressurizer contains replaceable direct immersion heaters, multiple safety and relief valves, a spray nozzle and interconnecting piping, valves, and instrumentation. The electric heaters located in the lower section of the vessel regulate the reactor coolant system pressure by

keeping the water and steam in the pressurizer at saturation temperature. The heaters are capable of raising the temperature of the pressurizer and contents at approximately 55°F/hr during startup of the reactor.

The pressurizer is designed to accommodate positive and negative surges caused by load transients. The surge line, which is attached to the bottom of the pressurizer, connects the pressurizer to a hot leg of a reactor coolant loop. During a positive surge caused by a decrease in plant load, the spray system, which is fed from the cold leg of a coolant loop, condenses steam in the vessel to prevent the pressurizer pressure from reaching the setpoint of the power operated relief valves. The spray valves on the pressurizer are power operated. In addition, the spray valves can be operated manually by a valve controller in the control room. A small continuous spray flow is provided to ensure that the pressurizer liquid is homogeneous with the coolant and to prevent excess cooling of the spray piping.

During a negative pressure surge caused by an increase in plant load, flashing of water to steam and generation of steam by automatic actuation of the heaters keep the pressure above the minimum allowable limit. Heaters are also energized on high water level during positive surges to heat the subcooled surge water entering the pressurizer from the reactor coolant loop.

The pressurizer is constructed of low alloy steel with internal surfaces clad with austenitic stainless steel. The heaters are sheathed in austenitic stainless steel.

The pressurizer vessel surge nozzle is protected from thermal shock by a thermal sleeve. A thermal sleeve also protects the pressurizer spray nozzle connection.

4.2.2.3 Steam Generators

Each loop contains a vertical shell and U-tube steam generator. A steam generator of this type is shown in Figure 4.2-5. Principal design parameters are listed in Table 4.1-4. The steam generators are designed and manufactured in accordance with Section III (Nuclear Vessels) of the ASME Boiler and Pressure Vessel Code. The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.

Reactor coolant enters the inlet side of the channel head at the bottom of the steam generator through the inlet nozzle, flows through the U-tubes to an outlet channel, and leaves the generator through another bottom nozzle. The inlet and outlet channels are separated by a partition. Manways are provided to permit access to the U-tubes and the moisture-separating equipment.

Feedwater to the steam generator enters just above the top of the U-tubes through a feedwater ring. The water flows downward through an annulus between the tube wrapper and the shell and then upward through the tube bundle where part of it is converted to steam. Certain plant operating conditions affecting the steam generator can result in the steam generator water level dropping below the feedwater sparging ring. As a result of these conditions and the waterhammer that can occur in the feedwater system, "J" tubes are installed on the feedwater sparging rings inside the steam generators. These "J" tubes preclude the rapid draining of the feedwater sparging rings and prevent steam from entering these rings even if they are uncovered. In the very remote event that the feedwater system would experience another large pressure wave, additional pipe restraints were installed in 1974 along the feedwater pipe. This modification is intended to preclude the rebound-type failure of the feedwater line at the containment penetration supports. All modifications that were made and the test program that

was performed as a result of these conditions were accomplished in accordance with the Quality Assurance Program for operating nuclear plants that was currently in effect.

The steam-water mixture from the tube bundle passes through a steam swirl vane assembly, which imparts a centrifugal motion to the mixture and separates the water particles from the steam. The water spills over the edge of the swirl vane housing and combines with the feedwater for another pass through the tube bundle. The steam rises through additional separators, which further reduce the moisture content of the steam.

A steam-generator blowdown system exists to perform several functions. Primarily it is used in maintaining the secondary side water chemistry of the steam generators within specifications. It also provides water samples from the secondary side of the steam generator as well as a means of draining the shell sides for inspection and/or maintenance.

The steam generator is constructed primarily of low alloy steel. The heat transfer tubes are Inconel. The tubes undergo thermal treatment following tube-forming operations. The interior surfaces of the channel heads and nozzles are clad with austenitic stainless steel and the side of the tube sheet in contact with the reactor coolant is clad with Inconel. The tube-to-tube sheet joint is welded. The primary nozzles are provided with safe ends with weld metal overlay.

Tubes are examined and defective tubes are repaired, plugged or sleeved as required by the technical specifications. The upper limit for tube plugging is 20-percent.

Nozzle dam retention rings are permanently welded to the channel head cladding and provide a means of attachment for the temporary installation of nozzle dams during refueling and/or maintenance outages.

The inspection ports and handholes are on the secondary side of the steam generators and any possible leakage (the only possible failure) would be inside containment. The possibility of secondary loss of water has been evaluated in Section 14.1.9, Loss of Normal Feedwater, and Section 14.2.5, Rupture of a Steam Pipe. These sections show that the types of failure possible due to secondary leakage or loss of water have already been analyzed.

4.2.2.4 Reactor Coolant Pumps

Each reactor coolant loop contains a vertical, single-stage centrifugal pump that employs a controlled leakage seal assembly. A view of a controlled leakage pump is shown in Figure 4.2-6 and the principal design parameters for the pumps are listed in Table 4.1-5. The reactor coolant pump estimated performance and net positive suction head characteristics are shown in Figure 4.2-7. The performance characteristic is common to all of the higher specific speed centrifugal pumps and the "knee" at about 45-percent design flow introduces no operational restrictions, because the pumps operate at full speed.

During normal operation, the reactor coolant pumps are supplied from the unit auxiliary bus and are therefore tied to the turbine-generator frequency (speed). On occurrence of unit turbine trip, the pump electrical buses are transferred from the unit auxiliary transformer to the station auxiliary transformer with an intentional delay of 30 sec. Further details are given in Section 14.1.8.3.

On most electrical events, which cause the turbine to be tripped, the reactor coolant pump buses are transferred to offsite power and the unit is tripped simultaneously and the pumps will

therefore not exceed their normal running speed. If for some unlikely reason the only plant trip is a turbine over-speed trip an over-frequency trip relay circuit is provided that will trip the turbine-generator. This trip circuit first locks out the 6.9 kV dead bus transfer at 62.2 ±0.1 Hz (1866 ± 3 rpm) and then trips the main generator at 62.5 ±0.1 Hz (1875 ± 3 rpm). Termination of power to the in-house 6.9 kV buses 1-4 limits the reactor coolant pumps overspeed to maintain the RCS flow condition below the design limit per section 4.2.2.5.4.

Reactor coolant is pumped by the impeller attached to the bottom of the rotor shaft. The coolant is drawn up through the impeller, discharged through passages in the diffuser, and exits through a discharge nozzle in the side of the casing. The motor-impeller can be removed from the casing of the piping. All parts of the pumps in contact with the reactor coolant are austenitic stainless steel or equivalent corrosion resistant materials.

The pump employs a controlled leakage seal assembly to restrict leakage along the pump shaft, a second seal that directs the controlled leakage out of the pump, and a third seal that minimizes the leakage of water and vapor from the pump into the containment atmosphere.

A portion of the high-pressure water flow from the charging pumps is injected into the reactor coolant pump between the impeller and the controlled leakage seal. Part of the flow enters the reactor coolant system through a labyrinth seal in the lower pump shaft to serve as a buffer to keep reactor coolant from entering the upper portion of the pump. The remainder of the injection water flows along the drive shaft, through the controlled leakage seal, and finally out of the pump. A very small amount that leaks through the second seal is also collected and removed from the pump.

Component cooling water is supplied to the motor bearing oil coolers of the reactor coolant pumps. The component cooling water system also provides cooling flow to the thermal barrier heat exchanger of the reactor coolant pumps to minimize heat transfer from the high-temperature primary coolant to the seal area environment, to cool primary system water that could leak through the thermal barrier labyrinth seals, and to provide adequate seal cooling in the event that seal injection flow was lost.

In the event of loss of offsite power, the reactor coolant pump motor is deenergized and both cooling water supplies (seal injection and component cooling flow) are terminated; however, the plant diesel generators are immediately started and the component cooling water pumps are automatically loaded (in sequence) onto the emergency buses and started (no operator action required). Once the automatic loading of the emergency buses has been completed, the operator has the option of manually loading a charging pump onto one of the diesels and reestablishing normal seal injection flow. The squirrel cage induction motor driving the pump is air cooled and has oil lubricated thrust and radial bearings. A water lubricated bearing provides radial support for the pump shaft.

An extensive test program was conducted for several years to develop the controlled leakage shaft seal for pressurized-water reactor applications. Long-term tests were conducted on less than full-scale prototype seals as well as on full-size seals. Operating experience with other large size, controlled leakage shaft seal pumps has also been available.

The reactor coolant pump motor bearings are of conventional design, the radial bearings are the segmented pad type, and the thrust bearings are tilting pad Kingsbury bearings. All are oil lubricated - the lower radial bearing and thrust bearings are submerged in oil, and the upper radial bearing is oil fed from an impeller integral with the thrust runner. Both high and low oil

levels would signal an alarm in the control room to alert the operator to possible pending bearing problems. Each motor bearing contains embedded temperature detectors, and so initiation of failure, separate from or in combination with a loss of oil, would be indicated and alarmed in the control room as a high bearing temperature. This would require a reactor trip followed by pump shutdown. Even if these indications were ignored, and the bearing proceeded to failure, the low melting point Babbitt metal on the pad surfaces would ensure that no sudden seizure of the bearing would occur. In this event the motor would continue to drive, as it has sufficient reserve capacity to operate even under such conditions. However, it would demand excessive currents and at some stage would be shut down because of high current demand.

It may be hypothesized that the pump impeller might severely rub on a stationary member and then seize. Analysis has shown that under such conditions, assuming instantaneous seizure of the impeller, the pump shaft would fail in torsion just below the coupling to the motor. This would constitute a loss of coolant flow in the one loop, the effect of which is analyzed in Section 14.1.6.

Following the seizure, the motor would continue to run without any overspeed and the flywheel would maintain its integrity, as it would still be supported on a shaft with two bearings.

There are no other credible sources of shaft seizure other than impeller rubs. Any seizure of the pump bearing would be precluded by shearing of the graphitar in the bearing. Any seizure in the seals would result in a shearing of the anti-rotation pin in the seal ring. The motor has adequate power to continue pump operation even after the above occurrences. Indications of pump malfunction in these conditions would be initially by high temperature signals from the bearing water temperature detector, and excessive No. 1 seal leakoff indications, respectively. Following these signals, pump vibration levels would be checked. These would show excessive levels, indicating some mechanical trouble. Again, the pump would be shut down for investigation.

The design specifications for the reactor coolant pumps include as a design condition that the pumps are designed to withstand seismic load equivalent to 0.28 g in the vertical direction and 0.40 g in the horizontal direction and the seismic loads shall be considered acting simultaneously. Besides examining the externally produced loads from the nozzles and support lugs, an analysis was made of the effect of gyroscopic reaction on the flywheel and bearings and in the shaft due to rotational movements of the pump about a horizontal axis during the maximum seismic disturbance. The pump would continue to run unaffected by such conditions. In no case does any bearing stress in the pump or motor exceed or even approach a value, which the bearing could not carry.

The design requirements of the bearings are primarily aimed at ensuring a long life with negligible wear so as to give accurate alignment and smooth operation over long periods of time. To this end, the surface bearing stresses are held at a very low value, and even under the most severe seismic transients or other accidents, do not begin to approach loads, which cannot be adequately carried for short periods of time.

Because there are no established criteria for short-time stress-related failures in such bearings, it is not possible to make a meaningful quantification of such parameters as margins-to-failure, safety factors, etc. A qualitative analysis of the bearing design, embodying such considerations, gives assurance of the adequacy of the bearing to operate without failure.

As is generally the case with machines of this size, the shaft dimensions are predicated on avoidance of shaft critical speed conditions, rather than actual levels of stress. There are many machines as large as, and larger than these, that are designed to run at speeds in excess of first shaft critical. However, it is considered desirable in a superior product to operate below first critical speed; the reactor coolant pumps are designed in accordance with this philosophy. This results in a shaft design, which even under the severest postulated transient, gives very low values of actual stress. While it would be possible to present quantitative data of imposed operational stress relative to maximum tolerable levels, if the mode of postulated failure were clearly defined, such figures would have little significance in a meaningful assessment of the adequacy of the shaft to maintain its integrity under operational transients. However, a qualitative assessment of such factors gives assurance of the conservative stress levels experienced during these transients.

So in each of these cases, where it is the functional requirements of the component that control its dimensions, it can be seen that if these are met, the stress-related failure cases are more than adequately satisfied.

It is thus considered to be out of the bounds of reasonable credibility that any bearing or shaft failure could occur that would endanger the integrity of the pump.

4.2.2.5 Reactor Coolant Pump Flywheel Integrity

The reactor coolant pump flywheels were fabricated from two rolled, vacuum-degassed, ASTM A-533 Grade B Class 1 steel plates. The plates are bolted together with bolts aligned perpendicular to the plane of the plates. Thus the bolts carry no stress during operation.

The flywheel blanks were flame-cut from the plates, with allowance for exclusion of flame-affected metal. They were then machined to the specified dimensions and the bolt holes were drilled.

Two plates were then bolted together, the finished flywheel attached to the motor shaft, and the whole unit balanced to yield vibration levels at operating speed less than 0.001-in. double amplitude. The reactor coolant pump flywheel is shown in Figure 4.2-8.

A nil-ductility transition temperature less than +10^{o}F was specified. A minimum of three Charpy tests, parallel and normal to the rolling direction, were made from each plate to determine that each blank satisfied design requirements.

The finished flywheels were subjected to 100-percent volumetric ultrasonic inspection.

The finished machined bores were also subjected to magnetic particle or liquid penetrant examination.

These design-fabrication techniques yield flywheels with primary stress at operating speed (shown in Figure 4.2-9) less than 50-percent of the minimum specified material yield strength at room temperature (100 to 150^{o}F).

A fracture mechanics evaluation was made on the reactor coolant pump flywheel. This evaluation considered the following assumptions:

1. Maximum tangential stress at an assumed overspeed of 125-percent compared with a maximum expected overspeed of 109-percent.

2. A through crack through the thickness of the flywheel at the bore.

3. 400 cycles of startup operation in 40 years.

   Using critical stress intensity factors and crack growth data attained on flywheel material, the critical crack size for failure was greater than 17-in. radially and the crack growth data was 0.030-in. to 0.060-in. per 1000 cycles.

   Periodic nondestructive examinations of the reactor coolant pump flywheels are included in the inservice inspection and testing program discussed in Chapter 1.

4. Normal operating speed of the flywheel (pump).

   The primary coolant pumps run at 1189 rpm and may operate briefly at overspeeds up to 109-percent (1295 rpm) during loss of offsite load. For conservatism, however, 125-percent of operating speed was selected as the design speed for the primary coolant pumps. For the overspeed condition, which would not persist for more than 30 sec, pump operating temperature would remain at about the design value. However, the limiting condition for the RCS system components is the effect the excess RCS flow has on the reactor internals. This limit is a statistically derived RCS flow value of 115.8% at a confidence level of 95%.

5. Bursting speed of the flywheel.

   Bursting speed of the flywheels was calculated on the basis of Robinson's results (Reference 1) to be 3900 rpm, more than three times the operating speed. This is confirmed using Griffith-Irwin theory as detailed in Reference 2.

4.2.2.6 Pressurizer Relief Tank

Principal design parameters of the pressurizer relief tank are given in Table 4.1-3. The tank is shown on Figure 4.2-10.

Steam and water discharge from the power relief and safety valves pass to the pressurizer relief tank, which is partially filled with water. The tank normally contains water in a predominantly nitrogen atmosphere. Steam is discharged under the water level to condense and cool by mixing with the water. The tank is equipped with a spray and a drain to the waste disposal system, which are operated to cool the tank following a discharge.

The tank size is based on the requirement to condense and cool a discharge equivalent to 110-percent of full power pressurizer steam volume.

The tank is protected against a discharge exceeding the design value by two rupture disks that discharge into the reactor containment. The rupture disks on the relief tank have a combined relief capacity equal to the combined capacity of the pressurizer safety valves. The tank design pressure (and the rupture disk's bursting pressure) is twice the calculated pressure resulting from the maximum safety valve discharge described above. This margin is to prevent

deformation of the disk. The tank and rupture disk holder are also designed for full vacuum to prevent tank collapse if the tank contents cool without nitrogen being added.

The discharge piping from the safety and relief valves to the relief tank is sufficiently large to prevent backpressure at the safety valves from exceeding 20-percent of the setpoint pressure at full flow.

The pressurizer relief tank, by means of its connection to the waste disposal system, provides a means for removing any noncondensable gasses from the reactor coolant system, which might collect in the pressurizer vessel.

The tank is constructed of carbon steel with a corrosion-resistant coating on the internal surface.

4.2.2.7 Piping

A schematic of the reactor coolant piping is shown on Figure 4.2-2. The general arrangement of the loop piping is shown on Plant Drawings 9321-2502, 9321-2506, 9321-2508 [Formerly UFSAR Figures 5.1-3, 5.1-5, and 5.1-7]. Piping design data are presented in Table 4.1-6.

The austenitic stainless steel reactor coolant piping and fittings that make up the loops are 29-in. ID in the hot legs, 27.5-in. ID in the cold legs, and 31-in. ID between the steam generator outlet and reactor coolant pump suction. The pressurizer relief line, which connects the outlets of the pressurizer safety and relief valves to the inlet nozzle flange on the pressurizer relief tank, is constructed of carbon steel.

Smaller piping, including the pressurizer surge and spray lines, drains, and connections to other systems are austenitic stainless steel. All piping connections are welded except for flanged connections at the pressurizer relief tank and at the safety valves, and the vacuum fill connection closure.

In response to NRC Bulletin 88-11, thermal stratification effects on the pressurizer surge line have been evaluated for the design life of the plant[18]. The stress and fatigue analyses results are within the ASME Code allowables for the surge line.

Thermal sleeves are installed at the following locations where high thermal stresses could otherwise develop due to rapid changes in fluid temperature during normal operational transients:

1. Return lines from the residual heat removal loop (safety injection lines) See Section 4.1.7.

2. Both ends of the pressurizer surge line.

3. Pressurizer spray line connection to the pressurizer.

4. Charging lines and auxiliary charging line connections.

4.2.2.8 Valves

All valve surfaces in contact with reactor coolant are austenitic stainless steel or equivalent corrosion-resistant materials. Connections to stainless steel piping are welded. Valves that

perform a modulating function are equipped with either two sets of packing and an intermediate leakoff connection or have been designed with live-loaded packing which will either control of mitigate the potential for valve stem leakage due to modulating service.

4.2.2.9 Component Supports

The support structures for the reactor coolant components are described in Appendix 4B and Chapter 5.

4.2.3 Pressure-Relieving Devices

The reactor coolant system is protected against overpressure by control and protective circuits such as the high-pressure trip and by code relief valves connected to the top head of the pressurizer. The relief valves discharge into the pressurizer relief tank, which condenses and collects the valve effluent. The schematic arrangement of the relief devices is shown in Figure 4.2-1; the valve design parameters are given in Table 4.1-3. Valve sizes are determined as indicated in Section 4.3.4.

Power-operated relief valves and code safety valves are provided to protect against pressure that is beyond the pressure limiting capacity of the pressurizer spray. Acoustic sensors installed on the code safety valve discharge lines provide indication in the control room of the "flow" or "no flow" condition of the safety valves. Direct valve position indication is also provided for the power-operated relief valves.

The pressurizer relief tank is protected against a steam discharge exceeding the design pressure value by two rupture discs, which discharge into the reactor containment. The rupture disc relief conditions are given in Table 4.1-3.

4.2.4 Protection Against Proliferation Of Dynamic Effects

Engineered safety features and associated systems are protected from loss of function due to dynamic effects and missiles, which might result from a loss-of coolant accident. Protection is provided by missile shielding and/or segregation of redundant components. This is discussed in detail in Chapter 6. The reactor coolant system is surrounded by a concrete shield wall. This wall provides shielding to permit access into the containment annular region during full-power operation for inspection and maintenance of miscellaneous equipment. This shielding wall also provides missile protection for the containment liner plate.

The concrete deck over the reactor coolant system also provides for shielding and missile damage protection.

Lateral bracing is provided near the steam-generator upper tube sheet elevation to resist lateral loads, including those resulting from seismic forces and pipe rupture forces. Additional bracing is provided at a lower elevation to resist pipe rupture loads.

Missile protection afforded by the arrangement of the reactor coolant system is illustrated in the containment structure drawings, which are given in Chapter 5.

[Historical Information Only] This paragraph is retained for historical purposes only.
The integrity of the reactor coolant system as may be affected by asymmetric loss-of-coolant accident loads due to postulated pipe breaks in the primary loop coolant piping was considered

in WCAP-9117 (Reference 3) and in a subsequent submittal to the NRC on June 15, 1978 (Reference 4). The combination of LOCA and safe shutdown earthquake loads applied to the results of WCAP-9117 was described in the September 3, 1980 follow-up submittal (Reference 5), and is applicable to both Indian Point Units 2 and 3, based on the similarity between the two units. The safe shutdown earthquake results for Unit 3 would be similar to those for Unit 2, and the total strains also apply to Unit 2. The NRC Safety Evaluation Report (Reference 6) concluded that the assessment of asymmetric loss-of-coolant accident and safe shutdown earthquake loads for the Indian Point Unit 2 was acceptable. This conclusion was based upon the installation of pipe motion limiters in the primary shield wall and was contingent upon the verification of shield plug assumptions, which included the determination of the effects of the plugs as missiles, and the determination that structural components do not inhibit plug displacement. This verification was provided in a subsequent submittal to the NRC on June 10, 1986 (Reference 22).

In 1989, the NRC approved changes to the design bases with respect to dynamic affects of postulated primary loop pipe ruptures, as discussed in Section 4.1.2.4.

4.2.5 Materials Of Construction

Each of the materials used in the reactor coolant system is selected for the expected environment and service conditions. The major component materials are listed in Table 4.2-1.

All reactor coolant system materials that are exposed to the coolant are corrosion resistant. They consist of stainless steels and Inconel, and they are chosen for specific purposes at various locations within the system for their superior compatibility with the reactor coolant. Reactor coolant chemistry is further discussed in Section 4.2.8.

It is characteristic of stress corrosion that combinations of alloy and environment, which result in cracking are usually quite specific. Environments that have been shown to cause stress-corrosion cracking of stainless steels are free alkalinity in the presence of a concentrating mechanism, and the presence of chlorides and free oxygen. With regard to the former, experience has shown that deposition of chemicals on the surface of tubes can occur in a steam blanketed area within a steam generator. In the presence of this environment, stress-corrosion cracking can occur in stainless steels having the nominal residual stresses resulting from normal manufacturing procedures. However, the steam generator contains Inconel tubes. Testing to investigate the susceptibility of heat exchanger construction materials to stress corrosion in caustic and chloride aqueous solutions has indicated that Inconel alloy has excellent resistance to general and pitting-type corrosion.

Considerable experience with Inconel in steam generator and heat exchanger applications has been accumulated in the industry. Since 1962, widespread adoption of Inconel for steam generator tubes in nuclear stations is evident: as for example, Connecticut-Yankee; San Onofre; PM-1, Sundance; PM-3A, McMurdo Sound; CVTR; NPD, and Hanford N-Reactor. Materials with lead traces in the overall composition were present in the secondary side of the referenced plants. The use of lead in the materials of the secondary side of the Indian Point plant has been minimized to the practical limit of that occurring as trace elements in metallurgical alloys and as such is insignificant.

All external insulation of reactor coolant system components is compatible with the component materials. The cylindrical shell exterior and closure flanges to the reactor vessel are insulated with metallic reflective insulation. The closure head is also insulated with metallic reflective

insulation. All other external corrosion-resistant surfaces in the reactor coolant system are insulated with low or halide-free insulating material as required.

The reactor vessel was fabricated by Combustion Engineering, Inc. A sketch of the reactor vessel showing all materials in the beltline region is shown in Figure 4.2-11. Information on each of the welds and plates in the beltline region is shown in Tables 4.2-2 through 4.2-5, and Tables 4.2-5 through 4.2-8, respectively. Information relative to weld and plate material included in the material surveillance program is shown in Tables 4.2-2 and 4.2-6 through 4.2-8. Details concerning the reactor vessel radiation surveillance program are provided in WCAP-7323 (Reference 7) and in the Technical Specifications.

The reactor vessel plate or forging material opposite the core is purchased to a specified Charpy V-notch test result of 30-ft-lb or greater at a corresponding nil-ductility transition temperature (NDTT) of 40^{o}F or less, and the material is tested to verify conformity to specified requirements and to determine the actual NDTT value (see Table 4.2-9). In addition, this plate is 100-percent volumetrically inspected by ultrasonic test using both longitudinal and shear wave methods.

The remaining material in the reactor vessel and other reactor coolant system components meets the appropriate design code requirements and specific component function.

The reactor vessel material is heat-treated specifically to obtain good notch-ductility, which ensures a low NDTT, and thereby gives assurance that the finished vessel can be initially hydrostatically tested and operated near room temperature within the restrictions of NDTT + 60°F. The stress limits established for the reactor vessel are dependent upon the temperature at which the stresses are applied. As a result of fast neutron irradiation in the region of the core, the material properties will change, including an increase in the NDTT. An initial maximum value of NDTT of 40°F has been established during fabrication in this region.

The techniques used to measure and predict the integrated fast neutron ($E > 1$ MeV) fluxes at the sample locations are described in Appendix 4A. The calculation method used to obtain the maximum neutron ($E > 1$ MeV) exposure of the reactor vessel is identical to that described for the irradiation samples. Since the neutron spectra at the samples and vessel inner surface are identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of reactor vessel for some later stage in plant life. The maximum exposure of the vessel will be obtained from the measured sample exposure by appropriate application of the calculated azimuthal neutron flux variation.

The evaluation of the second surveillance capsule is discussed in detail in Reference 8. The analysis for the third surveillance capsule, removed during the 1984 refueling outage, is documented in Reference 9.

The analysis of the fourth surveillance Capsule V, removed during the 1987-88 refueling outage (end of Cycle 8), is documented in Reference 11. The Capsule V received a fast neutron ($E > 1$ MeV) fluence of 5.3×10^{18} n/cm^2 in 8.6EFPY at the end of Cycle 8. See Reference 11.

The maximum integrated fast neutron ($E > 1$ MeV) exposure of the vessel for 32 EFPYs is calculated to be 1.39×10^{19} n/cm^2 based on the measurements from the fourth surveillance Capsule V. Fast neutron fluences corresponding to 32 EFPYs at various reactor vessel thicknesses are given in Table 4.2-10.

The calculated neutron exposure exceeds the value of 0.85 x 10^{19} n/cm^2 (E > 1MeV) reported in the First Supplement to the Preliminary Facility Description Safety Analysis Report. The reasons for the increase are:

1. Anticipated increase in reactor power from 2758 MWt to 3071.4 MWt in Cycle 10 and then to 3114.4 MWt in Cycle 16 and subsequently to 3216 MWt in Cycle 17.
2. Revision of analysis methodology including upgrading of neutron cross sections and codes.
3. Core design considerations involving changes in loading patterns.

The above projected exposure to reactor vessel from Capsule V measurements is based upon using the standard loading pattern (only fresh fuel assemblies at core periphery) in Cycles 1 thru 5 and the low leakage loading pattern (mixture of fresh and spent fuel assemblies) or L^3P starting from Cycle 6. Furthermore, it assumes Indian Point Unit 2 operation at stretch power operation at 3071.4 MWt core power level starting from Cycle 10, then at the Appendix K uprated power level of 3114.4 MWt starting in Cycle 16 and subsequently at 3216 MWt starting in Cycle 17.

The maximum reference temperature, RT_{NDT} for the Indian Point Unit 2 vessel core beltline materials at the 1/4 thickness and the 3/4 thickness after 32 effective full power years of operation are projected to be 240°F and 194°F, respectively, based on calculations performed per Regulatory Guide 1.99, Revision 2, using data obtained from evaluation of Surveillance Capsule V. (Ref.11). This data provides the basis for subsequent calculation of Adjusted Reference Temperature values for determination of allowable pressure/temperature limits for operation to 25 EFPY, as described in Reference 19.

To evaluate the NDTT shift of welds, heat affected zones, and base material for the vessel, test coupons of these material types have been included in the reactor vessel surveillance program described in Section 4.5.2. The methods used to measure the initial NDTT of the reactor vessel baseplate material are given in Appendix 4A.

The reference nil ductility transition temperatures for pressurized thermal shock evaluation (RT_{PTS}) have been estimated[11,12,13] in accordance with 10 CFR 50.61(b)(2). The values at 15 EFPY and also at the end of the license term are well below the screening criteria of 270°F (for plates and axial weld materials) and 300°F (for circumferential weld materials), based on a low-leakage core design. The NRC has accepted this analysis.[14] Additional information in response to Generic Letter 92-01, Revision 1, is given in reference 21.

With regard to electroslag welding of Class I components, the Indian Point Unit 2 90-degree elbows were electroslag welded. The following efforts were performed for quality assurance of these components.

1. The electroslag welding procedure employing one-wire technique was qualified in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section IX and Code Case 1355 plus supplementary evaluations as requested by Westinghouse. The following test specimens were removed from a 5-in. thick weldment and successfully tested:

   a. Six transverse tensile bars - as welded.
   b. Six transverse tensile bars - 2050°F, H_2O quench.

c. Six transverse tensile bars - 2050°F, H_2O quench + 750°F stress relief heat treatment.
d. Six transverse tensile bars - 2050°F, H_2O quench, tested at 650°F.
e. Twelve guided side bend test bars.

2. The casting segments were surface conditioned for 100-percent radiographic and penetrant inspections. The acceptance standards were ASTM E-186 severity level 2 (except no category D or E defectiveness was permitted) and ASME Section III, Paragraph N-627, respectively.

3. The edges of the electroslag weld preparations were machined. These surfaces were penetrant inspected prior to welding. The acceptance standards were ASME Section III, Paragraph N-627.

4. The completed electroslag weld surfaces were ground flush with the casting surface. Then the electroslag weld and adjacent base material were 100-percent radiographed in accordance with ASME Code Case 1355. Also, the electroslag weld surfaces and adjacent base material were penetrant inspected in accordance with ASME Boiler and Pressure Vessel Code Section III, Paragraph N-627.

5. Weld metal and base metal chemical and physical analyses were determined and certified.

6. Heat treatment furnace charts were recorded and certified.

Two of the Indian Point Unit 2 reactor coolant pump casings were electroslag welded. The following efforts were performed for quality assurance of these two components.

1. The electroslag welding procedure employing two- and three-wire technique was qualified in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section IX and Code Case 1355 plus supplementary evaluations as requested by Westinghouse. The following test specimens were removed from an 8-in.-thick and from a 12-in.-thick weldment and successfully tested for both the two-wire and the three-wire techniques, respectfully.

   a. Two-wire electroslag process - 8-in.-thick weldment.

      (1) 6 transverse tensile bars - 750°F postweld stress relief.
      (2) 12 guided side bend test bars.

   b. Three-wire electroslag process - 12-in.-thick weldment.

      (1) 6 transverse tensile bars - 750°F postweld stress relief.

      (2) 17 guided side bend test bars.

      (3) 21 Charpy V-notch specimens.

      (4) Full-section macroexamination of weld and heat affected zone.

(5) Numerous microscopic examinations of specimens removed from the weld and heat affected zone regions.

(6) Hardness survey across weld and heat affected zone.

c. A separate weld test was made using the two-wire electroslag technique to evaluate the effects of a stop and restart of welding by this process. This evaluation was performed to establish proper procedures and techniques as such an occurrence was anticipated during production applications due to equipment malfunction, power outages, etc. The following test specimens were removed from an 8-in.-thick weldment in the stop-restart-repaired region and successfully tested.

(1) 2 transverse tensile bars - as welded.

(2) 4 guided side bend test bars.

(3) Full section macroexamination of weld and heat affected zone.

d. All of the weld test blocks in (a), (b), and (c) above were radiographed using a 24-MeV Betatron. The radiographic quality level obtained was between 0.05 to 1-percent. There were no discontinuities evident in any of the electroslag welds.

(1) The casting segments were surface conditioned for 100-percent radiographic and penetrant inspections. The radiographic acceptance standards were ASTM E-186 severity level 2 (except no category D or E defectiveness was permitted) for section thickness up to 4.5-in. and ASTM E-280 severity level 2 for section thicknesses greater than 4.5-in. The penetrant acceptance standards were ASME Boiler and Pressure Vessel Code, Section III, Paragraph N-627.

(2) The edges of the electroslag weld preparations were machined. These surfaces were penetrant inspected prior to welding. The acceptance standards were ASME Boiler and Pressure Vessel Code, Section III, Paragraph N-627.

(3) The completed electroslag weld surfaces were ground flush with the casting surface. Then the electroslag weld and adjacent base material were 100-percent radiographed in accordance with ASME Code Case 1355. Also, the electroslag weld surfaces and adjacent base material were penetrant inspected in accordance with ASME Boiler and Pressure Vessel Code, Section III, Paragraph N-627.

(4) Weld metal and base metal chemical and physical analyses were determined and certified.

(5) Heat treatment furnace charts were recorded and certified.

The two remaining Indian Point Unit 2 reactor coolant pump casings were submerged arc welded. Quality Assurance procedures and Quality Assurance inspections equivalent to the above were also exercised on these casings.

4.2.6 Maximum Heating And Cooling Rates

The reactor system operating cycles used for design purposes are given in Table 4.1-8 and described in Section 4.1.5. The reactor coolant system heatup, cooldown, and leak test limitations curves are included in the Technical Specifications. Starting with a minimum water level, sufficient electrical heaters are installed in the pressurizer to permit a heatup rate of 55°F/hr. This rate takes into account the small continuous spray flow provided to maintain the pressurizer liquid homogeneous with the coolant. The fastest cooldown rates, which result from the hypothetical case of a break of a main steam line are discussed in Section 14.2.5.

4.2.7 Leakage

The existence of leakage from the reactor coolant system to the containment regardless of the source of leakage is detected by one or more of the following conditions:

1. Radiation sensitive instruments provide the capability for detection of leakage from the reactor coolant system. The containment air particulate monitors are quite sensitive to low leak rates. The containment radiogas monitors are less sensitive but are used in addition to the air particulate monitor.

2. A third mechanism used in leak detection is the humidity detectors. These provide a means of measuring overall leakage from all water and steam systems within the containment, which can affect containment humidity. The humidity monitoring method is considered supplemental to the radiation monitoring methods.

3. A leakage detection system collects and measures moisture condensed from the containment atmosphere by cooling coils of the main air recirculation units. The condenser moisture includes, of course, any leaks from the cooling coils themselves. This system provides a dependable and accurate means of measuring the total leakage from these sources. Condensate flows of approximately 1.0 gpm to 15 gpm per detector can be measured by this system. Condensate flows can be determined using weir calibration curves in conjunction with the weir water head displayed by the weir water meter, or by direct reading of the weir integrated condensate flow on the weir meter.

4. An increase in the amount of coolant makeup water, which is required to maintain normal level in the pressurizer or an increase in containment sump level provide additional means of detecting leakage.

The Technical Specifications provide the requirements and bases for leakage detection.

In considering potential leakage from the reactor coolant system containing primary coolant at high pressure, four categories are described and evaluated in Section 6.7.1. These include leakage paths to the reactor coolant drain tank, leakage paths to the pressurizer relief tank, leakage paths to the containment environment, and leakage paths to the interconnecting systems.

4.2.7.1 Maximum Leak Rates

The maximum leak rate from an unidentified source that will be permitted during normal operation is specified in the Technical Specifications. Leakage from the reactor coolant system is collected in the containment or by the other closed systems. These closed systems are: the steam and feedwater system, the waste disposal system, and the component cooling system. Assuming the existence of the maximum allowable activity in the reactor coolant, the rate of unidentified leakage is a conservative limit on what is allowable before the guidelines of 10 CFR 20 would be exceeded.

With the limiting reactor coolant activity and assuming initiation of a leak from the reactor coolant system to the component cooling system, the radiation monitor that samples the component cooling pump discharge downstream of the component cooling heat exchangers would annunciate in the control room and initiate closure of the surge tank vent line in the component cooling system. In the case of failure of the closure of the vent line and resulting continuous discharge in the atmosphere via the component cooling surge tank vent, the resultant dose at the site boundary would be within the limit allowed by 10 CFR 20.

Leakage directly into the containment indicates the possibility of a breach in the coolant envelope. The limitation specified by the Technical Specifications for a source of leakage not identified is sufficiently above the minimum detectable leakage rate to provide a reliable indication of leakage. The leakage limit is well within the capacity of one coolant charging pump.

The conservative approach that is used in the design and fabrication of the components that constitute the primary system pressure boundary together with the operating restrictions, which are imposed for system heatup and cooldown give adequate assurance that the integrity of the primary system pressure boundary is maintained throughout plant life. The periodic examination of the primary pressure boundary via the inservice inspection program (specified in the Technical Specifications) will physically demonstrate that the operating environment will have no deleterious effect on the primary pressure boundary integrity.

The maximum unidentified leak rate that is permitted during normal operation is well within the sensitivity of the leak detection systems incorporated within the containment, and it reflects good operating practice based on operating experience gained at other PWR plants. Detection of leakage from the primary system directs the operator's attention to potential sources of leakage, such as valves, and permits timely evaluation to ensure that any associated activity release does not constitute a public hazard, that the reactor coolant inventory is not significantly affected, and that the leakage is well within the capability of the containment drainage system. See also Section 6.7 for a further discussion of leakage detection.

4.2.7.2 Leakage Prevention

Reactor coolant system components are manufactured to exacting specifications, which exceed normal code requirements (as outlined in Section 4.1.7). In addition, because of the welded construction of the reactor coolant system and the extensive nondestructive testing to which it is subjected (as outlined in Section 4.5), it is considered that leakage through metal surfaces or welded joints is very unlikely.

However, some leakage from the reactor coolant system is permitted by design from the reactor coolant pump seals. Also, all sealed joints are potential sources of leakage even though the

most appropriate sealing device is selected in each case. Thus, because of the large number of joints and the difficulty of ensuring complete freedom from leakage in each case, a small integrated leakage is considered acceptable. Leakage from the reactor through its head flange will leak-off between the double O-ring seal and actuate an alarm in the control room.

4.2.7.3 Locating Leaks

Experience has shown that hydrostatic testing is successful in locating leaks in a pressure containing system.

The Reactor Coolant System shall be tested for leakage at normal operating pressure prior to plant startup following each refueling outage, in accordance with the requirements of the applicable edition and addenda of the ASME Section XI Code. Leak test of the Reactor Coolant System is required by the ASME Boiler and Pressure Vessel Code, Section XI, to ensure leak tightness of the system during operation. The test frequency and conditions are specified in the Code.

Testing of repairs, replacements or modifications for the Reactor Coolant System shall meet the requirements of the applicable edition and addenda of the ASME Section XI Code. For repairs on components, the thorough non-destructive testing gives a very high degree of confidence in the integrity of the system, and will detect any significant defects in and near the new welds. In all cases, the leak test will assure leak-tightness during normal operation.

Methods of leak location, which can be used during plant shutdown include visual observation for escaping steam or water, or for the presence of boric acid crystals near the leak. The boric acid crystals are transported outside the reactor coolant system in the leaking fluid and deposited by the evaporation process.

4.2.8 Water Chemistry

The water chemistry is selected to provide the necessary boron content for reactivity control and to minimize corrosion of reactor coolant system surfaces. All materials exposed to reactor coolant are corrosion resistant. Periodic analyses of the coolant chemical composition are performed to monitor the adherence of the system to the required reactor coolant water quality. Maintenance of the water quality to minimize corrosion is accomplished using the chemical and volume system and sampling system that is described in Chapter 9.

4.2.9 Reactor Coolant Flow Measurement

Elbow taps are used in the primary coolant system as an instrument device that indicates the status of the reactor coolant flow. The basic function of this device is to provide information as to whether or not a reduction in flow rate has occurred. The correlation between flow reduction and elbow tap read out has been well established by the following equation:

$$\frac{\Delta P}{\Delta P_o} = \left(\frac{\omega}{\omega_o}\right)^2$$

where ΔP_o is the reference pressure differential with the corresponding referenced flow rate ω_o, and ΔP is the pressure differential with the corresponding referenced flow rate. The full flow reference point was established during initial plant startup. The low flow trip point was then

established by extrapolating along the correlation curve. The technique has been well established in providing core protection against low coolant flow in Westinghouse PWR plants. The expected absolute accuracy of the channel is within 10-percent and field results have shown the repeatability of the trip point to be within 1-percent. The analysis of the loss-of-flow transient is presented in Section 14.1.6.

4.2.10 Reactor Coolant Vent System

4.2.10.1 Design Basis

The remote reactor coolant vent system has been designed and installed in accordance with NUREG-0737 Item II.B.1 to allow for remote manual venting of gases from the reactor vessel head should they accumulate there. The power-operated relief valve system acts as the remote operated vent system for the pressurizer (see Section 4.2.3) and as a redundant backup to the vessel head vent system.

4.2.10.2 System Description

4.2.10.2.1 Power-operated Relief Valve System

The power-operated relief valve system is discussed in Sections 4.2.3 and 4.3.4.

4.2.10.2.2 Remote Reactor Head Vent System

The original manual reactor vessel head vent line has been extended and two motor operated valves have been installed in series to facilitate venting of the reactor vessel head from the control room. The release point is located above the operating floor at an elevation of approximately 105-ft and is situated so that the discharge of the system will not impinge on any structures, systems, or components essential to the reactor safe shutdown or mitigation of a design basis accident.

The power-operated relief valve system relieves to the pressurizer relief tank. The remote reactor head vent, and the power-operated relief valves and their associated block valves are in three separate lines and are supplied with three independent emergency power sources so that at least one vent path will remain functional after the single failure of an emergency power train.

Potential seat leakage through both valves is vented directly to the containment atmosphere and is detected and monitored as part of the reactor coolant system leakage requirements specified in the Technical Specifications.

The two series motor operated head vent valves are closed and deenergized during normal plant operation. The circuit breakers will be locked open to prevent inadvertent operation. If the need should arise for venting the reactor, the two breakers of the remote head vent valves will be reenergized and the valves opened as necessary from the central control room accident assessment panel.

4.2.10.3 Design Criteria

The reactor coolant vent system piping, valves, components, and supports are classified seismic Class I and Class A. They have been designed and installed in accordance with the original requirements for reactor coolant pressure boundary installations, and ASME and ANSI

codes applicable to Indian Point Unit 2. The piping, valves, and fittings were fabricated from stainless steel and are compatible with reactor coolant chemistry.

To alleviate the potential hazard of missiles, the remote reactor head vent system was installed such that it does not come close to and have the ability to damage safety-related systems required for safe reactor shutdown or mitigation of a design basis accident.

4.2.10.4 Design Evaluation

Consistent with NUREG-0737 Item II.B.1 Clarification A(4), the new remote reactor head vent system was designed with sufficient flow restriction that in the event of inadvertent opening or line breaks, normal makeup charging flow from the chemical and volume control system is capable of precluding actuation of the safety injection system. The original reactor vessel head vent consisted of a 3/4-in. line with a manual (locally operated) shutoff valve and bolted blind flange and was used only for routine operations when the reactor was shut down. When the remotely operated head vent system was installed, the blind flange was removed and additional nominally 3/4-in. tubing (9/16-in. ID) was run from the existing 3/4-in. NPS line to the new motor-operated head vent valves. Those portions of the system, which were revised were designed and constructed to the same criteria as the original Indian Point Unit 2 pressure boundary components.

A specific calculation has been performed for the worst case break location for the revised vent system (i.e., the interface between the 9/16-in. tubing and the original 3/4-in. head vent piping). This calculation determined that even at this worst case location, the break flow would be well within the capacity of two chemical and volume control system charging pumps without actuating safeguards equipment. Thus, failure of the vent system would not result in a break size corresponding to the definition of a loss-of-coolant accident.

4.2.11 Reactor Vessel Level Indication System

The reactor vessel level indication system (RVLIS) has been installed in accordance with the requirements of NUREG-0737. The system is mainly part of the "Inadequate Core Cooling (ICC) Instrumentation" in improving the reliability of the plant operator to diagnose the approach of inadequate core cooling and to assess the adequacy of responses taken to restore cooling. The system also provides assistance to the operator in determining the presence of voids in the vessel. Additional information is given in Section 7.5.2.

4.2.11.1 Design Basis

The system has been designed to provide continuous indication of coolant level inside the reactor and to assist the operator in determining the presence of voids in the vessel. The system was designed by Westinghouse as a Class A-Class 1E system.

4.2.11.2 System Description

The RVLIS, shown in Plant Drawing 208798 [Formerly UFSAR Figure 4.2-12], is based on the differential pressure principle as sensed by taps located at the top and bottom of the reactor vessel. The top tap is installed on an unused control rod penetration and the bottom tap uses an unused incore instrument thimble to the seal table. The differential pressure is transmitted through filled capillary systems to transmitters outside containment. The temperature sensors are mounted on each capillary inside the containment. The signals from the temperature

sensors and transmitters are routed to a Class 1E panel in the cable spreading room. The temperature compensated signals of level indication are indicated on the accident assessment panel in the Unit 1/Unit 2 central control room.

The reactor vessel level indication system, which was installed in response to NUREG-0737, Inadequate Core Cooling Instrumentation, has been approved by the NRC.

REFERENCES FOR SECTION 4.2

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2. D. H. Winne and B. M. Wundt, "Application of the Griffity-Irwin Theory of Crack Propagation to the Bursting Behavior of Discs, Including Analytical and Experimental Studies," ASME Paper Number 57-A-249, December 1, 1957.

3. Westinghouse Electric Corporation, "Analysis of Reactor Coolant System for Postulated Loss-of-Coolant Accident: Indian Point Unit 3 Nuclear Power Plant," WCAP-9117 (Proprietary) and WCAP-9130 (Nonproprietary).

4. Letter from W. J. Cahill, Con Edison, to V. Stello, NRC, Subject: Reactor Vessel Supports, dated June 15, 1978.

5. Letter from P. Zarakas, Con Edison, to T. Novak, NRC, Subject: Effects of Combining Seismic (SSE) and LOCA Responses on WCAP-9117 Analysis, dated September 3, 1980.

6. Letter from S. A. Varga, NRC, to J. D. O'Toole, Con Edison, Subject: Safety Evaluation Report on Asymmetric LOCA Loads, dated March 28, 1984.

7. S. Yanichko, "Indian Point 2 Reactor Vessel Radiation Surveillance Program," WCAP-7323, Westinghouse Electric Corporation, May 1969.

8. Letter from J. D. O'Toole, Con Edison, to S. A. Varga, NRC, Subject: Application for Amendment to Operating License, dated May 5, 1982.

9. Letter from J. D. O'Toole, Con Edison, to S. A. Varga, NRC, Subject: Reactor Vessel Material Surveillance Program, Analysis of Capsule Z, dated May 7, 1984.

10. Deleted (superceded by Reference 11).

11. "Reactor Vessel Material Surveillance Program for Indian Point Unit No. 2 Analysis of Capsule V", SWRI Final Report Project No. 17-2108, March 1990.

12. Letter from J.D. O'Toole, Con Edison, to S.A. Varga, NRC, Subject: Reference Transition Temperature For Pressurized Thermal Shock Evaluations, dated January 22, 1986.

13. Letter from Murray Selman, Con Edison, to Document Control Desk, NRC, transmitting supplemental information, dated January 12, 1987.

14. Letter from Marylee M. Slosson, NRC, to Murray Selman, Con Edison, Subject: Projected Values of Material Properties for Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events, dated February 27, 1987.

15. Deleted

16. Deleted

17. Deleted

18. Letter from Stephen B. Bram, Con Edison, to Document Control Desk, NRC, Subject: Close-out for NRC Bulletin 88-11 Pressurizer Surge Line Stratification, dated October 1, 1991.

19. Letter from Jefferey F. Harold, NRC, to Paul H. Kinkel, Con Edison, Subject: Issuance of Amendment for Indian Point Nuclear Generating Unit No. 2 (TAC No. M96944), dated February 27, 1998.

20. Deleted

21. Letter from Stephen B. Bram, Con Edison, to Document Control Desk, NRC. Subject: Reactor Vessel Structural Integrity, 10 CFR 50.54(f), (Generic Letter 92-01, Revision 1), dated July 6, 1992.

22. Letter from J. O'Toole, Con Edison, to M Slosson, NRC, Dated June 10, 1986.

23. Letter from R. Capra, NRC, to S. Bram,, Con Edison, dated December 24, 1987, Subject: Indian Point Nuclear Generating Unit No. 2 Steam Generator Girth Weld Repair (TAC 66684).

24. Letter from R. Capra, NRC, to S. Bram,, Con Edison, dated October 28, 1988, Subject: Steam Generator Girth Weld Repair Safety Evaluation for Indian Point Nuclear Generating Unit No. 2 (TAC 66684).

25. Letter from R. Capra, NRC, to S. Bram,, Con Edison, dated January 4, 1991, Subject: Review of Mid-Cycle Steam Generator Girth Weld and Feedwater Inlet Nozzle Work (TAC 72961).

26. WCAP-15629, Rev. 1, "Indian Point Unit 2 Heatup and Cooldown Curves for Normal Operation and PTLR Support Documentation," Westinghouse Electric Co., December 2001.

TABLE 4.2-1
Materials of Construction of the Reactor Coolant System Components

Component	Section	Materials
Reactor vessel	Pressure plate	SA-302, Gr. B
	Shell and nozzle forgings	SA-302, Gr. B / SA-336
	Cladding, stainless weld rod	Type 304 equivalent
	Thermal shield and internals	A-240 Type 304
	Insulation	Stainless steel, Aluminum
Steam generator	Pressure plate	SA-533, Grade A Class 2
	Cladding, stainless weld rod	Type 304 equivalent
	Cladding for tube sheets	Inconel
	Tubes	SB-163, Thermally Treated (Code Case N-20)
	Channel head castings	SA-216 WCC
Pressurizer	Shell	SA-302 Gr. B
	Heads	SA-216 WCC
	External plate (support skirt)	SA-516, Gr. 70
	Cladding, stainless	Type 304 equivalent
	Internal plate	SA-240 Type 304
	Spray Nozzle	SA-376 Type 316
Pressurizer relief tank	Shell	A-285 Gr. C
	Heads	A-285 Gr. C
	Internal surface coating	Amercoat 55 system
Piping	Pipes	A-376 Types 304 and 316
	Fittings	A-351 CF8M
	Nozzles	A-182 Type F316
Pump	Shaft	Type 304
	Impeller	A-351 CF8
	Casing	A-351 CF8M
Valves	Pressure containing parts	A-351 CF8 and CF8M; A-182 Type F316, and ASME SA182 Type F316 ASTM A479 Type 316

TABLE 4.2-2
Identification of Indian Point Unit 2 Reactor Vessel Beltline Region Weld Metal

Weld Location	Welding Process	Weld Control No.	Weld Wire Type	Heat No.	Flux Type	Lot No.	Post-Weld Heat Treatment
Nozzle shell vertical seam 1-042 A, B, and C	Submerged Arc	-	RACO 3 +Ni 200	W5214 N7048A	Linde 1092	3600	1125 ± 25°F 25 hr-FC
Inter shell vertical seam circle seam 8-042	Submerged Arc	-	RACO 3 +Ni 200	W5214 N7048A	Linde 1092	3600	1125 ± 25°F 25 hr-FC
Nozzle shell to inter seam 2-042 A, B, and C	Submerged Arc	-	RACO 3 +Ni 200	W5214 N7048A	Linde 1092	3600	1125 ± 25°F 25 hr-FC
Inter shell to lower shell circle seam 9-042	Submerged Arc	M1.03	RACO 3 +Ni 200	34B009 N9867A	Linde 1092	3708	1150 ± 25°F 40 hr-FC
Lower shell vertical seams 3-042 A and B	Submerged Arc	-	RACO 3 +Ni 200	W5214 -	Linde 1092	3576	1150 ± 25°F 40 hr-FC

Surveillance weld	Submerged Arc	-	RACO 3 +Ni 200	W5214 N7048A	Linde 1092	3600	1150 ± 25°F 19 3/4 hr-FC

TABLE 4.2-3
Chemical Composition of Reactor Vessel Beltline Region Weld Metal

Weld Wire Type	Weld Wire Heat No.	Flux Type	Lot No.	C	Mn	P	S	Si	Mo	Cr	Ni	Cu
				Weight Percent								
RACO 3	W5214	Linde 1092	3600	.11	1.20	.021	.012	.19	.52	--	--	--
RACO 3	34B009	Linde 1092	3708	.14	2.01	.010	.017	.04	.51	--	--	--[1]
RACO 3	W5214	Linde 1092	3576	.12	1.15	.021	.012	.21	.56	--	--	--

Surveillance Weld - Not Performed

Notes:

1. Chemical analysis of bare wire - No as-deposited analysis available.

TABLE 4.2-4
Mechanical Properties of Reactor Vessel Beltline Region Weld Metal

Weld Wire Type	Weld Wire Heat No.	Flux Type	Flux Lot No.	T_{NDT}[1] (°F)	Energy at 10°F (ft-lbs)	RT_{NDT}[1] (°F)	Shelf Energy (ft-lbs)	YS (ksi)	UTS (ksi)	Elong Percent	RA Percent
RACO 3	W5214	Linde 1092	3600	0	103,93,95	0	--	65.5	80.0	31.0	71.5
RACO 3	34B009	Linde 1092	3708	0	84,71,90	0	--	67.9	84.2	31.0	69.8
RACO 3	W5214	Linde 1092	3576	0	57,51,69	0	--	68.5	85.0	27.5	68.5
Surveillance Weld				0	78,74,81	0	121	64.75	80.85	27.7	72.7

NOTES:

1. Estimated per NRC Standard Review Plan Section 5.3.2.

TABLE 4.2-5
Maximum End-of-Life Fluence at Vessel Inner Wall Locations

Plate or Weld Location	Seam or Plate No.	Fluence (n/cm^2)
Nozzle Shell Vertical Seam	1-042A	6.6×10^{17}
Nozzle Shell Vertical Seam	1-042B	4.4×10^{17}
Nozzle Shell Vertical Seam	1-042C	1.1×10^{18}
Nozzle Shell to Inter. Shell Circle Seam	8-042	1.3×10^{18}
Intermediate Shell Vertical Seam	2-042A	8.8×10^{18}
Intermediate Shell Vertical Seam	2-042B	8.8×10^{18}
Intermediate Shell Vertical Seam	2-042C	5.0×10^{18}
Intermediate Shell to Lower Shell Circle Seam	9-042	1.6×10^{19}
Lower Shell Vertical Seam	3-042A	7.0×10^{18}
Lower Shell Vertical Seam	3-042B	7.0×10^{18}
Nozzle Shell Plate	B2001-1	1.3×10^{18}
Nozzle Shell Plate	B2001-2	1.3×10^{18}
Nozzle Shell Plate	B2001-3	1.3×10^{18}
Intermediate Shell Plate	B2002-1	1.6×10^{19}
Intermediate Shell Plate	B2002-2	1.6×10^{19}
Intermediate Shell Plate	B2002-3	1.6×10^{19}
Lower Shell Plate	B2003-1	1.6×10^{19}
Lower Shell Plate	B2003-2	1.6×10^{19}

TABLE 4.2-6
Identification of Reactor Vessel Beltline Region Plate Material

Component	Plate No.	Heat No.	Mat'l Spec No.	Supplier	Austenitize	Heat Treatment Temper	Stress Relief
Nozzle shell	B2001-1	B4679	A302B Mod.	Lukens	1550-1650°F 4 hr-WQ	1200-1250°F 4hr-AC	1125-1175°F 60hr-FC
Nozzle shell	B2001-2	B4701	A302B Mod.	Lukens	1550-1650°F 4 hr-WQ	1200-1250°F 4hr-AC	1125-1175°F 60hr-FC
Nozzle shell	B2001-3	A9870	A302B Mod.	Lukens	1550-1650°F 4 hr-WQ	1200-1250°F 4hr-AC	1125-1175°F 50hr-FC
Inter shell	$B2002\text{-}1_1$	B4688	A302B Mod.	Lukens	1550-1650°F 4 hr-WQ	1200-1250°F 4hr-AC	1125-1175°F 50hr-FC
Inter shell	$B2002\text{-}2_1$	B4701	A302B Mod.	Lukens	1550-1650°F 4 hr-WQ	1200-1250°F 4hr-AC	1125-1175°F 50hr-FC
Inter shell	$B2002\text{-}3_1$	B4922	A302B Mod.	Lukens	1550-1650°F 4 hr-WQ	1200-1250°F 4hr-AC	1125-1175°F 40hr-FC
Lower shell	B2003-1	B4791	A302B Mod.	Lukens	1550-1650°F 4 hr-WQ	1200-1250°F 4hr-AC	1125-1175°F 40hr-FC
Lower shell	B2003-2	B4782	A302B Mod.	Lukens	1550-1650°F 4 hr-WQ	1200-1250°F 4hr-AC	1125-1175°F 40hr-FC

Notes:

1. Surveillance Material.

TABLE 4.2-7
Chemical Composition of Reactor Vessel Beltline Region
Plate Material, Weight Percent

Plate No.	C	Mn	P	S	Si	Ni	Mo	Cu
B2001-1[1]	0.22	1.35	0.010	0.022	0.24	0.50	0.46	0.20
B2001-2[1]	0.23	1.27	0.011	0.021	0.23	0.43	0.47	0.14
B2001-3[1]	0.23	1.35	0.012	0.025	0.26	0.50	0.48	0.19
B2002-1[2]	0.20	1.28	0.010	0.019	0.25	0.65	0.46	0.19
B2002-2[2]	0.22	1.30	0.014	0.020	0.22	0.46	0.50	0.17
B2002-3[2]	0.22	1.29	0.011	0.018	0.25	0.60	0.46	0.25
B2003-1[1]	0.23	1.33	0.011	0.025	0.23	0.66	0.48	0.20
B2003-2[1]	0.21	1.30	0.010	0.021	0.23	0.48	0.45	0.19

Notes:

1. Surveillance Material - No analysis performed other than reported by supplier.
2. Best estimate Cu and Ni weight percent [26].

TABLE 4.2-8
Mechanical Properties of Reactor Vessel Beltline Region Plate Material

Plate No.	^{T}NDT (°F)	^{RT}NDT$_1$ (°F)	Shelf Energy$_1$ (ft-lb)	YS (ksi)	UTS (ksi)	Elongation (percent)	RA (percent)	
B2001-1	-10	24	69	67.25	87.75	26.00	64.45	
B2001-2	-10	18	63.5	63.25	85.25	27.25	65.75	
B2001-3	-10	25	69	65.25	86.75	25.00	63.75	
B2002-1	-20	34	70	70.75	91.50	25.00	64.75	
B2002-2	-30	21	73	65.00	85.25	26.50	67.00	
B2002-3	-10	21	73.5	68.95	90.50	26.75	67.75	
B2003-1	-20	20	71	65.75	87.25	27.75	65.50	
B2003-2	-20	-20	88	61.25	81.60	30.75	70.50	
B2002-1	-	34	76	67.17	88.40	25.20	67.6	
B2002-2	-	34	75	64.55	87.15	27.65	69.8	Surveillance
B2002-3	-	39	72.5	65.32	87.32	26.30	67.0	Test Data

Notes:

1. Estimated from longitudinal data per NRC Standard Review Plan Section 5.3.2.

TABLE 4.2-9
Summary of Charpy V-notch and Drop Weight Tests

Component	Grade	30-ft-lb Fix (°F)	Drop Weight NDT (°F)
Head dome	A533B CL1	-2	10
Head peel segment	A533B CL1	-10	10
Head peel segment	A533B CL1	12	0
Upper shell plate	A533B CL1	33	-10
Upper shell plate	A533B CL1	31	-10
Upper shell plate	A533B CL1	9	-10
Intermediate shell plate	A533B CL1	14	-20
Intermediate shell plate	A533B CL1	-11	-30
Intermediate shell plate	A533B CL1	18	-10
Lower shell plate	A533B CL1	-5	-20
Lower shell plate	A533B CL1	-32	-20
Bottom peel segment	A533B CL1	-12	-20
Bottom peel segment	A533B CL1	-9	-10
Bottom dome	A533B CL1	8	-30
Head flange	A508 CL2	10	-
Vessel flange	A508 CL2	-18	-
Inlet Nozzle	A508 CL2	-102	-
Inlet Nozzle	A508 CL2	-84	-
Inlet Nozzle	A508 CL2	-95	-
Inlet Nozzle	A508 CL2	-51	-
Outlet Nozzle	A508 CL2	-32	-
Outlet Nozzle	A508 CL2	<10	-
Outlet Nozzle	A508 CL2	-45	-
Outlet Nozzle	A508 CL2	<10	-

TABLE 4.2-10
Reactor Vessel Beltline Fluence

Fast Neutron Fluence (>1 MeV)
32 Effective Full Power Years
(n/cm^2)$_1$

Reactor vessel Interior surface	1.39×10^{19}
1/4 vessel thickness (1/4 T)	9.04×10^{18}
3/4 vessel thickness (3/4 T)	3.48×10^{18}

Notes:
1. These values are calculated based upon experimental results from the measurements on the fourth surveillance capsule V. See Reference 11.

4.2 FIGURES

4.3 SYSTEM DESIGN EVALUATION

4.3.1 Safety Factors

The safety of the reactor vessel and all other reactor coolant system pressure-containing components and piping is dependent on several major factors including design and stress analysis, material selection and fabrication, quality control, and operations control.

4.3.1.1 Reactor Vessel

A stress evaluation of the reactor vessel has been carried out in accordance with the rules of Section III of the ASME Nuclear Pressure Vessel Code. The evaluation demonstrates that stress levels are within the stress limits of the code. Table 4.3-1 presents a summary of the results of the stress evaluation.

The most significant transients with regard to cumulative fatigue of the reactor vessel are loss of load transient and loss-of-flow transients. A summary of fatigue usage factors for components of the reactor vessel is given in Table 4.3-2. The effect of gamma-ray heating on the cumulative usage factor is negligible.

The cycles specified for the fatigue analysis are the results of an evaluation of the expected plant operation coupled with experience from nuclear power plants now in service. These cycles include five heatup and cooldown cycles per year, a conservative selection considering that the vessel may not complete more than one cycle per year during normal operation.

The vessel design pressure is 2485 psig, while the normal operating pressure will be 2235 psig. The resulting operating membrane stress is therefore amply below the code-allowable membrane stress to account for operating pressure transients.

To preclude the possibility of brittle failure, a reactor vessel material surveillance program that meets the requirements of 10CFR50 App. H, is implemented to monitor the change in reactor vessel materials due to neutron radiation.

The radiation induced shift in Reference Temperature nil-ductility transition (RT_{NDT}) is periodically assessed during the life of the plant by testing of vessel material samples that are irradiated cumulatively by securing them near the inside wall of the vessel in the core area. Regulatory Guide 1.99, Rev.2, “Radiation Embrittlement of Reactor Vessel Materials”, is utilized to predict the radiation induced change in the (RT_{NDT}) and calculate a new Adjusted Reference Temperature (ART). To compensate for any increase in the (RT_{NDT}) caused by irradiation, the heatup and cooldown pressure temperature limits given in the Technical Specifications are periodically changed to comply with 10CFR50, Appendix G, “Fracture Toughness Requirements”.

The vessel closure contains fifty-four 7-in. studs. The stud material has a minimum yield strength of 104,100 psi at design temperature. The membrane stress in the studs when they are at the steady state operational condition is approximately 40,000 psi. This means that 21 of the 54 studs have the capability of withstanding the hydrostatic end load on the vessel head without the membrane stress exceeding yield strength of the stud material at design temperature.

The normal operating temperature always exceeds even the highest anticipated DTT during the life of the plant. Thus the emphasis of conservative operation is placed on heatup and cooldown because long term irradiation of the vessel raises the DTT and thereby limits the heatup or cooldown rates. The conservatism in setting up the temperature-pressure relationship limits stated above are:

1. Use of a stress concentration factor of 4 on assumed flaws in calculating the stresses.
2. Use of nominal yield of material instead of actual yield.

3. Neglecting the increase in yield strength resulting from radiation effects.

4.3.1.2 Steam Generators

Calculations confirm that the steam generator tube sheet will withstand the loading (which is a quasi-static rather than a shock loading) by loss of reactor coolant.

The rupture of primary or secondary piping has been assumed to impose a maximum pressure differential of 2485 psi across the tubes and tube sheet from the primary side or a maximum pressure differential of 1035 psi across the tubes and tube sheet from the secondary side, respectively. Under these conditions there is no rupture of the primary to secondary boundary (tubes and tube sheet). This criterion prevents any violation of the containment boundary.

An examination of stresses under these conditions shows that for the case of a 2485 psi maximum tube sheet pressure differential, the stresses are within acceptable limits.

The tubes were designed to the requirements (including stress limitations) of Section III for normal operation, assuming 1700 psi as the normal operating pressure differential. Hence, the secondary pressure loss accident condition imposes no extraordinary stress on the tubes beyond that normally expected and considered in Section III requirements.

An evaluation determined the extent of tube wall thinning that could be tolerated under accident conditions. The worst-case loading conditions are assumed to be imposed upon uniformly thinned tubes at the most critical location in the steam generator. Under such a postulated design basis accident, vibration is short enough duration that there is no endurance issue to be considered.

The steam generator tubes, existing originally at their minimum wall thickness and reduced by a conservative general corrosion and erosion loss, provide an adequate safety margin (sufficient wall thickness) in addition to the minimum required for a maximum stress less than the allowable stress limit, as defined by the ASME Code.

Studies have been made on tubing of the size in the replacement steam generators under accident loadings. The results show that the maximum Level D Service condition stress due to combined pipe rupture and safe shutdown earthquake loads is less than the allowable limit. The tube thickness required to achieve the acceptable stress is less than the minimum steam generator tube wall thickness, which is reduced to account for assumed general corrosion and erosion rate. Thus, an adequate safety margin is exhibited. The general corrosion rate is based on a conservative weight-loss rate for Alloy 600 tubing in flowing, 650°F primary-side reactor coolant fluid. The estimated weight loss, based on testing when equated to a thinning rate and projected over a 60-year design objective, is much less than the assumed corrosion allowance of 3 mils. This leaves the remainder of the general corrosion allowance for thinning on the secondary side.

Potential sources of tube excitation are considered, including primary fluid flow within the U-tubes, mechanically induced vibration, and secondary fluid flow on the outside of the U-tubes. The effects of primary fluid flow and mechanically induced vibration, including those developed by the canned-motor pump, are acceptable during normal operation. The primary source of potential tube degradation due to vibration is the hydrodynamic excitation of the tubes by the secondary fluid. This area has been emphasized in both analyses and tests, including evaluation of steam generator operating experience.

Three potential tube vibration mechanisms related to hydrodynamic excitation of the tubes have been identified and evaluated. These include potential flow-induced vibrations resulting from vortex shedding, turbulence, and fluid-elastic vibration mechanisms.

Nonuniform, two-phase turbulent flow exists throughout most of the tube bundle. Therefore, vortex shedding is possible only for the outer few rows of the inlet region. Moderate tube response caused by vortex shedding is observed in some carefully controlled laboratory tests on idealized tube arrays. However, no evidence of tube response caused by vortex shedding is observed in steam generator scale model tests simulating the inlet region. Bounding calculations consistent with laboratory test parameters confirmed that vibration amplitudes would be acceptably small, even if the carefully controlled laboratory conditions were unexpectedly reproduced in the steam generator.

Flow-induced vibrations due to flow turbulence are also small. Root mean square amplitudes are less than allowances used in tube sizing. These vibrations cause stresses that are two orders of magnitude below fatigue limits for the tubing material. Therefore, neither unacceptable tube wear nor fatigue degradation due to secondary flow turbulence is anticipated.

Fluid elastic tube vibration is potentially more severe than either vortex shedding or turbulence because it is a self-excited mechanism. Relatively large tube amplitudes can feed back proportionally large tube driving forces if an instability threshold is exceeded. Tube support spacing in both the tube support plates and the anti-vibration bars in the U-bend region provides tube response frequencies such that the instability threshold is not exceeded for secondary fluid flow conditions for tubes effectively supported. This approach provides large margins against initiation of fluid elastic vibration for tubes effectively supported by the tube support system.

Small clearances between the tubes and the supporting structure are required for steam generator fabrication. These clearances introduce the potential that any given tube support location may not be totally effective in restraining tube motion if there is a finite gap around the tube at that location. Fluid-elastic tube response within available support clearances is therefore theoretically possible if secondary flow conditions exceed the instability threshold when no support is assumed at the location with a gap around the tube. This potential has been investigated both with tests and analyses for the U-bend region where secondary flow conditions have the potential to exceed the instability threshold if a tube does not contact provided supports as a result of fabrication tolerances.

Tube vibration response is shown to have wear potential within available design margins even for limiting tube fit-up conditions, based on previous experience in fabricating steam generators with fit-up control typical of the replacement steam generator. The replacement steam generator includes a number of features that minimize the potential for tube wear at tube supports. Provisions to minimize the potential for wear include the spacing between the tube supports, the configuration of the broached hole through the support plate, the surface finish of the broached hole in the tube support plate, the clearance between the tube and the hole in the tube support plate, and the tube support plate material selection.

Tube bending stresses corresponding to tube vibration response remain more than two orders of magnitude below fatigue limits as a consequence of vibration amplitudes constrained by available clearances. The analyses and tests for limiting postulated fit-up conditions include simultaneous contributions from flow turbulence.

As outlined, analyses and tests demonstrate that unacceptable tube degradation resulting from tube vibration is not expected for the replacement steam generators. Operating experience with steam generators having the same size tubes and similar flow conditions supports this conclusion.

The U-bend fatigue (discussed in NRC Bulletin 88-02) is not a consideration in the replacement steam generators. The mechanism considered in Bulletin 88-02 requires denting of the top tube support plate. But this is not expected with the stainless steel tube support plates in the replacement steam generator.

The stress limits for Service Level D that allow inelastic deformation are supplemented with the requirements of "Rules for Evaluation of Service Loadings with Level D Service Limits," Appendix F of ASME Code, Section III. The limits and rules of Appendix F confirm that pressure boundary integrity and core support structural integrity are maintained but do not confirm operability. The limits and rules of Appendix F do not apply to the portion of the component or support in which the failure has been postulated.

The structural stress analyses performed on the replacement steam generators consider the loadings specified. These loads result from thermal expansion, pressure, weight, earthquake, pipe rupture, and plant operational thermal and pressure transients. Dynamic effects of pipe rupture, including the loss of coolant accident, are not included in loading combinations when the leak-before-break criteria are satisfied.

The combination of safe shutdown earthquake plus pipe rupture loads by square-root-sum-of-the-squares is considered. The dynamic effects of pipe rupture that are combined with safe shutdown earthquake in loading combinations are combined using the square-root-sum-of-the-squares method.

The integrity of the pressure boundary of safety-related components is provided by the use of the ASME Code. The replacement steam generators, including the transition cone, lower shell, tubesheet, and channel head are constructed to the requirements of the ASME Code, Section III, 1980 Edition, plus winter 1981 addenda, which is reconciled to the design code of record, the 1965 ASME Boiler and Pressure Vessel Code, Section III, plus Addenda thru summer 1966. Using the methods and equations in the ASME Code, stress levels in the components and supports are calculated for various load combinations. These load combinations may include the effects of internal pressure, dead weight of the component and insulation, and fluid, thermal expansion, dynamic loads due to seismic motion, and other loads. The evaluation of the stress levels and fatigue usage for the steam generator pressure boundary is calculated for the specified loading conditions and demonstrates that the values are less than the allowable limits. These calculations are documented in a Stress Report as required by the ASME Code. Evaluation of the secondary shell in contact with secondary water assumes a 0.050 inch corrosion allowance. The analysis of the support plates assumes no corrosion allowance.

The ASME Code, Section III requires that a design specification be prepared for ASME components. The specification conforms to and is certified to the requirements of ASME Code, Section III. The Code also requires a design report for safety-related components, to demonstrate that the as-built component meets the requirements of the relevant ASME Design Specification and the applicable ASME Code. The design specifications and design reports will be completed by the Combined License applicant or his agent. Design specifications for ASME components and piping are prepared utilizing procedures that meet the ASME Code. The design report includes as-built reconciliation.

4.3.1.3 Piping

The reactor coolant system piping has been designed for normal and emergency conditions. For the emergency condition, the piping has been designed and analyzed for seismic loads and blowdown forces due to a loss-of-coolant accident. By design, the main piping of the reactor coolant loop is not subjected to induced pressure pulse vibrations from the reactor coolant pump impeller or from the pistons of the charging pump.

In 1989, the NRC approved changes to the design bases with respect to dynamic affects of postulated primary loop pipe ruptures, as discussed in Section 4.1.2.4.

4.3.2 Reliance On Interconnected Systems

The principal heat removal systems, which are interconnected with the reactor coolant system are the steam and power conversion, the safety injection, and residual heat removal systems. The reactor coolant system is dependent upon the steam generators, and the steam, feedwater, and condensate systems for decay heat removal from normal operating conditions to a reactor coolant temperature of approximately 350^{o}F. The layout of the system ensures the natural circulation capability to permit adequate core cooling following a loss of power to all main reactor coolant pumps. Further details are given in Section 14.1.6.1.

The NRC reviewed the Indian Point 2 response to issues concerning natural circulation cooldown in their safety evaluation report dated August 1, 1983 (reference 20), and determined that Con Edison met the requirements of Generic Letter 81-21.

The flow diagram of the steam and power conversion system is shown on Plant Drawings 227780, 9321-2017, 235308 [Formerly UFSAR Figure 10.2-1 sheets 1 to 3], 9321-2025 [Formerly UFSAR Figure 10.2-4], 9321-2018 and 235307 [Formerly UFSAR Figure 10.2-5 sheets 1 and 2], and 9321-2019 [Formerly UFSAR Figure 10.2-7]. In the event that the condensers are not available to receive the steam generated by residual heat, the water stored in the feedwater system may be pumped into the steam generators and the resultant steam vented to the atmosphere. The auxiliary feedwater system will supply water to the steam generators in the event that the main feedwater system is unavailable.

The safety injection system is described in Section 6.2. The residual heat removal system is described in Section 9.3.

4.3.3 System Integrity

A complete stress analysis that reflects consideration of all design loadings detailed in the design specification has been prepared by the manufacturer. The analysis shows that the reactor vessel, steam generator, reactor coolant pump casing, and pressurizer comply with the stress limits of Section III of the ASME Code. A similar analysis of the piping shows that it complies with the stress limits of the applicable USAS Code.

As part of the design control on materials, Charpy V-notch toughness test curves were run on all ferritic material used in fabricating pressure parts of the reactor vessel, steam generator, and pressurizer to provide assurance for hydrotesting and operation in the ductile region at all times. In addition, drop weight tests were performed on the reactor vessel plate material. As an

assurance of system integrity, all components in the system were hydrotested at 3110 psig prior to initial operation.

4.3.4 Overpressure Protection

The reactor coolant system is protected against overpressure by safety valves located on the top of the pressurizer. The safety valves on the pressurizer are sized to prevent system pressure from exceeding the design pressure by more than 10-percent, in accordance with Section III of the ASME Boiler and Pressure Vessel Code. The capacity of the pressurizer safety valves is determined from considerations of: (1) the reactor protective system, and (2) accident or transient conditions, which may potentially cause overpressure.

The combined capacity of the safety valves is equal to or greater than the maximum surge rate resulting from complete loss of load without a direct reactor trip or any other control, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve setting.

Details of the analysis are reported in Section 14.1.8. Experience has shown that the safety valve capacity so determined is adequate for all the other transients as the results of Section 14.1 show.

The report "Summary Report of Safety and Relief Valve Installation and Re-Analysis for ASME Class 1 and Class 2 Systems in Indian Point Unit No. 2" (Reference 5) describes the general scope, design and installation criteria, significant assumptions, methods of analysis, and maximum combined stresses for those applicable safety and relief valves in the reactor coolant system, main steam system, chemical and volume control system, safety injection system, component cooling water system, and service water system.

In response to NUREG-0737 Section II.D.1, a test program for the pressurizer safety and relief valves was formulated by the Electric Power Research Institute to provide full-scale test data confirming the functional ability of the reactor coolant system power-operated relief valves and safety valves for expected operating and accident conditions, and to obtain sufficient piping thermal-hydraulic load data to permit confirmation of models that may be used for plant-unique analysis of safety and relief discharge piping systems. The Indian Point 2 plant-specific evaluations regarding this generic issue are contained in Con Edison submittals to NRC dated July 1, 1982, September 15, 1982, June 15, 1984, June 14, 1985 and October 18, 1985. This program satisfied the requirements of NUREG-0737, as documented in NRC's Safety Evaluation Report (SER) dated August 5, 1987 (Reference 16).

Item II.K.3.2 of NUREG-0737 required licensees of pressurized water reactors to submit a report to the NRC staff documenting the various actions taken to decrease the probability of a small break LOCA caused by a stuck-open PORV and show how these actions constitute sufficient improvements to safety. Based upon the results of the report submitted in response to item II.K.3.2, licensees were to assess whether an automatic PORV isolation system was required. If required, licensees were to submit a system design that uses the PORV block valve to automatically protect against a small break LOCA caused by a stuck open PORV.

The Westinghouse Owners Group submitted a generic report to the NRC staff in response to Item II.K.3.2 (Reference 17). Con Edison's response to the NRC on this matter (Reference 18) adopted the conclusions reached in the aforementioned report as applicable to IP2, namely that the concept of an automatic PORV block valve closure system cannot be warranted on the

basis of providing additional protection against a PORV LOCA. On this basis, Con Edison proposed no modifications to provide automatic isolation of the PORVs.

The NRC reviewed Con Edison's submittal and found that the requirements of NUREG-0737, Item II.K.3.2 were met with the existing PORV, safety valve and reactor high pressure trip setpoints and that an automatic PORV isolation system was not required for IP2 (Reference 19).

4.3.4.1 Reactor Coolant System Overpressure Protection System

An overpressure protection system to prevent reactor coolant system pressure exceeding the 10 CFR 50 Appendix G curves has been installed. It is a three-channel, analog, curve-tracking arrangement, which would initiate an appropriate chain of coincidence logic for the purpose of automatically preventing a violation of the operating Technical Specifications temperature/pressure curves for the reactor vessel.

In order to develop the overpressure protection system setpoint limit curve for the Technical Specifications, heatup and cooldown limit curves are calculated (Ref.21), using the adjusted RT_{NDT} (reference nil-ductility temperature) corresponding to the limiting beltline region material of the reactor vessel. The adjusted RT_{NDT} (ART) of the limiting material in the core region of the reactor vessel is determined by using unirradiated reactor vessel material fracture toughness properties, estimating the radiation induced change in RT_{NDT}, and adding margin for uncertainty. The unirradiated RT_{NDT} is designed as the higher of either the drop weight nil ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35 mils lateral expansion (transverse to the primary working direction), less 60F degrees. The method used to calculate the ART values at 1/4T and 3/4T locations, (where T is the thickness of the reactor vessel at the beltline region not including the cladding), complies with Nuclear Regulatory Commission Regulatory Guide 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials”.

The heatup and cooldown curves are generated using the most limiting ART values and the methodology documented in Westinghouse Report WCAP-14040-NP-A, Rev.2, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown limit curves”, with the following exceptions:

1) The fluence values used are calculated fluence values, rather than best estimate fluence values.
2) The Kic critical stress intensities are used in place of Kia critical stress intensities, in compliance with ASME Code Case N-640,
3) The 1996 version of Appendix G to ASME Section XI is used instead of the 1989 version, and
4) Pressure-temperature limit curves were generated with the most limiting circumferential weld ART in conjunction with Code Case N-588. These curves are bounded by curves using the standard axial flaw methodology of the ASME Code 1996, App. G with the ART from the limiting plate material.

The heatup and cooldown pressure-temperature limit curves (“10CFR50 App. G limits”) so obtained (Ref.21) are valid for 29.2 EFPY.

Thermal-hydraulic analysis (Ref.22) accounting for the effects of pressure bias and pressure overshoot during mass and heat input transients, is utilized to develop setpoint curves to actuate the Power Operated Relief Valves (PORVs) and prevent the reactor coolant system from exceeding the 10 CFR50 App. G limits. The analysis demonstrates that a single power operated relief valve is capable of mitigating the worst possible mass or heat input transient,

thereby ensuring that the peak reactor coolant system pressure for the Indian Point Unit No. 2 remains below the 10 CFR50 Appendix G limits were such a transient to occur. Overpressure Protection System setpoint curves are further adjusted for instrument error, and these latter curves are utilized as heatup and cooldown limits in plant operating procedures. Also, additional administrative controls are utilized to protect the Residual Heat Removal System from reactor coolant overpressurization events.

The overpressure protection system does not change the primary system operation or relief system operation during normal plant operation. The system allows for more close control of system heatup and cooldown through more accurate instrumentation and monitoring. Spurious opening and/or closing of the power operated relief valves is essentially eliminated by the new two-out-of-three logic (if one channel were to fail the valve would not malfunction). Thus, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated is not increased.

The overpressure protection system maintains the existing operational function of all existing plant components. There is an expansion of certain component functions to enhance the controllability of primary system pressure during heatup and cooldown. Inasmuch as there is better control of existing plant components and no change to their operation, the possibility for an accident or malfunction of a different type than any evaluated previously is not increased.

The overpressure protection system allows heatup and cooldown guidelines to be strictly followed both by automatic and manual means, thereby reducing the possibility of violating significant parameters and maintain an orderly heatup and cooldown. Thus the margin of safety as defined in the bases for the facility Technical Specifications is not reduced.

NRC acceptance of the Indian Point 2 low temperature overpressure protection system and the relevant Consolidated Edison submittals are contained in References 12, 13, 14 and 15.

4.3.4.2 <u>Nitrogen System</u>

A nitrogen system actuates the power operated relief valves (PORVs) PCV-455C and PCV-456. The PORV nitrogen system is tapped from the existing nitrogen supply header to Safety Injection (SI) system accumulators at a location downstream of pressure regulator valve PCV-942 and relief valve RV-1816, which is set at 1100 psig. The nitrogen pressure to the PORVs is reduced to 100 psig by pressure regulator valves PRV-3100 and PRV-3101. Containment isolation of the PORV nitrogen system is provided by valves 4312 and 863.

The instrument nitrogen system includes two accumulators, each holding approximately 13-ft^3 of nitrogen. In case the nitrogen supply is lost, these accumulators, with a minimum initial pressure of 600 psig, can support cycling (full open/close) the power operated relief valves for a minimum of 10 minutes.

The nitrogen system is provided with pressure indicating alarms located on the SKF panel in the control room to provide information to the operator in case of low pressure in the nitrogen accumulators. Pressure alarms also on the SKF panel provide indication for nitrogen supply regulator malfunction.

The PORV nitrogen system is designated Class A, Seismic Class I. The accumulators are designed to ASME Section VIII, Div.1 and piping is designed to ANSI B31.1. The PORV nitrogen system piping can withstand 1100 psi, the relief setting of valve RV-1816. The design

of the PORV nitrogen system further considered the potential to generate missiles. To ensure that none of the components of the nitrogen system would become a source of missiles, the valves are forged, have a bolted clamseal bonnet and have stems which back seat. This rules out the possibility of ejecting valve stems as, even if it were assumed that the stem threads fail, the back seat or the upset end cannot penetrate the bonnet and thereby become a missile.

Also, the valves have been designed against bonnet-body connection failure and subsequent bonnet ejection by means of (1) using the design practice of ASME Section VIII, which limits the allowable stress of bolting material to less than 20-percent of its yield strength, (2) using the design practice of ASME Section VIII for flange design, and (3) by controlling the load during the bonnet-body connection stud tightening process. The pressure-containing parts except the flange and studs are designed per criteria established by USAS B16.5. Flanges and studs, where used, are designed in accordance with ASME Section VIII.

4.3.4.3 Evaluation of the Overpressure Protection System

With the overpressure protection system enabled, the power-operated relief valves will open automatically to prevent the reactor pressure vessel pressure from exceeding the Appendix G limits during a temperature range and for the effective full power years as defined in the Indian Point Unit 2 Technical Specifications, and there is a pressure excursion over the setpoint. With the Overpressure Protection System enabled, the power-operated relief valve isolation motor-operated valves are in the open position. Existing wide-range cold leg reactor coolant system temperature signals (TE-413, 433, and 443) are designed to perform two primary functions in this system: (1) provide the arming and disarming function and (2) serve as the independent variable in computing the reference Appendix G limit to which the system pressure limit must be adhered.

The arming function is initiated when the reactor coolant system temperature falls below a temperature defined by the Technical Specification. At the OPS enable temperature, the motor-operated valves (MOV-535 and 536) on the pressurizer will either be manually or automatically opened and the overpressure protection system logic system will be armed to prevent a possible overpressurization condition. Also, one half of a two-out-of-two coincidence logic will be satisfied to allow the relief valves to open in the event of an impending overpressure condition.

These same temperature signals are also fed into three respective function generators whose task is to output values of pressure as a function of the input temperature, which are the maximum reactor coolant system pressures (Appendix G limit pressures) allowed at those temperatures. The difference between these maximum permissible reactor coolant system pressures and the actual reactor coolant system pressure, transmitted by 0 to 1500-psig transmitters (PT-413, 433, 443), is computed in each of the three channels, and if any two-out-of-three of these differences is smaller than a preset minimum, a trip open condition will be initiated for each pressurizer power operated relief valve designated as train "A" (MOV-536 and PCV-456) and train "B" (MOV-535 and PCV-455C).

Various alarms and lights related to reactor coolant system overpressurization arming, actuation or nonavailability of train "A" or "B" are located in the SGF and FB panels.

The alarms to indicate arming of the reactor coolant system overpressurization trains and actuation of the reactor coolant system overpressurization train "A" and train "B" are located on the SG panel. The motor-operated valves can be closed in the armed region by putting the motor-operated valves selector switch into the full locked position. White lights (one for each

train), which indicate that the reactor coolant system overpressurization train is not available are located on the FB panel above the control switches.

As a protection against a common air supply failure causing inoperability of both power operated relief valves, the air system has been replaced by a nitrogen system with accumulators to supply each valve. (See description of nitrogen supply system, Section 4.3.4.2.) The electrical supply is from the 125VDC power panels, which are supplied by 480VAC through 125VDC battery chargers with backup by the station emergency batteries. The electrical activation uses two-out-of-three logic for valve actuation.

Manual disconnect switches provide a means to interrupt the power to a SOV, which will then result in the closure of the associated PORV. Operation with the switches closed permits the PORVs to open or close automatically or be manually operated to perform their pressure relief function. In the event of a fire in most of the fire zones in Fire Area A, these switches can be manually opened to prevent or mitigate the spurious opening of the PORVs due to a hot short in the control circuitry. To ensure that the PORV's can perform their pressure relief function, the block valves are interlocked to open automatically when the pressurizer pressure reaches a preset limit below the pressure at which the PORVs open. In addition, the PORV actuation and reclosure setpoint calibration is checked each 24 months. Operation with the PORVs and block valves closed will prevent the spurious opening of both a PORV and its associated block valve in the event of a fire in certain fire zones.

4.3.5 Incident Potential

The potential of the reactor coolant system as a cause of accidents is evaluated by investigating the consequences of certain credible types of components and control failures as discussed in Sections 14.1 and 14.2. Reactor coolant pipe rupture is evaluated in Section 14.3.

4.3.6 Redundancy

Each loop of the reactor coolant system contains a steam generator and a reactor coolant pump. Operation at reduced reactor power is possible with one loop out of service as limited by the facility Technical Specifications. For added reliability, power to the reactor coolant pumps is normally supplied by electrically separated buses as shown in Plant Drawing 231592 [Formerly UFSAR Figure 8.2-5]. The remote reactor head vent valves and the power operated relief valves and block valves are supplied with diverse and independent emergency power sources as described in Section 4.2.10.

REFERENCES FOR SECTION 4.3

1. H. Kihara and K. Masubuchi, "Effect of Residual Stress on Brittle Fracture," Welding Journal, Volume 38, April 1959.

2. Robertson, "Propagation From Brittle Fracture in Steel," Journal of the Iron and Steel Institute, 1953.

3. L. Porse, "Reactor Vessel Design Considering Radiation Effects," ASME Paper Number 63-WA-100.

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5. Letter from W. J. Cahill, Con Edison, to J. F. O'Leary, NRC, Subject: "Summary Report of Safety and Relief Valve Installation and Re-Analysis of ASME Class 1 and Class 2 Systems in Indian Point Unit 2," dated July 13, 1972.

6. Deleted

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9. Westinghouse Electric Corporation, "Consolidated Edison Company of New York, Inc., Indian Point Unit 2 NSSS Stretch Rating - 3083.4 MWT Engineering Report," WCAP 12187 (Proprietary).

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12. Letter (with attachments) from A. Schwencer, NRC to W. Butler, NRC, Subject: Safety Evaluation of the Electrical, Instrumentation and Control (EI&C) Aspects of the Low Temperature Overpressure Protection System, dated June 14, 1978.

13. Letter (with attachments) S. Varga, NRC to J. O’Toole, Con Edison, Subject: Indian Point 2- Low Temperature Overpressure Protection System, dated April 24, 1984.

14. Letter (with attachments) S. Varga, NRC to J. O’Toole, Con Edison, Subject: Low Temperature Overpressure Protection System at Indian Point Nuclear Generating Plant, Unit 2, dated June 28, 1984.

15. Letter (with attachments) J. O’Toole, Con Edison to S. Varga, NRC, Subject: Response to NRC Safety Evaluation Report, dated July 17, 1984.

16. Letter (with attachments) from M. Slosson, NRC to M. Selman, Con Edison, Subject: Safety Evaluation Report, NUREG-0737, Item II.D.1, Performance Testing of Relief and Safety Valves for Indian Point Nuclear Generating Unit No. 2, dated August 5, 1987.

17. WCAP-9804, "Probabilistic Analysis and Operational Data is Response to NUREG-0737, Item II.K.3.2, for Westinghouse NSSS Plants", Westinghouse Electric Corporation, February 1981

18. Letter (with attachments) from J. D. O'Toole, Con Edison, to D. G. Eisenhut, NRC, Response to NUREG-0737, Clarification of TMI Action Plan Requirements, Dated February 26, 1981

19. Letter (with attachments) from S. A. Varga, NRC, to J. D. O'Toole, Con Edison, Subject: NUREG-0737 Items II.K.3.1 - Automatic PORV Isolation and II.K.3.2 - Report on PORVs the Indian Point Nuclear Generating Plant, Dated September 13, 1983

20. Letter (with attachment) from S.A. Varga, NRC, to J.D. O'Toole, Con Edison, Subject: Natural Circulation Cooldown For The Indian Point Nuclear Generating Plant, Unit No. 2 (IP-2), Dated August 1, 1983.

21. Westinghouse Electric Company Report: WCAP16752-NP, "Indian Point Unit 2 Heatup and Cooldown Limit Curves for Normal Operation, January 2008.

22. Con Edison Calculation No. FMX-00270-01, "Indian Point Unit 2 Overpressure Protection System (OPS) Thermal Hydraulic Analysis, Setpoint Development and Technical Specification Revision February 26, 2009.

TABLE 4.3-1
Summary of Primary Plus Secondary Stress Intensity
for Components of the Reactor Vessel

Area	Stress Intensity (psi)	Allowable Stress 3Sm(psi) (Operating Temperature)
Control rod housing	77,700 (1)	69,900
Head flange	45,370	80,100
Vessel flange	52,140	80,100
Closure studs	109,400	110,400
Primary nozzles – inlet outlet	45,500 49,390	80,100 80,100
Core support pad	55,280	69,900
Bottom head to shell	34,100	80,100
Bottom instrumentation	55,500	69,900
Nozzle belt to shell	37,900	80,100
Head Adapter Plugs	27,630	48,600

Note:
1. A simplified elastic plastic analysis was performed to justify exceeding the $3S_m$ limit.

TABLE 4.3-2
Summary of Cumulative Fatigue Usage Factors for
Components of the Reactor Vessel

Item	Usage Factor$_1$
Control rod housing	0.01
Head flange	0.0107
Vessel flange	0.0229
Stud bolts	0.944
Primary nozzles - inlet	0.050
outlet	0.281
Core support pad (lateral)	0.904
Bottom head to shell	0.004
Bottom instrumentation	0.201
Nozzle belt to shell	0.0029
Head Adapter Plugs	0.0036

Notes:
1. As defined in Section III of the 1965 ASME Boiler and Pressure Vessel Code, Nuclear Vessels.

TABLE 4.3-3
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TABLE 4.3-4
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4.4 SAFETY LIMITS AND CONDITIONS

4.4.1 System Heatup And Cooldown Rates

Operating limits for the reactor coolant system with respect to heatup and cooldown rates are defined in the Technical Specifications.

The stress level of material in the reactor vessel, or in other reactor coolant system components, is a combination of stresses caused by internal pressures and by thermal gradients. The latter are significant as they may result from a rate of change of reactor coolant temperature. Operating restrictions are imposed to limit the combined stresses to 20-percent of minimum yield stress when at the design transition temperature (DTT). The DTT is defined as the initial nil-ductility transition temperature (NDTT) plus the increase in NDTT due to irradiation experienced, plus 60^{o}F. This stress limit (20-percent of yield strength) is reduced linearly to a value of 10-percent of yield at a temperature 200^{o}F below DTT. Curves are incorporated in the plant operating procedures, which define the operating limits for initial operation and for end of life operation. To establish the latter, an adjustment is made for the maximum expected NDTT shift (240^{o}F), which the reactor vessel material will experience because of the fast neutron dose it will receive. The predicted shift will be verified by the surveillance program testing. The limits for initial operation are used to define operational limitations, and these curves are periodically updated to reflect irradiation exposure of the vessel and the results of the surveillance program.

4.4.2 Reactor Coolant Activity Limits

The plant systems are designed for operation with activity in the reactor coolant systems corresponding to 1-percent fuel defects. The accident analyses presented in Chapter 14 include the calculation of doses resulting from the release of activity initially contained in the primary system. The reactor coolant system operational activity limit is defined in the Technical Specifications.

4.4.3 Maximum Pressure

The reactor coolant system serves as a barrier preventing radionuclides contained in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the reactor coolant system is the primary barrier against the uncontrolled release of fission products. By establishing a system pressure limit, the continued integrity of the reactor coolant system is assured. Thus, the safety limit of 2735 psig (110-percent of design pressure) has been established. This represents the maximum transient pressure allowable in the reactor coolant system under the ASME Code, Section III. Reactor coolant system pressure settings are given in Table 4.1-1.

4.4.4 System Minimum Operating Conditions

Minimum operating conditions for the reactor coolant system for all phases of operation are given in the Technical Specifications.

4.5 INSPECTIONS AND TESTS

4.5.1 Inspection Of Materials And Components Prior To Operation

Table 4.5-1 summarizes the quality assurance program for all reactor coolant system components. In this table all of the nondestructive tests and inspections, which were required by Westinghouse specifications on reactor coolant system components and materials are specified for each component. All tests required by the applicable codes are included in this table. Westinghouse requirements, which are more stringent in some areas than those requirements specified in the applicable codes, are also included. The fabrication and quality control techniques used in the fabrication of the reactor coolant system are equivalent to those used for the reactor vessel.

Westinghouse required, as part of its reactor vessel specification, that certain special tests that were not specified by the applicable codes be performed. These tests are listed below:

1. Ultrasonic testing - Westinghouse required that a 100-percent volumetric ultrasonic test of reactor vessel plate for both shear wave and longitudinal wave be performed. Section III Class A vessel plates were required by code to receive only a longitudinal wave ultrasonic test on a 9-in. x 9-in. grid. The 100-percent volumetric ultrasonic test is a severe requirement, but it ensured that the plate is of the highest quality.

2. Radiation surveillance program - In the surveillance programs, the evaluation of the radiation damage is based upon pre-irradiation and post-irradiation testing of Charpy V-notch, tensile, and wedge opening loading fracture mechanism type.

4.5.2 Reactor Vessel Surveillance Program

This program is directed toward evaluation of the effects of radiation on the fracture toughness of reactor vessel steels based on the transition temperature and fracture mechanics approaches, and is in accordance with ASTM E-185, "Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors."

The reactor vessel surveillance program uses eight specimen capsules, which are located about 3-in. from the vessel wall directly opposite the center portion of the core. The capsules can be removed when the vessel head is removed. The capsules contain reactor vessel steel specimens from the shell plates and forgings located in the core region of the reactor, associated weld metal, and heat affected zone metal. In addition, correlation monitors made from fully documented specimens of SA302 Grade B material obtained through Subcommittee II of ASTM Committee E10, Radioisotopes and Radiation Effects, are inserted in the capsules. The 8 capsules contain at least 27 tensile specimens, 256 Charpy V-notch specimens (which will include weld metal and heat-affected zone material), and 42 wedge opening loading specimens. Dosimeters including pure Ni, Al-Co (0.15-percent Co), Cd shielded Al-Co, Cd0 shielded Np-237, and Cd0 shielded U-238 are placed in the impact specimens, tensile specimens, or filler blocks drilled to contain the dosimeters. The dosimeters permit evaluation of the flux seen by the specimens and vessel wall. In addition, thermal monitors made of low melting alloys are included to monitor temperature of the specimens. The specimens are enclosed in a tight fitting stainless steel sheath to prevent corrosion.

Irradiation of the specimens will be higher than the irradiation of the vessel because the specimens are located in the vicinity of the core corners and are closer to the core than the vessel itself. Since these specimens will experience higher irradiation and are actual samples from the materials used in the vessel, the nil-ductility transition temperature (NDTT) measurements will be representative of the vessel at a later time in life. Data from fracture toughness samples (wedge opening loading specimens) are expected to provide additional information for use in determining allowable stresses for irradiated material.

The Indian Point Unit 2 reactor vessel surveillance program was developed on the requirements provided in ASTM E-185 in effect at the time of construction. The details of the program are provided in WCAP-7323, "Consolidated Edison Co., Indian Point Unit No. 2 Reactor Vessel Surveillance Program", Dated May 1969. The requirements of this program, currently form the basis for the reactor vessel surveillance program, as modified by the requirements of 10CFR50, Appendix H which state that the "… test procedures and reporting requirements must meet the requirements of ASTM E-185-82 to the extent practicable for the configuration of the specimens in the capsule."

The following is a list of the surveillance program capsules along with the actual (past) and anticipated (future) withdrawal schedule based on the latest fluence and embrittlement calculations performed in accordance with the requirements of Regulatory Guide 1.99, Revision 2 (WCAP-15629).

Capsule	Location	Lead Factor	Withdrawal Date
T	320°	3.42	End of Cycle 1
Y	220°	3.48	End of Cycle 2
Z	40°	3.53	End of Cycle 5
V	4°	1.18	End of Cycle 8
S	140°	3.5	Retired in Place**
U*	176°	1.2	Spare
W*	184°	1.2	End of Life***
X*	356°	1.2	Spare

*The withdrawal schedule of these capsules is interchangeable due to common materials and lead factors.
**Capsule S may be withdrawn during the RFO 19 if modified tooling capable of removing the capsule is available. If not withdrawn, no capsule is required. If withdrawn, testing will be coordinated with industry to optimize the usefulness of the test data.
***At the end of life as currently licensed, Capsule W (or U or X) will be withdrawn.

Results of Surveillance Capsule analyses are discussed in Section 4.2.5.

4.5.3 <u>Primary System Quality Assurance Program</u>

Table 4.5-1 summarizes the quality assurance program with regard to inspections performed on primary system components. In addition to the inspections shown in Table 4.5-1, there are those that the equipment supplier performed to confirm the adequacy of material he received and those performed by the material manufacturer in producing the basic material. The inspections of reactor vessel, pressurizer, and steam generator are governed by ASME Code requirements. The inspection procedures and acceptance standards required on pipe materials and piping fabrication are governed by USAS B31.1 and Westinghouse requirements, and are equivalent to those performed on ASME coded vessels. Procedures for performing the

examinations are consistent with those established in the ASME Code Section III and are reviewed by qualified Westinghouse engineers. These procedures were developed to provide the highest assurance of quality material and fabrication. They considered not only the size of the flaws, but equally as important, how the material was fabricated, the orientation and type of possible flaws, and the areas of most severe service conditions. In addition, the surfaces most subject to damage as a result of the heat treating, rolling, forging, forming, and fabricating processes received a 100-percent surface inspection by magnetic particle or liquid penetrant testing after all these operations were completed, although flaws in plates are inherently laminations in the center. All reactor coolant plate material was subject to shear as well as longitudinal ultrasonic testing to give maximum assurance of quality. All forgings received the same inspection. In addition, 100-percent of the material volume was covered in these tests as an added assurance over the grid basis required in the code.

Westinghouse quality control engineers monitored the supplier's work, witnessing key inspections not only in the supplier shop but in the shops of subvendors of the major forgings and plate material. Normal surveillance included verification of records of material, physical and chemical properties, required tests, and qualification of supplier personnel. An independent surveillance of the conformance to the fabrication and installation specifications and the quality control requirements of, among other things, the reactor coolant system components, was carried out by the United States Testing Company for Con Edison.

Equipment specifications for fabrication required that suppliers submit the manufacturing procedures (welding, heat treating, etc.) to Westinghouse where they were reviewed by qualified Westinghouse engineers. This also was done on the field fabrication procedures to ensure that installation welds were of equal quality.

Con Edison engineers witnessed the hydrostatic test of the reactor vessel.

Cleaning of reactor coolant system piping and equipment was accomplished before and/or during erection of various equipment. Stainless steel piping was cleaned in sections as specific portions of the systems were erected. Pipe and units large enough to permit entry by personnel were cleaned by locally applying approved solvents (Stoddard solvent, acetone, and alcohol) and demineralized water, and by using a rotary disc sander or 18-8 wire brush to remove all trapped foreign particles.

Section III of the ASME Boiler and Pressure Vessel Code requires that nozzles carrying significant external loads shall be attached to the shell by full penetration welds. This requirement has been carried out in the reactor coolant piping where all auxiliary pipe connections to the reactor coolant loop were made using full penetration welds.

The reactor coolant system components were welded under procedures, which required the use of both preheat and postheat. Preheat requirements, not mandatory under code rules, were performed on all weldments, including P1 and P3 materials, which are the materials of construction in the reactor vessel, pressurizer, and steam generators. Preheat and postheat of weldments both served a common purpose: the production of tough, ductile metallurgical structures in the completed weldment. Preheating produces tough ductile welds by minimizing the formation of hard zones, whereas postheating achieves this by tempering any hard zones, which may have formed due to rapid cooling.

4.5.4 Inservice Inspection Considerations

The inservice inspection and testing program is discussed in Chapter 1.

4.5.5 Reactor Coolant System Surveillance

A preoperational and inservice structural surveillance program for the reactor vessel and reactor coolant system boundary was originally established as part of the Indian Point Unit 2 initial plant conditions. This program was designed to ensure the continued integrity of the reactor coolant system boundary and included specifications, as follows:

1. Prior to initial plant operation, an ultrasonic survey was made of reactor vessel shell welds, vessel nozzles, vessel flange welds, piping system butt welds, and major welds on the pressurizer, steam generator, coolant piping and components to establish preoperational system integrity, and establish baseline data.

2. An inspection interval of 10 years was established.

3. Postoperational nondestructive inspections were provided for. The results obtained from compliance with this specification were to be evaluated after 5 years, and the conclusions of this evaluation reviewed with the NRC.

4. The structural integrity of the reactor coolant system boundary was to be maintained throughout the life of the plant at the level required by the original acceptance standards. Any evidence as a result of the inspections that defects have initiated or grown, were to be investigated, including evaluation of comparable areas of the reactor coolant system.

5. The following definitions apply to the nondestructive inspection methods.

    a. UT - Volumetric examination using ultrasonic techniques.

    b. RT - Volumetric examination using radiography.

    c. PT - Surface examination using liquid penetrant methods.

    d. V - Visual examination by direct vision or by means of remote viewing devices.

    e. IV - Indirect visual examination performed during periods when the reactor coolant system is subjected to hydrostatic test pressure.

6. Detailed records of each inspection shall be maintained to allow comparison and evaluation of future inspections.

Current requirements for the primary system surveillance program are discussed in Section 5.5.6 of the facility Technical Specifications and in the Inservice Inspection and Testing Program, Chapter 1.

During the first ten year inspection of the reactor vessel, an indication was discovered in a longitudinal weld in the lower shell course. While the NRC in their October 16, 1984 safety

evaluation concurred that the size of the indication was acceptable for plant operation, they required an augmented inspection program for the reactor vessel, which was incorporated into the Technical Specifications. By safety evaluation dated July 12, 1988, the NRC concluded that the required augmented inspection could be discontinued.

In addition, inservice surveillance of the steam generator tubes that are part of the primary coolant pressure boundary is detailed in Section 5.5.7 of the Technical Specifications. This surveillance program is to ensure their continued integrity and includes inspection requirements, corrective measures, reports, and NRC approval as a condition for plant operability. This program for inservice inspection of steam generator tubes exceeds the requirements of Regulatory Guide 1.83, Revision 1, July 1975.

4.5.6 Reactor Coolant Vent System Testing

The testing of the remote reactor head vent and power operated relief valves system valves is performed in accordance with ASME Code Section XI requirements for Category B valves.

TABLE 4.5-1 (Sheet 1 of 6)
Reactor Coolant System Quality Assurance Program

Component	RT_1	UT_2	PT_3	MT_4	ET_5
1. Steam generator					
1.1 Tube sheet					
1.1.1 Forging		Yes		Yes	
1.1.2 Cladding		Yes_6	Yes_7		
1.2 Channel head					
1.2.1 Casting	Yes			Yes	
1.2.2 Cladding			Yes		
1.3 Secondary shell and head					
1.3.1 plates		Yes			
1.4 Tubes	Yes			Yes	
1.5 Nozzles (forgings)		Yes		Yes	
1.6 Weldments					
1.6.1 Shell, longitudinal	Yes			Yes	
1.6.2 Shell, circumferential	Yes			Yes	
1.6.3 Cladding (channel head-tube sheet joint cladding restoration)			Yes		
1.6.4 Steam and feedwater nozzles to shell	Yes			Yes	
1.6.5 Support brackets				Yes	
1.6.6 Tube to tube sheet			Yes		

TABLE 4.5-1 (Sheet 2 of 6)
Reactor Coolant System Quality Assurance Program

Component	RT1	UT2	PT3	MT4	ET5
1.6.7 Instrument connections (primary and secondary)				Yes	
1.6.8 Temporary attachments after removal				Yes	
1.6.9 After hydrostatic test (all welds and complete channel head)				Yes	
1.6.10 Nozzle safe ends (if forgings)	Yes		Yes		
1.6.11 Nozzle safe ends (if weld deposit)			Yes		
2. Pressurizer					
2.1 Heads					
2.1.1 Casting	Yes			Yes	
2.2.2 Cladding			Yes		
2.2 Shell					
2.2.1 Plates		Yes		Yes	
2.2.2 Cladding			Yes		
2.3 Heaters					
2.3.1 Tubing8		Yes	Yes		
2.3.2 Centering of element	Yes				

TABLE 4.5-1 (Sheet 3 of 6)
Reactor Coolant System Quality Assurance Program

Component	RT_1	UT_2	PT_3	MT_4	ET_5
2.4 Nozzle	Yes	Yes			
2.5 Weldments					
2.5.1 Shell, longitudinal	Yes			Yes	
2.5.2 Shell, circumferential	Yes			Yes	
2.5.3 Cladding			Yes		
2.5.4 Nozzle safe end (if forging)	Yes		Yes		
2.5.5 Nozzle safe end (if weld deposit)			Yes		
2.5.6 Instrument connections			Yes		
2.5.7 Support skirt				Yes	
2.5.8 Temporary attachments after removal				Yes	
2.5.9 All welds and cast heads after hydrostatic test				Yes	
2.6 Final Assembly					
2.6.1 All accessible weld surfaces after hydrostatic test				Yes	
3. Primary Coolant Piping					
3.1 Fittings (castings)	Yes		Yes		
3.2 Fittings (forgings)		Yes	Yes		

TABLE 4.5-1 (Sheet 4 of 6)
Reactor Coolant System Quality Assurance Program

Component	RT_1	UT_2	PT_3	MT_4	ET_5
3.3 Pipe[9]	Yes	Yes			
3.4 Weldments					
3.4.1 Circumferential	Yes		Yes		
3.4.2 Nozzle to run pipe (no RT for nozzles less than 3-in.)	Yes		Yes		
3.4.3 Instrument connections			Yes		
4. Pumps					
4.1 Casting	Yes		Yes		
4.2 Forgings		Yes	Yes		
4.2.1 Main shaft		Yes	Yes		
4.2.2 Main studs		Yes	Yes		
4.2.3 Flywheel (rolled plate)		Yes			
4.3 Weldments					
4.3.1 Circumferential	Yes		Yes		
4.3.2 Instrument connections			Yes		
5. Reactor Vessel					
5.1 Forgings					
5.1.1 Flanges		Yes		Yes	
5.1.2 Studs		Yes		Yes	

TABLE 4.5-1 (Sheet 5 of 6)
Reactor Coolant System Quality Assurance Program

Component	RT[1]	UT[2]	PT[3]	MT[4]	ET[5]
5.1.3 Head adapters		Yes	Yes		
5.1.4 Head adapter tube		Yes	Yes		
5.1.5 Instrumentation tube		Yes	Yes		
5.1.6 Main nozzles		Yes		Yes	
5.1.7 Nozzle safe ends (if forging is employed)		Yes	Yes		
5.2 Plates	Yes		Yes		
5.3 Weldments					
5.3.1 Main seam	Yes			Yes	
5.3.2 CRD head adapter connection			Yes		
5.3.3 Instrumentation tube connection			Yes		
5.3.4 Main nozzles	Yes			Yes	
5.3.5 Cladding		Yes[10]		Yes	
5.3.6 Nozzle-safe ends (if forging)	Yes		Yes		
5.3.7 Nozzle safe ends (if weld deposits)	Yes		Yes		
5.3.8 Head adaptor forging to head adaptor tube	Yes		Yes		
5.3.9 All welds after hydrotest				Yes	

TABLE 4.5-1 (Sheet 6 of 6)
Reactor Coolant System Quality Assurance Program

Component	RT_1	UT_2	PT_3	MT_4	ET_5
6. Valves					
6.1 Castings	Yes		Yes		
6.2 Forgings (No UT for valves two inches and smaller)	Yes	Yes			

Notes:
1. RT - Radiographic.
2. UT - Ultrasonic.
3. PT - Dye Penetrant.
4. MT - Magnetic Particle.
5. ET - Eddy Current.
6. Flat Surfaces Only.
7. Weld Deposit Areas Only.
8. Or a UT and ET.
9. Except pressurizer surge line - UT only.
10. UT of Clad Bond-to-Base Metal.

4.6 METAL IMPACT MONITORING SYSTEM

4.6.1 General

The metal impact monitoring system is designed to enable early detection of any debris, detached internal structural items, and hardware present in the reactor coolant system.

A metal impact monitoring system for Indian Point Unit 2 was installed during the 1976 refueling outage and was operational when the plant returned to service in September 1976. At that time, component "signature acquisition" of the nuclear steam supply system components (baseline data) was obtained at selected plant operating conditions for future reference. The metal impact monitoring system was modified during the 1982 refueling outage.

4.6.2 Description

This system involves the use of a metal impact monitoring system capable of detecting changes in reactor coolant system vibrations and converting that input into an electronic signal thereby providing an indication to operating personnel that an undesireable level of foreign material may be present in the reactor coolant. While the installed system has no control capability, it is nevertheless quite valuable as an advisory system.

Metal impact monitoring is accomplished by the installation of specially developed transducers (accelerometers) mounted on the exterior of the reactor coolant system and steam generators. When the interior of the reactor coolant system is struck by bouncing debris, the structure is shock excited producing local wall accelerations that are detected by the transducers, amplified, conditioned, and fed to the metal impact monitoring system. The metal impact monitoring system further conditions the signals for recording and display in the control room.

The transducers are located on the following equipment:

1. Reactor vessel head.
2. Incore instrumentation penetration (below reactor vessel).
3. Steam generators.

APPENDIX 4A
DETERMINATION OF REACTOR PRESSURE VESSEL NIL-DUCTILITY TRANSITION TEMPERATURE (NDTT)

4A.1 MEASUREMENT OF INTEGRATED FAST NEUTRON (E > 1.0 MEV) FLUX AT THE IRRADIATION SAMPLES

The energy dependent neutron fluxes at the irradiation samples are obtained from the DOT(1), a two-dimensional discrete ordinates transport theory code. Dosimeters in the surveillance program include CdO shielded U-238, Np-237, Co-Al, Cu, Ni, Cd shielding Co-Al, and Fe from specimens, which will be contained in the capsule assemblies.

The specific activities of the dosimeters are to be determined by the multichannel analyzer and NAI scintillation detector. The equipment calibration shall be accomplished with ^{54}Mn and ^{60}Co radioactivity standards obtained from the U.S National Bureau of Standards or the equivalent. All activities will be corrected to the time-of-removal (TOR) at reactor shutdown.

Infinite dilute saturated activities (A_{SAT}) will be calculated for each of the dosimeters because A_{SAT} is directly related to the product of the energy dependent microscopic activation cross-section and the neutron flux density. The relationship between A_{TOR} and A_{SAT} is given by:

$$\frac{A_{TOR}}{A_{SAT}} = \sum_{m=1}^{m=n} P_m \left(1 - e^{-\lambda T} m\right) \left(e^{-\lambda t} m\right)$$

Where: λ = decay constant for the activation product, 1/day
t_m = decay time after operating period m, days
T_m = operating days P_m = average fraction of full power during operating period
P_m = average fraction of full power during operating period

The primary result desired from the dosimeter analysis in the total neutron fluence (E > 1 MeV) that the surveillance specimens and pressure vessel have received. The average flux density at full power is given by:

$$\phi = A_{SAT}/N_o\bar{\sigma}$$

Where: ϕ = energy dependent neutron flux density, n/cm^2-sec
$\bar{\sigma}$ = spectrum averaged activation cross-section, cm^2
N_o = number of target atoms per mg

The total neutron flux fluence is then equal to the product of the averaged neutron flux and the equivalent reactor operating time at full power.

4A.2 CALCULATION OF INTEGRATED FAST NEUTRON (E > 1.0 MEV) FLUX AT THE IRRADIATION SAMPLES

In the analysis of the neutron environment within a pressurized water reactor geometry, predictions of the spatial neutron flux magnitude, and energy spectra are made with the DOT (two-dimensional discrete ordinates transport theory code). First, the radial and azimuthal distributions are obtained from an R, θ computation normalized to the reactor core power

density representative of the axial midplane. A second calculation in R, Z geometry is used to provide relative axial variations of neutron flux in the pertinent regions of the pressure vessel. A three-dimensional description of the neutron environment is then constructed by assuming separability and using the relation:

$$\phi(R,\theta, Z,E) = \phi\,(R,\theta, E) \times\; F(Z,E)$$

Where ϕ (R, θ, E) represents the absolute neutron flux magnitude at the core midplane as determined from the R, θ computation and F(Z, E) is the relative axial distribution obtained from the R, Z analysis and normalized to unity at the core midplane.

From a neutronic standpoint, the inclusion of the surveillance capsule structures in the R, θ analytical model is significant. Neutron dosimetry from these capsules provides a means for evaluating the analytical model by direct comparison with measurement. Since the presence of the capsules has a marked impact on both the neutron flux magnitude and energy spectrum, a meaningful comparison of measurement and calculation can be made only if these perturbation effects are properly accounted for in the analysis.

Two distinct sets of transport calculations are carried out. The first, a single computation in the conventional forward mode, is used primarily to obtain relative neutron energy distributions throughout the reactor geometry as well as to establish relative radial distributions of exposure parameters (ϕ (E > 1.0 MeV), ϕ (E > 0.1 MeV), and dpa) through the vessel wall. The neutron spectral information is required for the interpretation of neutron dosimetry withdrawn from surveillance capsules as well as for the determination of exposure parameter ratios: i.e., dpa/ϕ(E > 1.0 MeV), within the pressure vessel geometry. The relative radial gradient information is required to permit the projection of measured exposure parameters to locations interior to the pressure vessel wall; i.e., the 1/4T, 1/2T, and 3/4T locations.

The second set of calculations consists of a series of adjoint analyses relating the fast neutron flux (E > 1.0 MeV) at surveillance capsule positions, and several azimuthal locations on the pressure vessel inner radius to neutron source distributions within the reactor core. The importance functions generated from these adjoint analyses provide the basis for all absolute exposure projections and comparison with measurement. These importance functions, when combined with cycle specific neutron source distributions, yield absolute predictions of neutron exposure at the locations of interest for each of the operating fuel cycles; and establish the means to perform similar predictions and dosimetry evaluations for all subsequent fuel cycles. It is important to note that the cycle specific neutron source distributions utilized in these analyses include not only spatial variations of fission rates within the reactor core; but, also account for the effects of varying neutron yield per fission and fission spectrum introduced by the build-in of plutonium as the burnup of individual fuel assemblies increased.

The absolute cycle specific data from the adjoint evaluations together with relative neutron energy spectra and radial distribution information from the forward calculation provide the means to:

1. Evaluate neutron dosimetry obtained from the surveillance capsule program.

2. Extrapolate dosimetry results to key locations at the inner radius and through the thickness of the pressure vessel wall.

3. Enable a direct comparison of analytical prediction with measurement.

4. Establish a mechanism for projection of pressure vessel exposure as the design of each new fuel cycle evolves.

The forward transport calculation is carried out in R, θ geometry using the DOT two-dimensional discrete ordinates code[1] and the SAILOR cross-section library[2]. The SAILOR library is a 47 group ENDFB-IV based data set produced specifically for light water reactor applications. In these analyses anisotropic scattering was treated with a P_3 expansion of the cross-sections and the angular discretization was modeled with an S_8 order of angular quadrature. The reference forward calculation is normalized to a core power.

All adjoint analyses are also carried out using an S_8 order of angular quadrature and the P_3 cross-section approximation from the SAILOR library. Adjoint source locations are chosen at several azimuthal locations at the pressure vessel inner radius and at the geometric center of surveillance capsules positioned at 4^o and 40^o relative to the core cardinal axes. Again these calculations are run in R, θ geometry to provide neutron source distribution importance functions for the exposure parameter of interest; in this case, ϕ (E > 1.0 MeV). Having the importance functions and appropriate core source distributions, the response of interest could be calculated as:

$$R(r,\theta) = \int_r \int_\theta \int_E I(r,\theta,E) S(r,\theta,E) r \, dr \, d\theta \, dE$$

where:	R(r, θ)	=	ϕ (E > 1.0 MeV) at radius r and azimuthal angle θ
	I (r, θ, E)	=	Adjoint importance function at radius r, azimuthal angle θ, and neutron source energy E.
	S (r, θ, E)	=	Neutron source strength at core location r, θ. and energy E.

Forward transport as well as the adjoint analyses for Indian Point Unit 2 were carried out and summarized in Reference 3.

In the R, θ analysis, the discretization of the angular flux is represented by a symmetric S_8 quadrature. However, in the R, Z case the use of this relatively low order quadrature set can often prove to be inadequate. At large depths within the pressure vessel, the axial distribution of neutron flux is dominated by neutron streaming in the annulus between the pressure vessel wall and the primary biological shield. To account for this effect a high resolution angular quadrature is required. Therefore, in this analysis a 124 angle asymmetric quadrature is employed. For regions of the reactor, which are above the core midplane, this quadrature is constructed with 109 angles biased in the upward directions, i.e., the direction of prime interest, and 15 angles biased downward. For analysis below the core midplane, the quadrature is reversed with 109 angles biased in the downward direction. Complete descriptions of both the symmetric S_8 and the asymmetric 124 angle quadratures are given in Reference 1.

The calculated fast neutron flux distributions may be used in conjunction with damage trend curves to predict the degree of embrittlement of the reactor vessel steel over its service life. The accuracy of these neutron flux profiles depends on the analyst's ability to define an appropriate core power distribution, the adequacy of the cross-sections used in the transport analysis, and the applicability of the geometric modeling of the reactor. Taken as a whole, these factors combine to yield an overall uncertainty of 20-percent in the prediction of neutron flux and fluence within the pressure vessel wall.

4A.3 MEASUREMENT OF THE INITIAL NIL-DUCTILITY TRANSITION TEMPERATURE OF THE REACTOR PRESSURE VESSEL BASE PLATE AND FORGINGS MATERIAL

The unirradiated or initial NDTT of pressure vessel reactor materials was measured by two methods. These methods were the drop weight test per ASTM E208 and the Charpy V-notch impact test (Type A) per ASTM E23.

The NDTT is defined in ASTM E208 as the temperature at which a drop weight test specimen is broken in a series of tests in which duplicate no-break performance occurs at a temperature of 10^{o}F higher.

The NDTT temperature, as determined by drop weight tests is the RT_{NDT} if, at 60^{o}F above the NDTT, at least 50-ft-lbs of energy and 35 mils lateral expansion are obtained in Charpy V tests on specimens oriented in the weak direction (traverse to the direction of maximum working).

The NDTT has been correlated with Charpy V-notch impact tests results.

For SA 302B and A508 Class 2 steels the Charpy V-notch "fix" temperature, which corresponds to NDTT is the temperature at 30-ft-lbs in accordance with Section III Table N-421 of the ASME Code for Nuclear Vessels. The curve of the temperature versus energy observed in breaking the specimen was plotted.

To obtain this curve 15 tests were performed, which include three tests at five different temperatures. The intersection of the energy versus temperature curve with the 30-ft-lbs ordinate is designated as NDTT.

As part of the Westinghouse surveillance program referred to above, Charpy V impact tests, tensile tests, and fracture mechanics specimens are taken from the plate of forging material. To assess any possible uncertainties in the consideration of NDTT shift for welds, heat affected zone and base metal, test specimens of these three "material types" have been also included in the reactor vessel surveillance program.

Encapsulated specimens are located on the outside diameter surface of the thermal shield where the fast neutron flux density is about three times that at the adjacent vessel wall surface. The capsules also contain several dosimeter materials for experimentally determining the average neutron flux density at each capsule location during the exposure period.

REFERENCES FOR APPENDIX 4A

1. R. G. Soltesz, et al, "Nuclear Rocket Shielding Methods, Modification, Updating and Input Data Preparation - Volume 5, Two-Dimensional Discrete Ordinates Transport Techinque," WANL-PR-(LL)-304, August, 1970.

2. "ORNL RSIC Data Library Collection DLC-76, SAILOR Coupled Self Shielded, 47 Neutron, 20 Gamma-Ray, P_3, Cross Section Library for Light Water Reactors".

3. S.L. Anderson "Plant Specific Fast Neutron Exposure Evaluation of the Indian Point Unit 2 Reactor Pressure Vessel and Surveillance Capsules Fuel Cycles 1 through 9". Westinghouse report PSE-REA-88/127, July 1988.

APPENDIX 4B
SUPPORT STRUCTURES FOR REACTOR COOLANT SYSTEM COMPONENTS

The reactor coolant system components and their supports are designed as seismic Class I components as discussed in Section 1.11. In 2003, the reactor coolant loop and its component supports were re-analyzed due to a power uprate. This latest analysis does not consider the coincident combination of blowdown and seismic loads.

4B.1 REACTOR VESSEL

The reactor vessel support structure consists of a circular box section ring girder fabricated of carbon steel plates. The bottom flange of the girder is in continuous contact (except for openings for neutron detectors) with a non-yielding concrete foundation.

The reactor vessel has four supports located at alternate nozzles and cooled by the component cooling system. Each support bears on a support shoe, which is fastened to the support structure. The support shoe is a structural member that transmits the support loads to the supporting structure. The support shoe is designed to restrain vertical, lateral, and rotational movement of the reactor vessel, but allows for thermal growth by permitting radial sliding at each support on bearing plates.

4B.2 STEAM GENERATORS

The steam generators are supported within a caged structural system consisting of four connected columns welded together, fabricated of carbon steel members, with provisions for limited movement of the structure in a horizontal direction with a system of "Lubrite" plates, hydraulic snubbers, guides, and stops to accommodate piping expansion. The "Lubrite" plates, hydraulic snubbers, guides, and stops were originally designed as a rigid support to resist the action of seismic and pipe break loads.

In 2000, the number of hydraulic snubbers supporting the steam generator frame in the direction of the hot leg, has been reduced from the original six down to two per steam generator. The two remaining snubbers are located at the upper support point of the frame at Elevation 92'-0". The analysis of the reactor coolant loop and of the steam generator support structure accounts for the replacement steam generator and for the reduced number of hydraulic snubbers.

4B.3 REACTOR COOLANT PUMP

Each reactor coolant pump is supported on a three-legged structural system consisting of three connected columns fabricated of carbon steel members, structural sections, and pipe. Provisions for limited movement of the structure in any horizontal direction to accommodate piping expansion is accomplished with a sliding "Lubrite" base plate arrangement, and a system of tie rods and anchor bolts, which restrain the structure from movement beyond the calculated limits.

4B.4 PRESSURIZER

The pressurizer is supported on a free-standing structural system consisting of six connected columns fabricated of carbon steel members, all welded together and secured at the base by anchor bolts.

4B.5 PIPING

The reactor coolant piping layout is designed on the basis of providing "floating" supports for the steam generator and reactor coolant pump in order to absorb the thermal expansion from the fixed or anchored reactor vessel. A comprehensive thermal analysis has been performed to ensure that stresses induced by linear thermal expansion are within code limits.

4B.6 APPLICABILITY OF UNIT 3 PIPE BREAK ANALYSES TO UNIT 2

A report (Reference 1) entitled, "Analysis of Reactor Coolant System for Postulated Loss-of-Coolant Accident: Indian Point Unit 3 Nuclear Power Plant," has been submitted to the NRC. This report postulates pipe breaks at the locations in the primary loop, which induce the most severe asymmetric loads on the reactor vessel. The analyses performed included the effects of the addition of pipe motion limiters and demonstrate the adequacy of the entire system.

Reference 4 of Section 4.2 addresses the applicability of this report to Unit 2. Because of the similarity of the plants the nature of the system response, and the installation in Unit 2 of the modifications discussed in that report, the conclusions stated for Unit 3 in that report are found to be applicable to Unit 2.

4B.7 LEAK BEFORE BREAK

In 1989, the NRC approved elimination of the necessity for considering and protecting against dynamic effects of postulated primary loop pipe ruptures from the design basis of Indian Point Unit 2 as discussed in Section 4.1.2.4. "Leak before break" technology was applied as permitted by revised General Design Criterion 4 of 10CFR50, Appendix A. References 2, 3, 4 and 5 contain further information.

REFERENCES FOR APPENDIX 4B

1. "Analysis of Reactor Coolant System for Postulated Loss-Of-Coolant Accident: Indian Point Unit 3 Nuclear Power Plant." WCAP-9117 (Proprietary) and WCAP-9130 (Non-Proprietary), Westinghouse Electric Corporation.

2. May 23, 1988 letter, Bram to Document Control Desk, subject: Leak-Before-Break (LBB).

3. November 18, 1988 letter, Bram to Document Control Desk, subject: Leak-Before-Break (LBB) Submittal (TAC 68318).

4. January 12, 1989 letter, Bram to Document Control Desk, subject: Leak-Before-Break (LBB) Submittal (TAC 68318).

5. Letter from Donald Brinkman, NRC, to Stephen B. Bram, Con Edison, Subject: Safety Evaluation Report on Elimination of Dynamic Effect of Postulated Primary Loop Pipe Ruptures from Design Basis for Indian Point Unit 2 (TAC No. 68318), dated February 23, 1989.

APPENDIX 4C
SENSITIZED STAINLESS STEEL

4C.1 INTRODUCTION

Westinghouse has evaluated the use of sensitized stainless steel for reactor components in pressurized water reactors. The results of this evaluation are summarized in WCAP 7477-L (Reference 1), which cover the nature of sensitization conditions leading to stress corrosion and associated problems with both sensitized and non-sensitized stainless steel. The results of extensive testing and service experience that justify the use of stainless steel in the sensitized condition for components in Westinghouse systems is presented in the report.

Sensitized stainless steel is subject to stress corrosion and must not be exposed to certain environments that will cause cracking. Chlorides and fluorides are the most important contaminants, although oxygen, low pH, elevated temperature, and high stress generally must also be present to cause cracking. When subjected to environments that cause cracking, the cracks are usually intergranular in sensitized stainless steel.

The stainless steel safe-ends on the reactor vessel, pressurizer, and steam generator nozzles may become somewhat sensitized during stress relief of the vessel. The post weld heat treatment (PWHT) temperatures and minimum time are consistent with ASME Section III requirements. The degree of sensitization of the safe-ends varies from plant to plant, depending on the materials used and the detailed processing performed by the various vendors. For Indian Point Unit 2, the specific design and construction practices are discussed in the following sections. The outer diameter and inner diameter safe-ends of the reactor vessel were overlaid with type 308L and Inconel weld metal to eliminate any question of intergranular attack in areas where there is limited accessibility for inservice inspection and plant maintenance. There is complete accessibility to the remaining reactor coolant system components. The pre-operational inspection of the reactor coolant system components provides assurance that there is no stress corrosion cracking of sensitized stainless steel.

4C.2 REACTOR COOLANT SYSTEM NOZZLE SAFE-ENDS

4C.2.1 Reactor Vessel Primary Nozzle Safe-Ends

1. <u>Method of Fabrication</u> (See Figure 4C-1)

   a. Wrought stainless steel - Type 316 Forging welded to SA-336 nozzle with Inconel weld metal. Attached prior to final post weld heat treatment.

   b. Forging was overlaid on ID and OD with type 308L stainless and Inconel weld metal. This was performed in the field after the primary coolant piping was attached to the nozzles.

2. Inspection

a. Forging safe-ends were examined by ultrasonic testing and penetrant testing at Combustion Engineering using Section III acceptance standards.

b. Weld overlay of the ID and OD surfaces was examined by ultrasonic testing and penetrant testing. The acceptance standards are shown below:

(1) Ultrasonic Acceptance Standards

Each discontinuity that produced a response equal to or exceeding the calibration reference line and was 0.5-in. or greater in length was considered rejectable and removed.

Discontinuities that produced a response equal to or greater than the calibration reference line and exceed 0.25-in., but were less than 0.5-in. in length were considered acceptable if separated by a minimum distance of 2-in. from similar discontinuities.

Each discontinuity that produced a response between 50 and 100-percent of the calibration reference line and exceeded one inch but was not more than 1.5-in. in length, were acceptable if separated by a minimum distance of 2-in. from similar indications.

(2) Penetrant Inspection Acceptance Standards

(a) Examination of welds by liquid penetrant methods were made over an area including the welds and base metal extending for at least 0.5-in. on each side of weld.

(b) Surfaces examined by fluid penetrant methods were free of laps, fissures, cracks, other linear indications.

(c) Weld area and adjacent wrought type base metal(s) - In any 6-in. length of weld and adjacent base metal examined, there were no indications greater than 0.62-in. in maximum dimension, nor were there more than six indications with sum of maximum dimensions specified herein. Any 6-in. length of weld was interpreted to denote the 6-in. length selected in the least favorable location with respect to the discontinuities disclosed by the inspection test. All surfaces examined were free of linearly disposed indications of four or more indications in a line and each separated by 1/16-in. or less, edge to edge.

(d) Weld area and adjacent cast type base metal(s) - In any 6-in. length of weld examined, there were no indications greater than those defined in 3, above. The adjacent cast base metal was free of random indications in excess of those shown in the following table for a distance of not less than 0.5-in. from toe(s) of weld:

Size of Indications, In.	Number per Square In.
> 1/8	None
>1/16 < 2/8	2
< 1/16	10

(e) All surfaces examined were free of linearly disposed indications of four or more indications in a line and each separated by 1/16-in. or less, edge to edge. Rounded indications were those which were circular or elliptical with the length less than twice the width.

4C.2.2 STEAM GENERATOR PRIMARY NOZZLE SAFE-ENDS (See Figure 4C-2)

1. Method of Fabrication

   Weld metal buttering applied to carbon steel (A-216 Casting) nozzles prior to final post weld heat treatment. Stainless weld metal for the first layer was type 309L, and for the balance was type 308L.

2. Inspection

   Buttered safe-ends were examined by penetrant testing and radiography testing using ASME Boiler and Pressure Vessel Code Section III acceptance standards.

4C.2.3 PRESSURIZER (See Figure 4C-3)

1. Method of Fabrication

   Wrought stainless steel pipe or forgings welded to carbon steel (A-216 Casting) nozzles with type 309 weld metal before post weld heat treatment. The surge nozzle safe-end was fabricated from SA-312 pipe, type 316, and the spray, relief, and safety nozzle safe-ends from SA-182 forgings, type 316.

2. Inspection

   Wrought material was examined by ultrasonic testing and penetrant testing using Section III acceptance standards.

4C.3 REACTOR COOLANT SYSTEM CONSTRUCTION

All primary piping and fittings were given a solution annealing treatment consisting of heating to 1900 - 1950^{o}F, holding 1 hr/in. of thickness, and water quenching. This ensured that the material would not be sensitized.

Main coolant pipe welds are of type 308 or type 316 stainless steels. Welding was performed by the manual metal arc process after the root pass was completed using an insert followed by three layers using the manual gas shielded tungsten arc process. The maximum energy input possible with the manual metal arc process is on the order of 20,000 joules per linear inch of weld. With the large heat sink available in this thick-walled pipe (2.375 to 3.00-in.) and the interpass temperature control of 350^{o}F maximum, there will be no sensitization of the solution-treated pipe during welding.

Venting provisions have been made at high points throughout the reactor coolant system to relieve entrapped air when the system is filled and pressurized. Principally, vents are installed on the reactor coolant pumps with additional vents available on the control rod drive mechanisms, on instruments, and on a number of connecting pipes. For normal venting of the reactor coolant system, only the principal venting points are used. The amount of oxygen, which could be trapped in the remaining small volumes becomes negligible as the system is pressurized and the oxygen is scavenged by the hydrazine specifically added for this purpose prior to operation. During operation, the oxygen levels are kept low consistent with water chemistry requirements as described in the Technical Specifications. In addition to the high point vents, a connection is installed downstream of the Power-operated Relief Valves to permit pulling the air out of the system under vacuum during system refilling.

4C.4 REACTOR COOLANT SYSTEM OPERATIONAL STRESSES

To avoid unusual stresses in areas where nozzle safe-ends are joined to the piping, precautions were taken to eliminate unnecessary stresses due to erection of the various components of the reactor coolant system. The primary coolant system piping closure pieces were two pipe fitting subassemblies located between the steam generator and the primary coolant pump. The 40-degree elbow of the loop piping was first installed on the steam generator outlet nozzles. Then the gap to be closed by the closure pieces was physically measured between the 40-degree elbow outlet and the inlet nozzle of the pump. These measured dimensions for each individual loop were compensated and adjusted for the expected field weld shrinkage. The resulting net true dimensions were then transmitted to the pipe shop fabricator who prepared the final closure pipe subassemblies for each primary coolant loop. Upon welding these specially dimensioned pipe subassemblies in place, the primary coolant system closure was accomplished for each loop in a condition, which is free from cold spring.

As a precaution that the behavior of the reactor coolant system during operating conditions would be as predicted, measurements were made at incremental temperature increases during the hot functional test. The measurements were made to check the movement of the components at temperature and pressure to ensure interferences were not present. Data taken during the test were compared with the flexibility analysis predictions and evaluated.

4C.5 INSERVICE INSPECTION CAPABILITY

As a final check on the adequacy of the precautions taken to avoid any reactor coolant system failure as a result of severely sensitized stainless steel, a postoperational inspection plan was developed for the nozzle safe-ends within the reactor coolant system boundary. The pressurizer and steam generator stainless steel safe-ends that were subjected to the furnace atmosphere during final stress relief are accessible for visual, surface, and volumetric inspection upon removal of the insulation at each safe-end. The reactor vessel safe-ends, which were subjected to the furnace atmosphere, are accessible for limited inspection by removal of the special access plugs provided in the primary concrete just above each nozzle. Upon removal of these plugs and the insulation on the safe-end, approximately 120-degrees of the top segment of the safe-ends are accessible for direct visual, surface, and remote volumetric inspection.

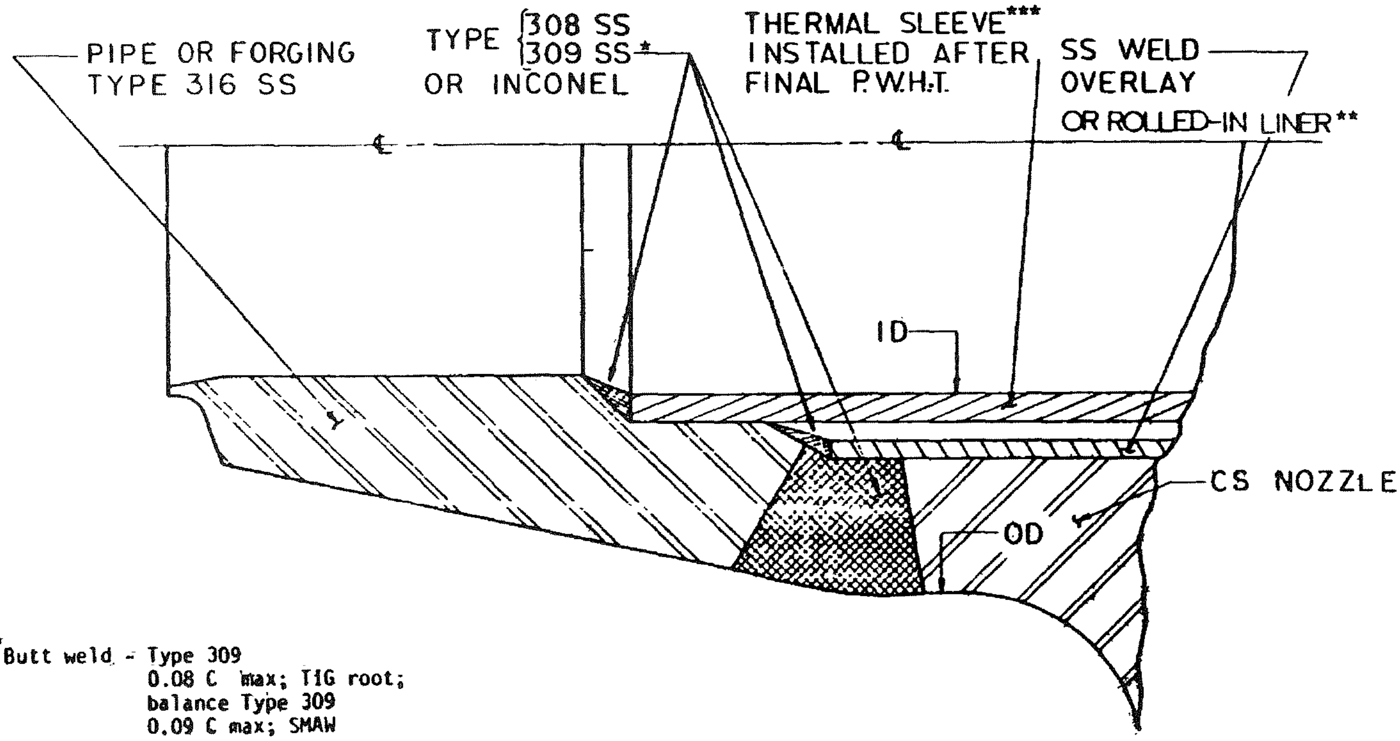
As specially designed devices for remote ultrasonic inspection and applicable procedures become available, and when metallurgical considerations indicate that this type of inspection is appropriate and necessary, such inspections will be accomplished utilizing the internal access

to the reactor vessel safe-ends. Requirements for inspection of the reactor coolant system are detailed in the facility Technical Specifications.

REFERENCES FOR APPENDIX 4C

1. WCAP-7477L (Proprietary), Westinghouse Electric Corporation.

APPENDIX 4C FIGURES



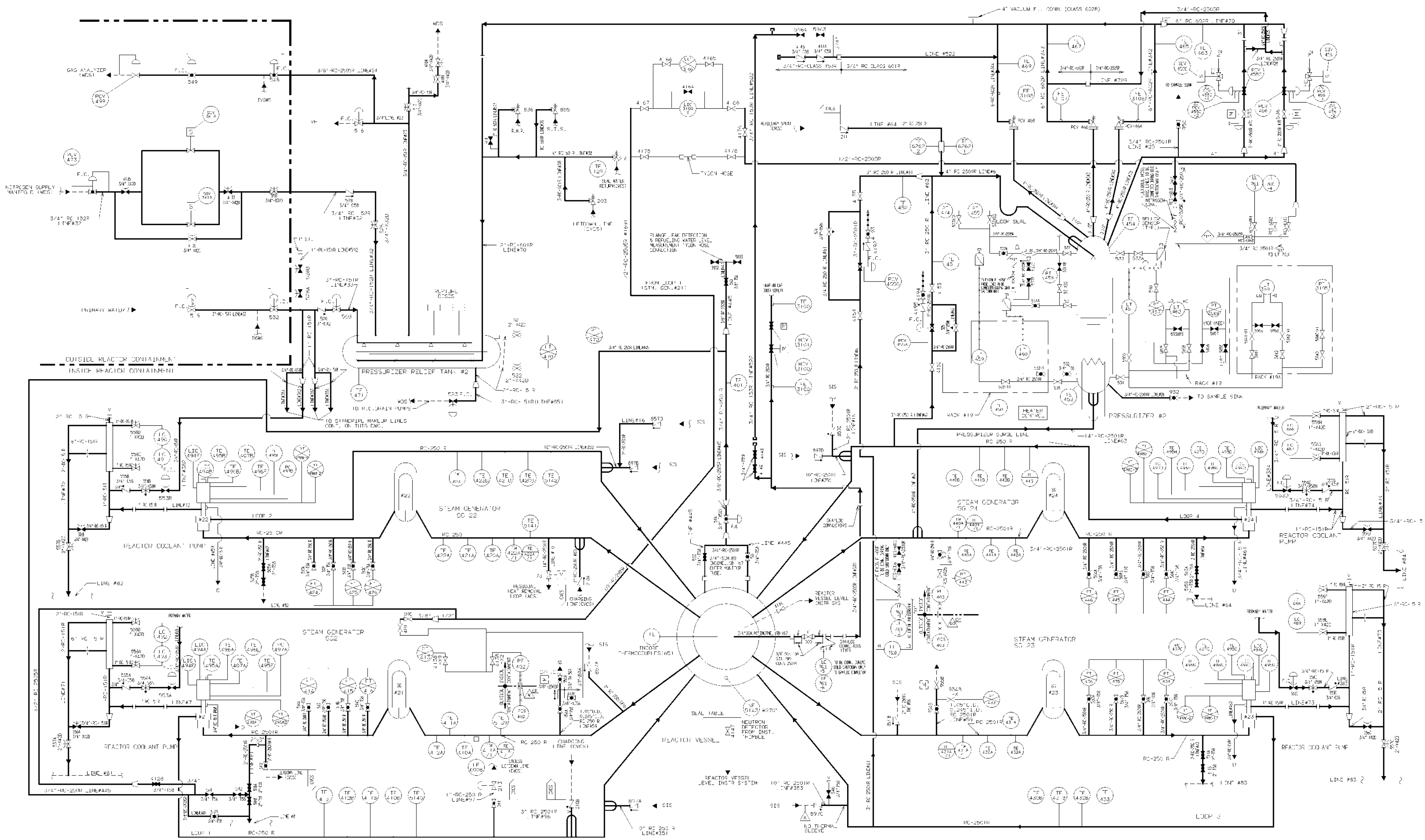
*Butt weld - Type 309
0.08 C max; TIG root;
balance Type 309
0.09 C max; SMAW

Attachment weld of thermal sleeve and rolled-in liner - Type 308 L 0.04 C max; TIG (made after final PWHT)

**Rolled-in liner welded top and bottom for spray, safety, and relief nozzles - Type 309 followed by Type 308 L weld overlay for surge nozzle

***Thermal sleeve welded for 45° of 360°

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 4C-3 SPRAY OR SURGE NOZZLE TAMPA PRESSURIZER	
MIC. No. 1999MC3745	REV. No. 17A



LEGEND

IVSWS - ISOLATION VALVE SEAL WATER SYS.
CVCS - CHEMICAL & VOLUME CONTROL SYS.
ACS AUXILIARY COOLANT SYS.
SIS SAFETY INJECTION SYS.
SS - SAMPLING SYS.
WDS WASTE DISPOSAL SYS.

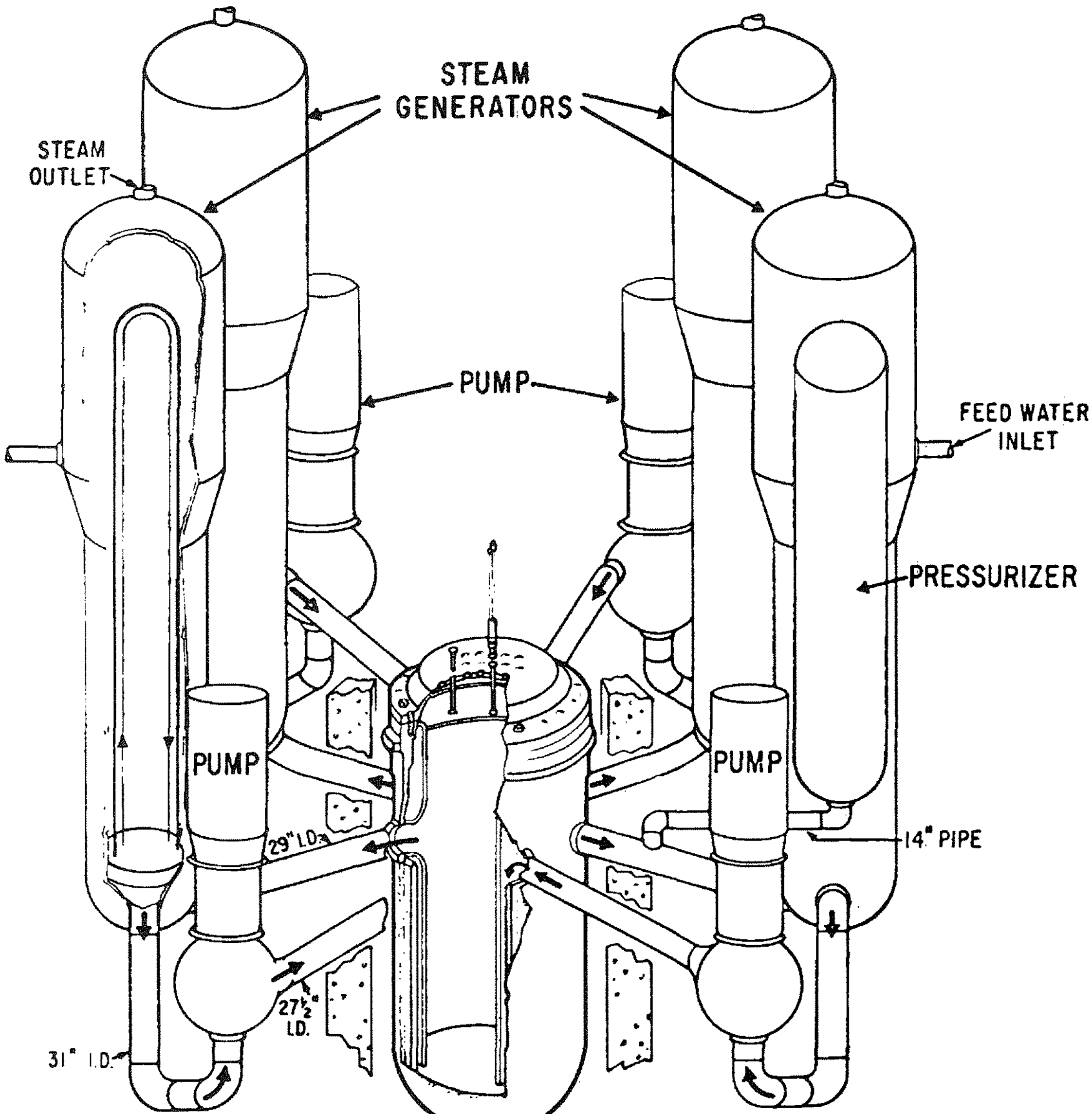
INDIAN POINT UNIT No. 2

UFSAR FIGURE 4.2-1

REACTOR COOLANT SYSTEM
FLOW DIAGRAM

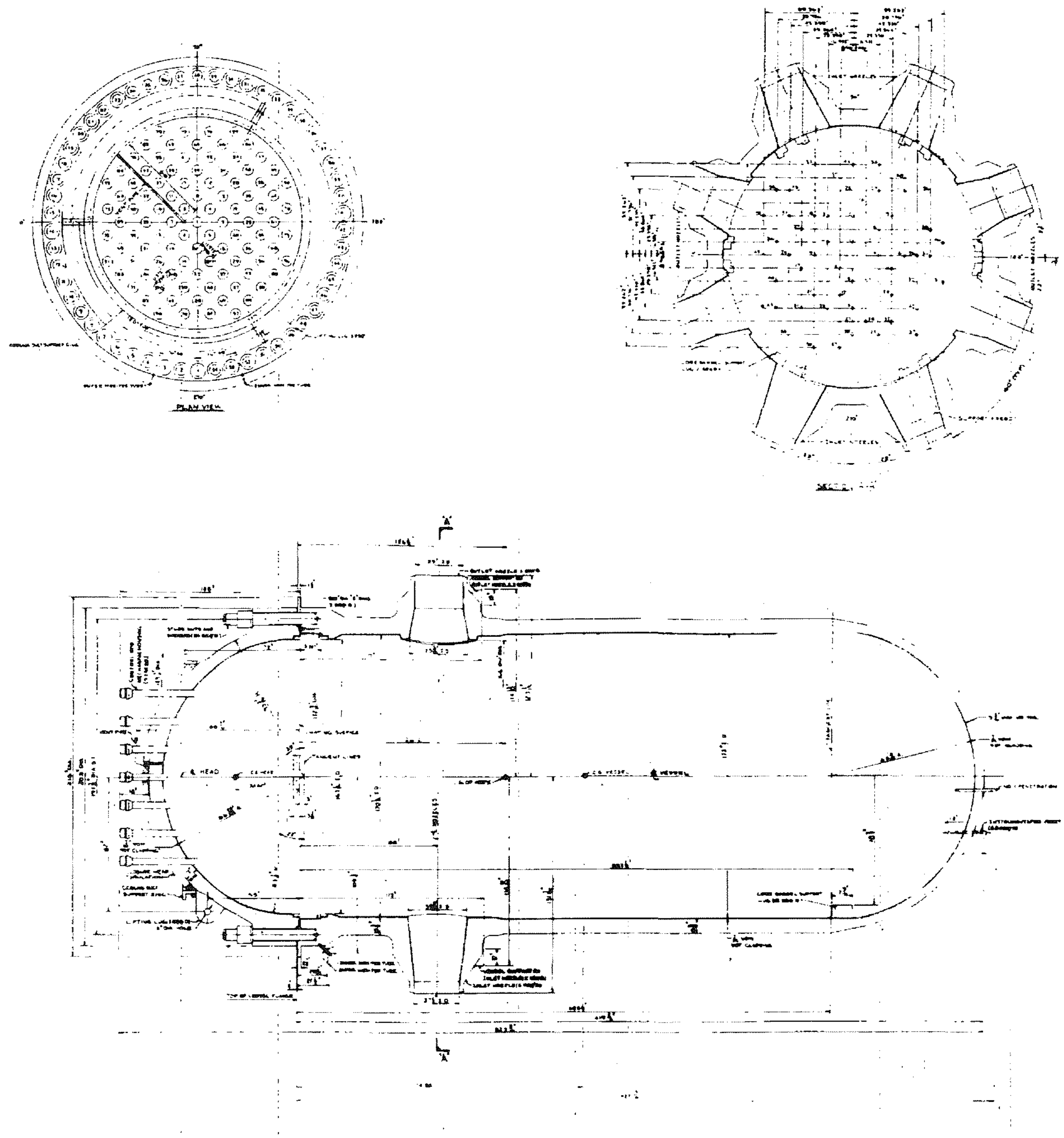
MIC. No. 1999MC3430 REV. No. 17B

VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE



VIEW LOOKING WEST

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 4.2-2	
REACTOR COOLANT SYSTEM SCHEMATIC FLOW DIAGRAM	
MIC. No. 1999MC3733	REV. No. 17A

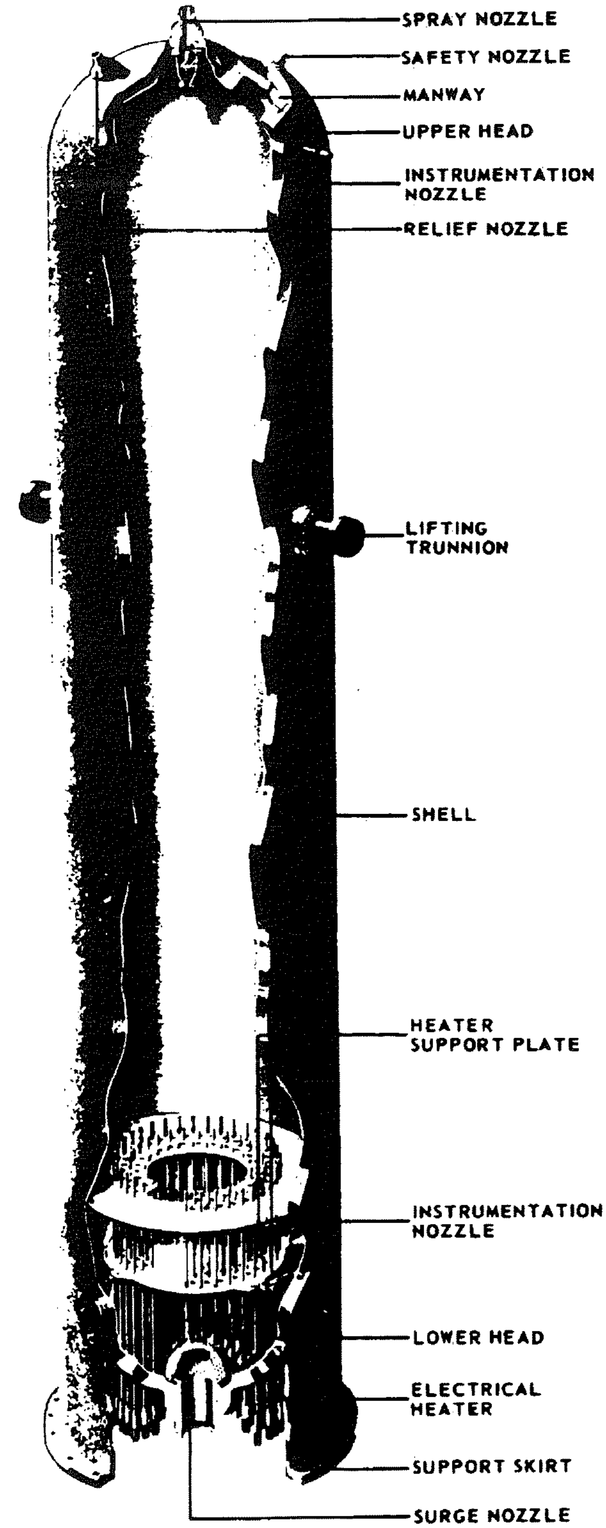


INDIAN POINT UNIT No. 2

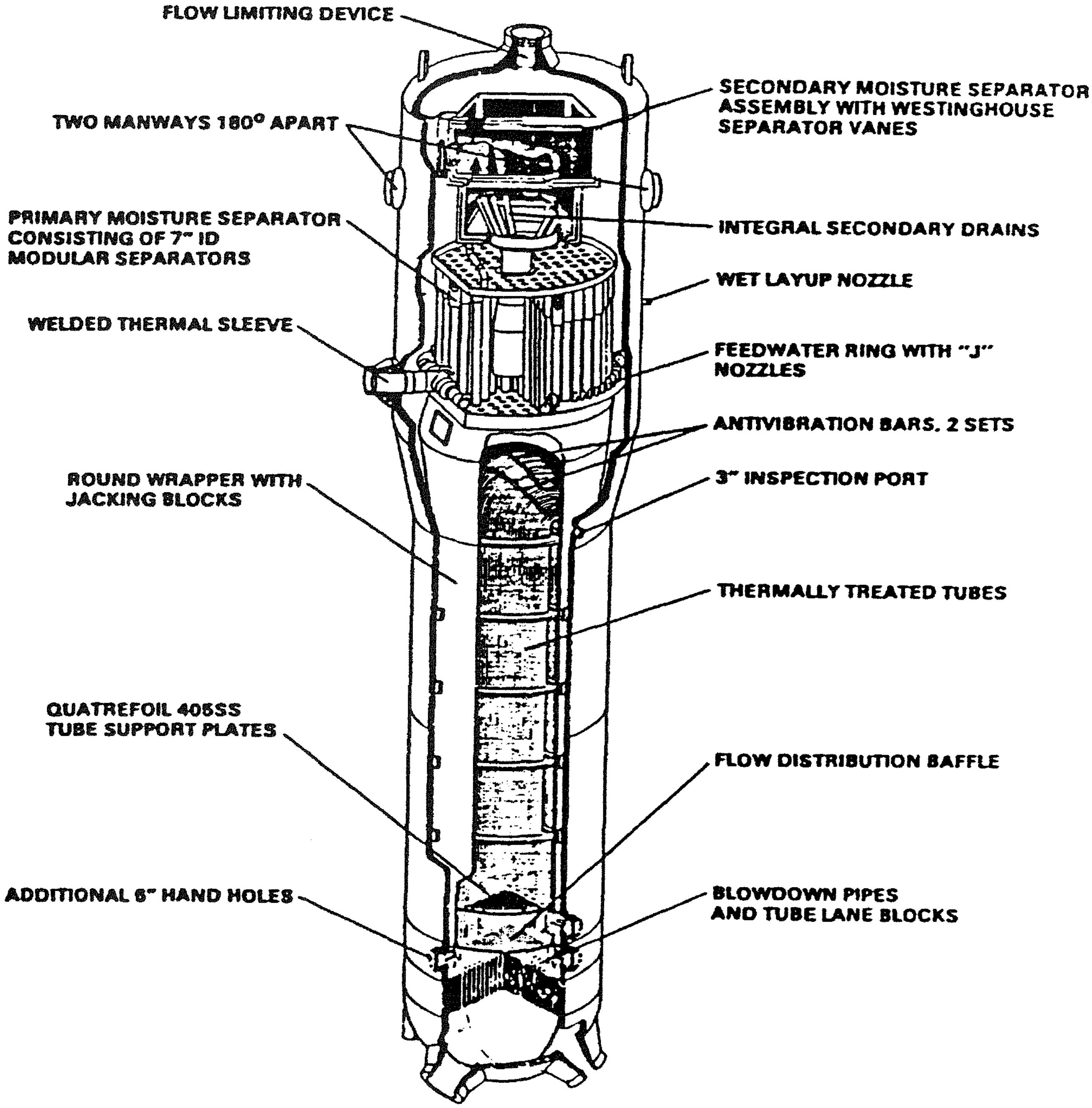
UFSAR FIGURE 4.2–3

REACTOR VESSEL

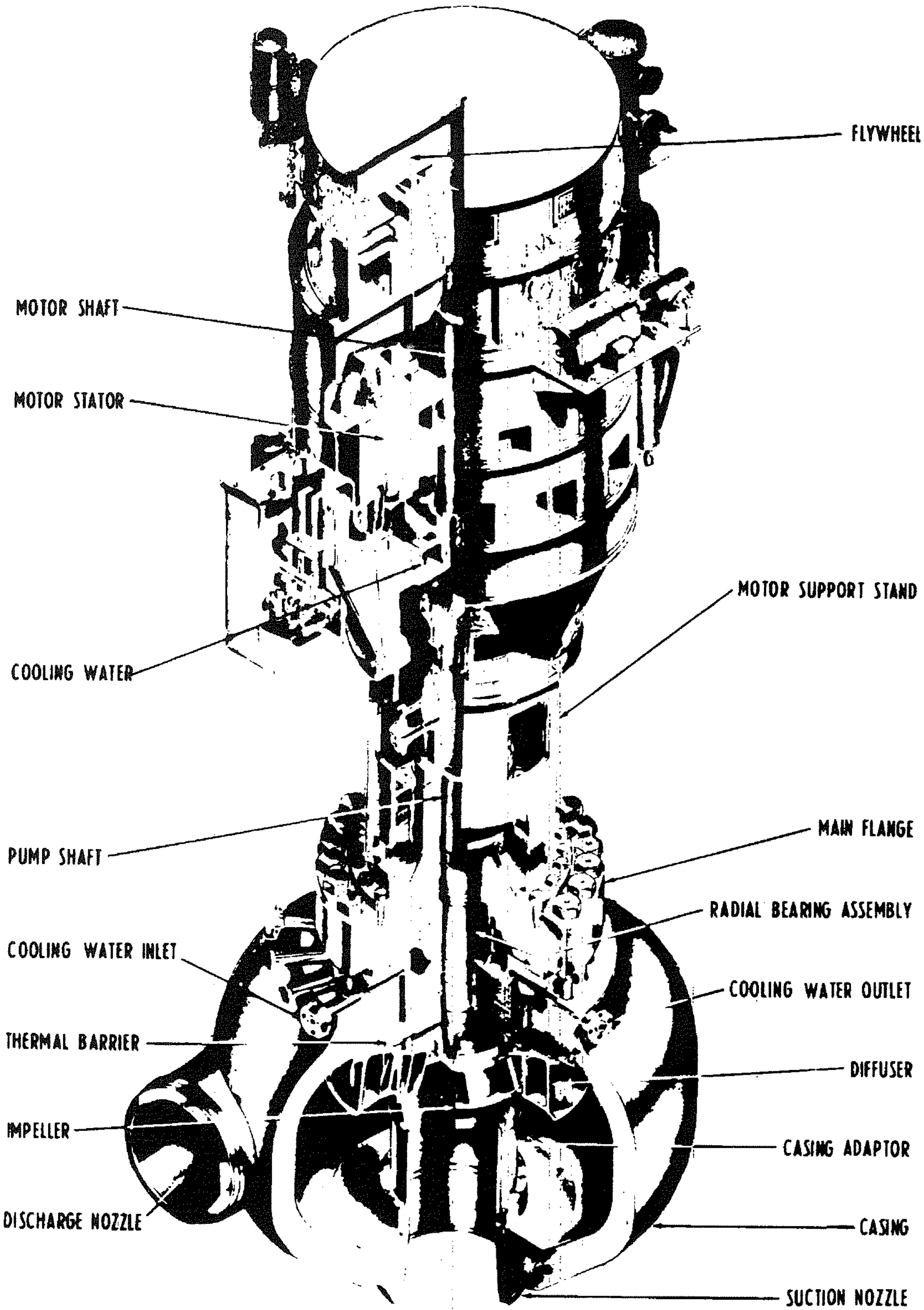
MIC. No. 1999MC3734	REV. No. 17A



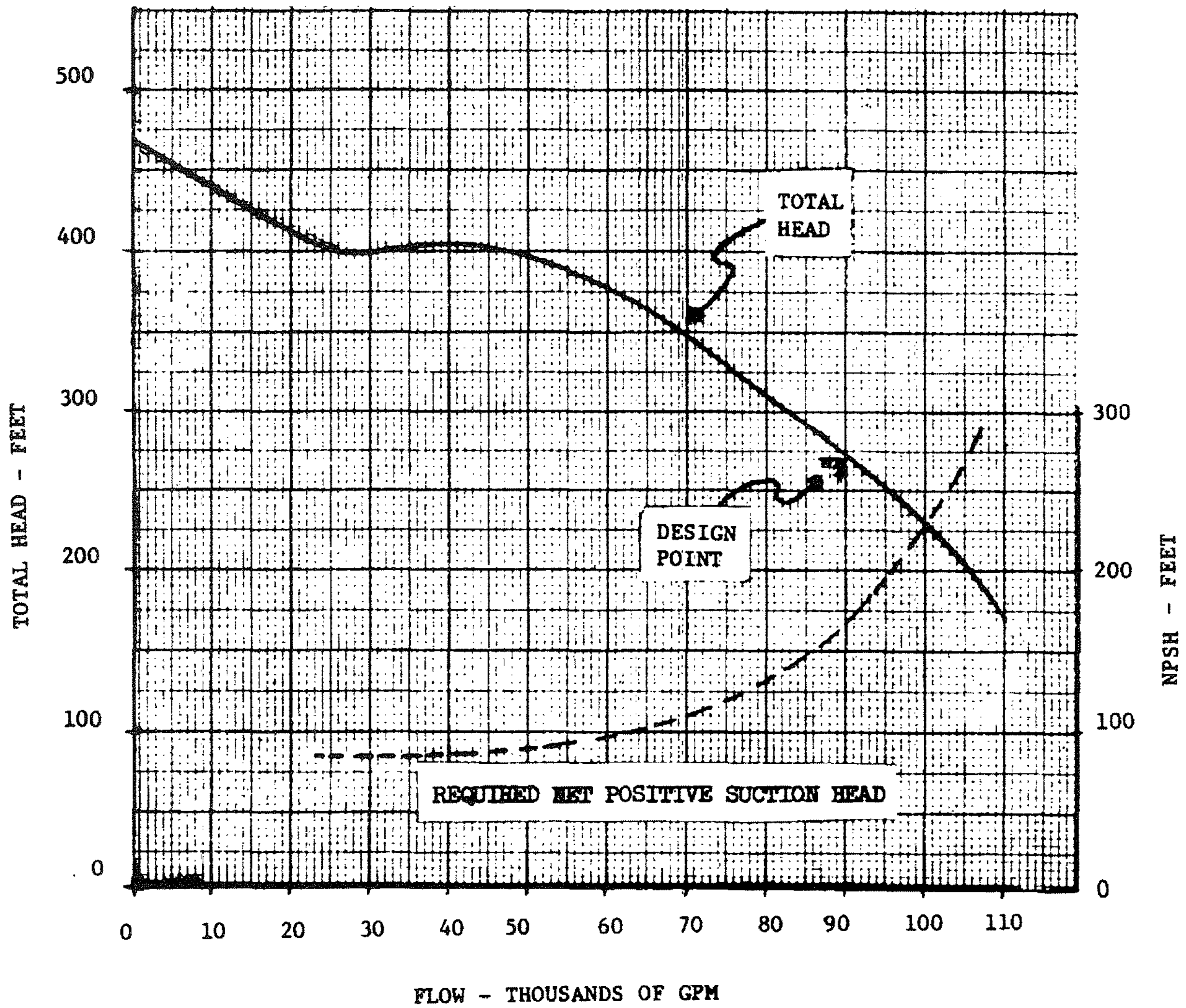
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 4.2–4 PRESSURIZER	
MIC. No. 1999MC3735	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 4.2-5 STEAM GENERATOR ASSEMBLY	
MIC. No. 1999MC3736	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 4.2-6 REACTOR COOLANT PUMP	
MIC. No. 1999MC3737	REV. No. 17A



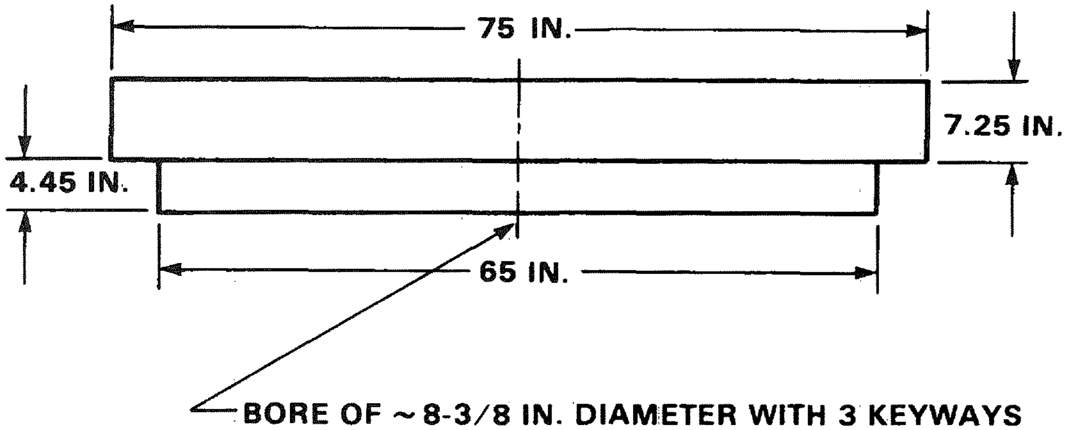
INDIAN POINT UNIT No. 2

UFSAR FIGURE 4.2-7

REACTOR COOLANT PUMP ESTIMATED PERFORMANCE CHARACTERISTICS

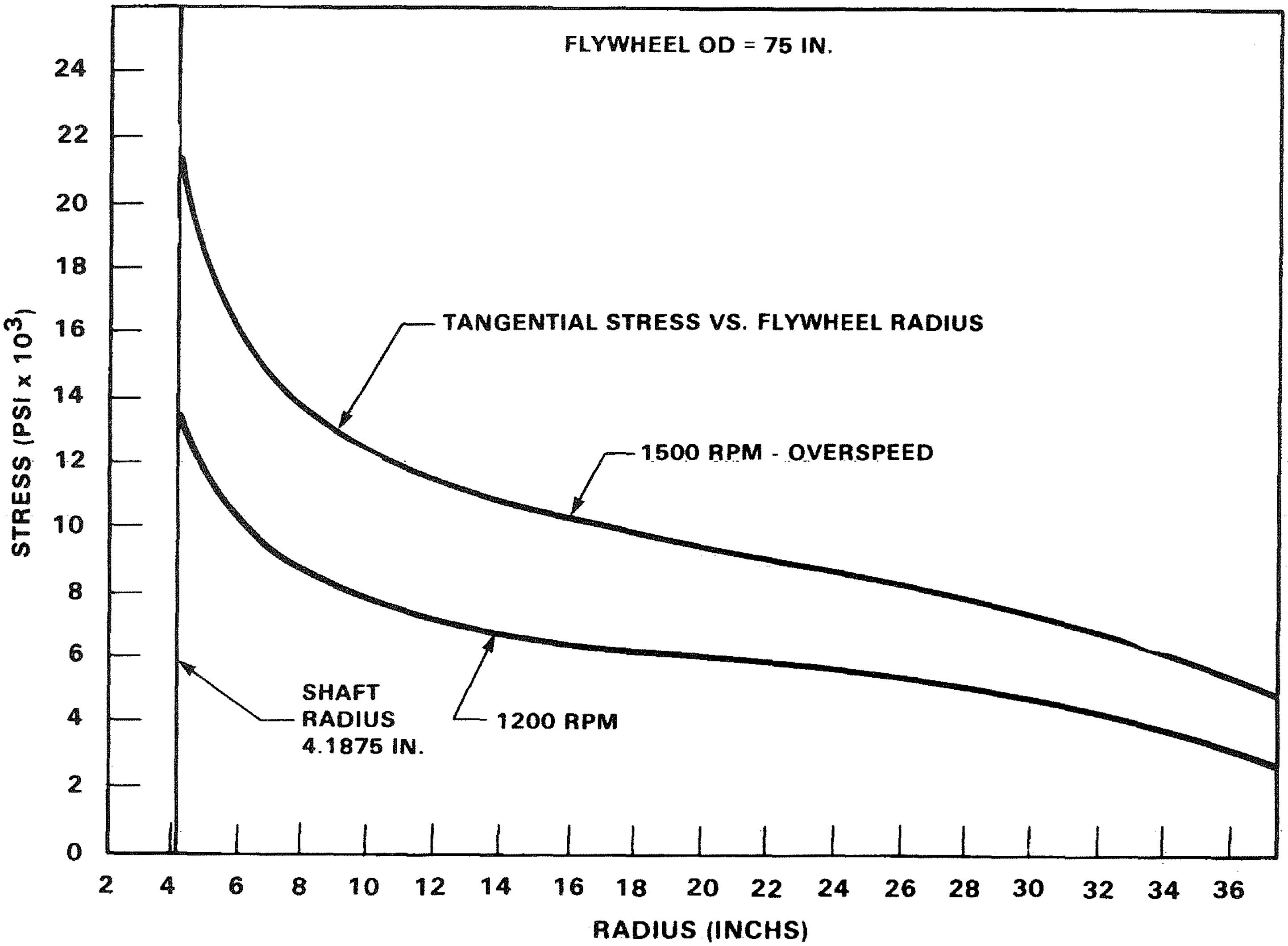
MIC. No. 1999MC3738	REV. No. 17A

FLYWHEEL



NOTE: THE PLATES ARE BOLTED TOGETHER WITH THE BOLTS ALIGNED PERPENDICULAR TO THE PLANES OF THE PLATES.

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 4.2-8 FLYWHEEL	
MIC. No. 1999MC3739	REV. No. 17A

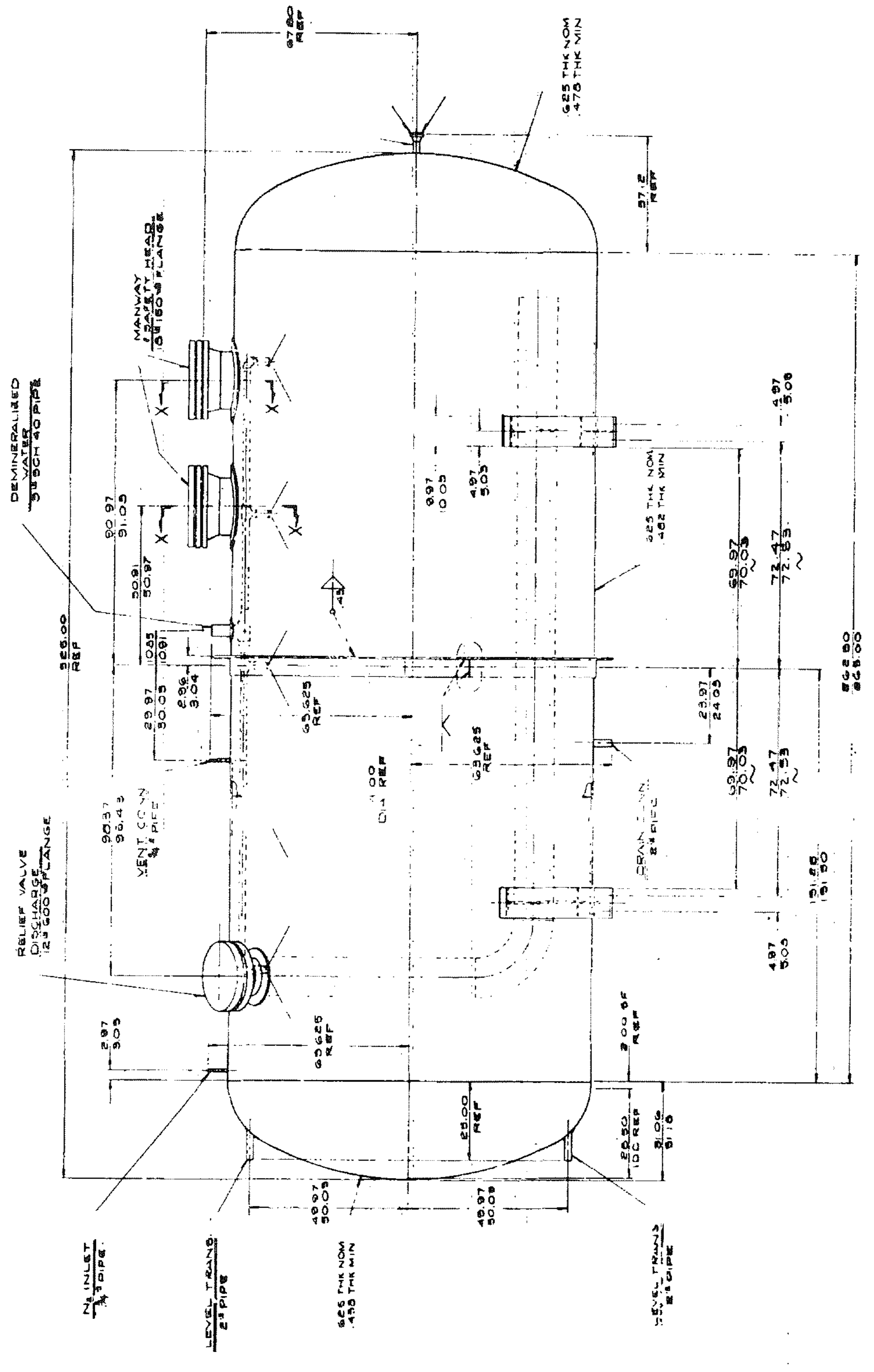


INDIAN POINT UNIT No. 2

UFSAR FIGURE 4.2–9

REACTOR COOLANT PUMP FLYWHEEL TANGENTIAL STRESS VS RADIUS

MIC. No. 1999MC3740	REV. No. 17A

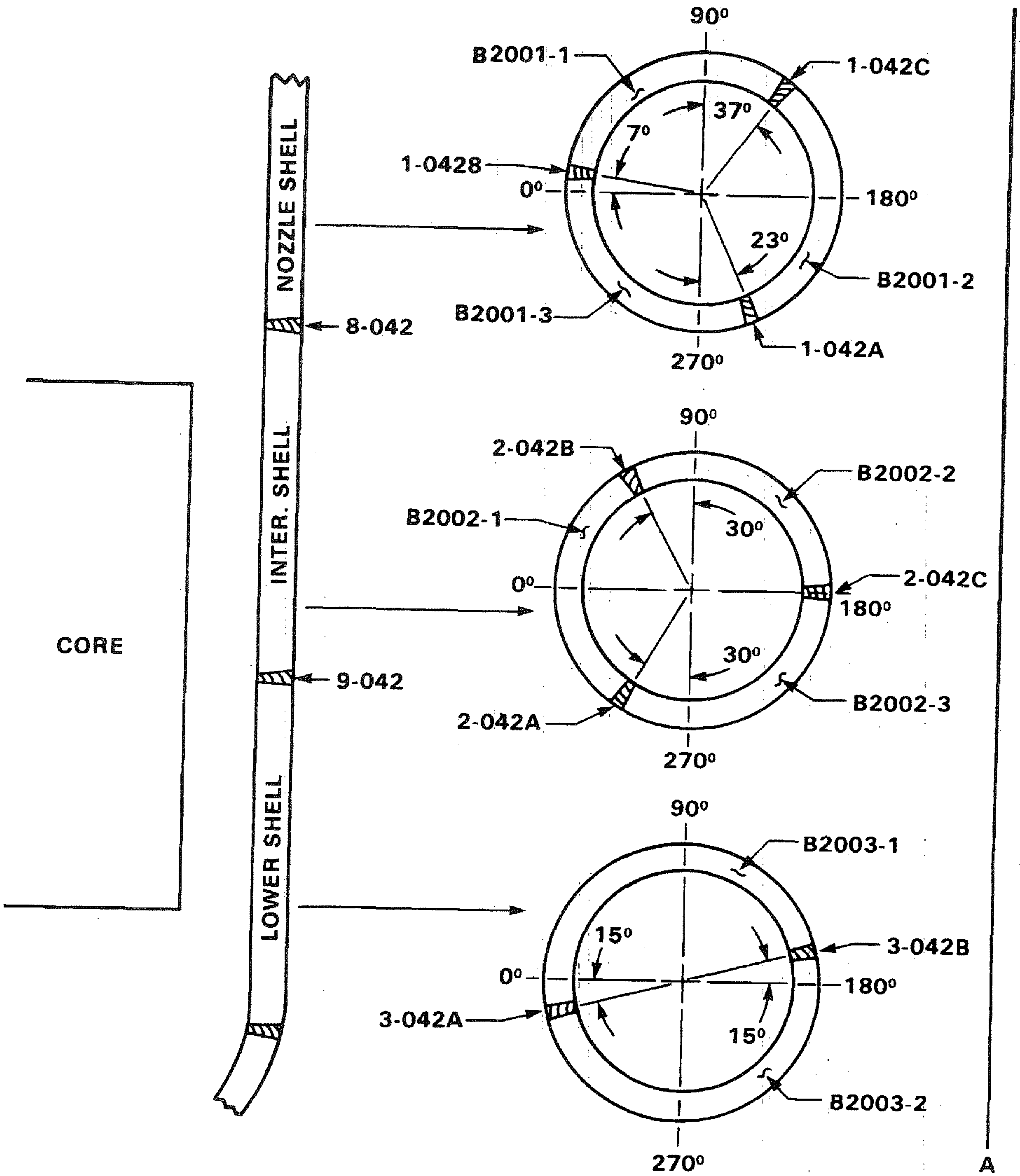


INDIAN POINT UNIT No. 2

UFSAR FIGURE 4.2-10

PRESSURIZER RELIEF TANK

MIC. No. 1999MC3741	REV. No. 17A

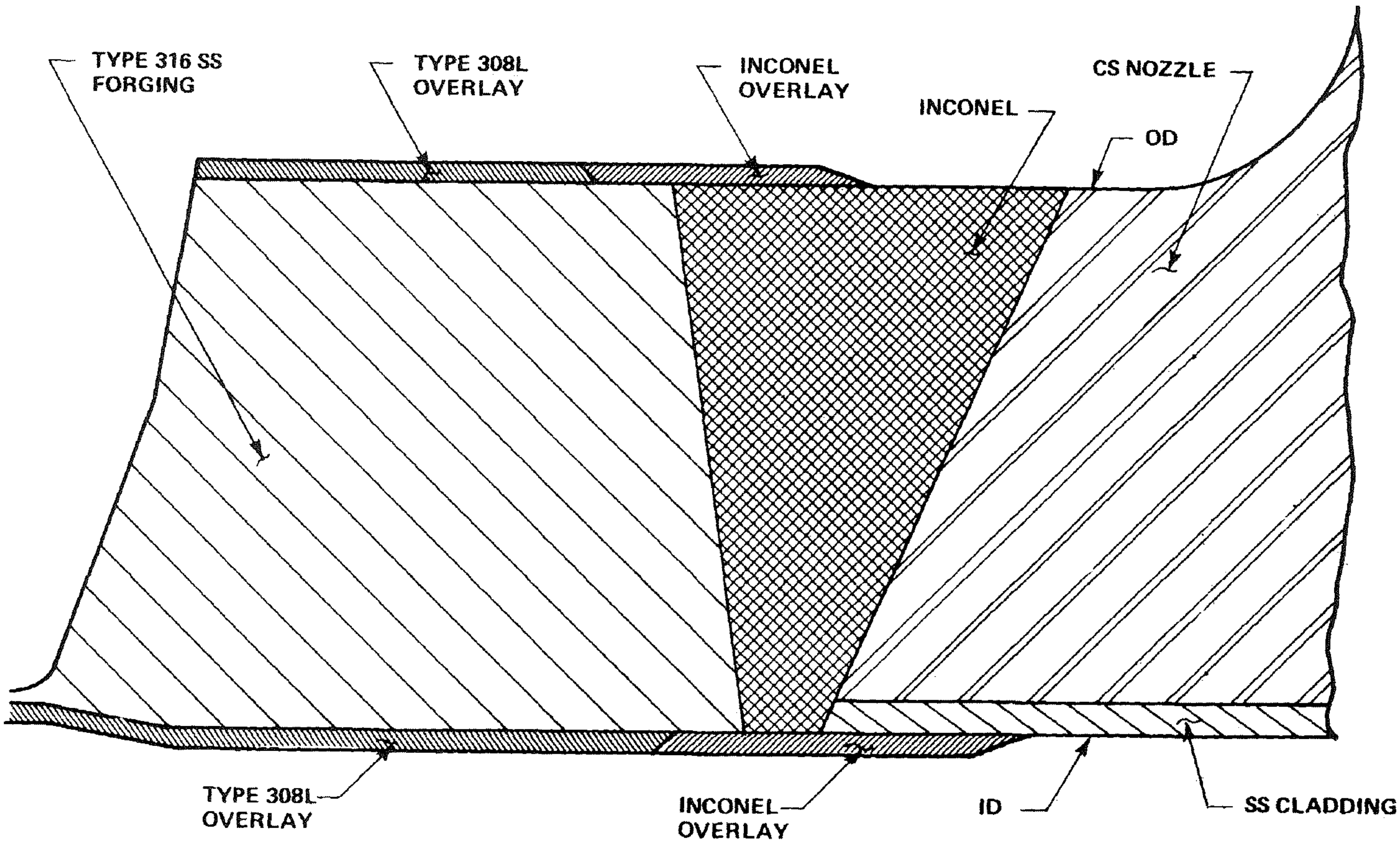


INDIAN POINT UNIT No. 2

UFSAR FIGURE 4.2–11

IDENTIFICATION & LOCATION OF BELTLINE REGION MATERIAL FOR THE INDIAN POINT UNIT 2 REACTOR VESSEL

MIC. No. 1999MC3742	REV. No. 17A

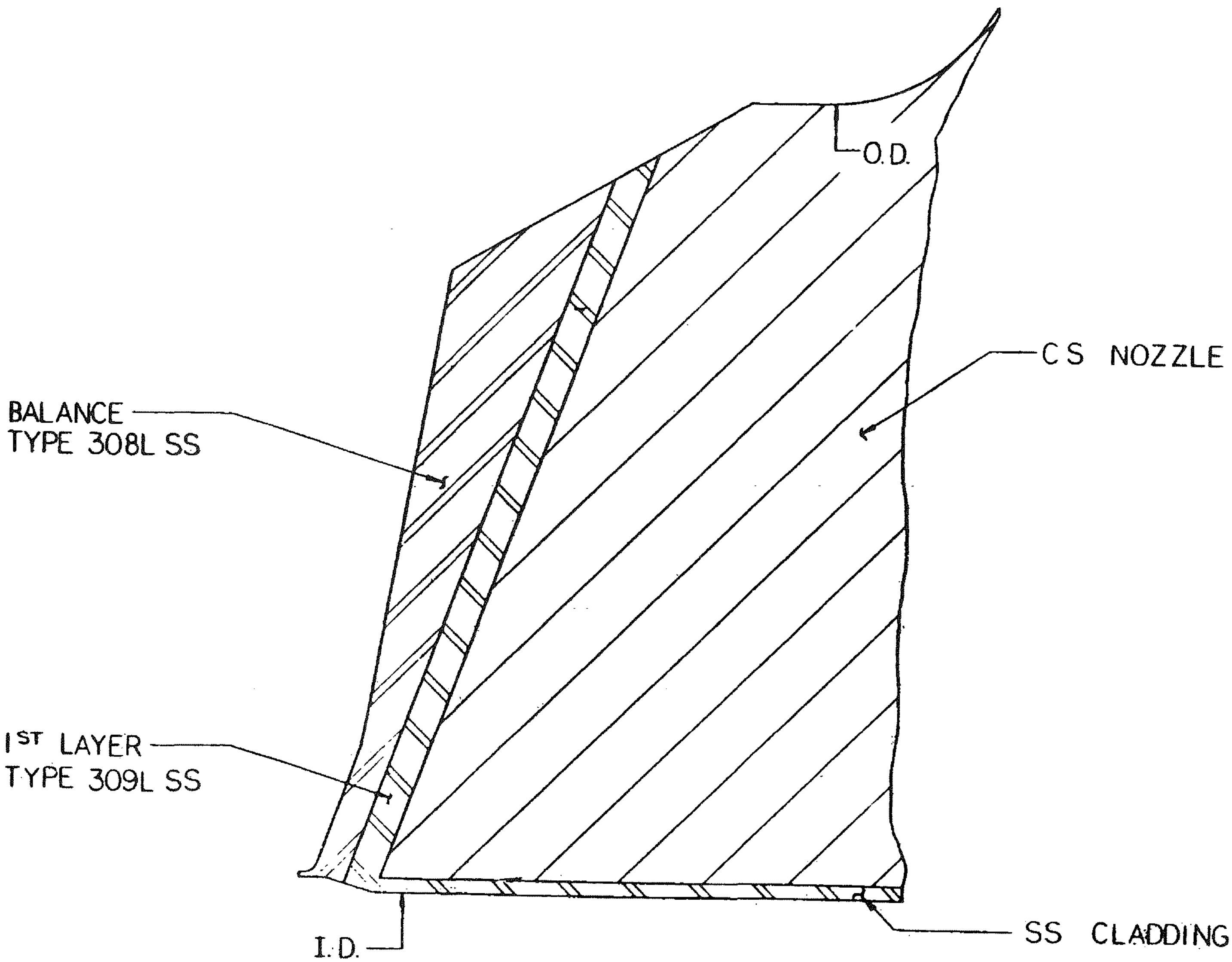


INDIAN POINT UNIT No. 2

UFSAR FIGURE 4C-1

PRIMARY NOZZLE
COMBUSTION ENGINEERING
REACTOR VESSEL

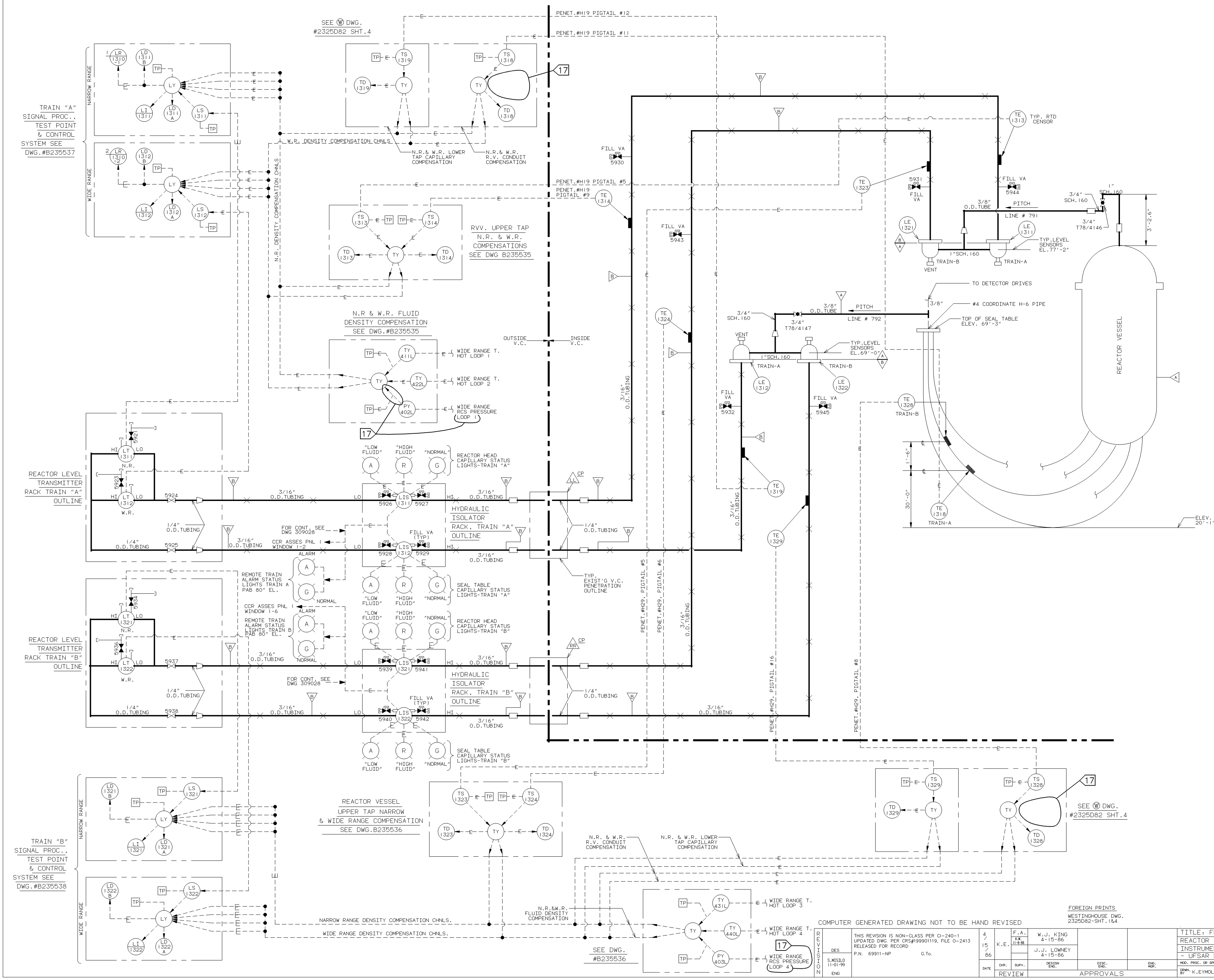
MIC. No. 1999MC3743	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 4C-2 PRIMARY NOZZLE TAMPA STEAM GENERATORS	
MIC. No. 1999MC3744	REV. No. 17A

2/21/2004 12:48:15 AM 208798-0-17.dwg

"CALL ANA X-3831"



LEGEND:

- FIELD MOUNTED INSTR.
- PANEL MOUNTED INSTR.
- FILLED CAPILLARY TUBING
- ELECTRIC SIGNAL LINE
- LEVEL SENSOR
- RESISTANCE TEMP. DETECTOR (RTD)
- INDICATING LIGHT

R.V. REACTOR VESSEL
N.R. NARROW RANGE
W.R. WIDE RANGE

CAPILLARY RTD	LENGTH
TE 1313	50 FEET
TE 1323	50 FEET
TE 1318	24 FEET
TE 1328	24 FEET
TE 1319	24 FEET
TE 1329	24 FEET
TE 1314	24 FEET
TE 1324	24 FEET

NOTES:

FOR ADDITIONAL NOTES & REFERENCE DWGS. SEE DRAWINGS A208077, A208799 & A208800.

1. PIPING AND TUBING SHALL BE IN ACCORDANCE WITH WESTINGHOUSE SPEC. #E-SPEC. G953174/0
2. LOCATE ISOLATORS CLOSE TO CONTAIMENT & AT THE SAME ELEVATION FOR EACH TRAIN. ROUTE CAPILLARIES FROM PIPING PENETRATION TO ISOLATORS IN MISSILE BARRIER, GUARD PIPE, ETC., TO AFFORD ADDED INTEGRITY TO CONTAINMENT PRESSURE BOUNDARY.
3. TRANSMITTER ELEVATIONS TO BE AT OR BELOW SEAL TABLE ELEVATION. TRANSMITTERS TO BE IN A "PROTECTED ENVIRONMENT" LOCATION TO PRECLUDE ENVIRONMENTAL INDUCED "ERRORS" UNDER ALL ACCIDENT CONDITIONS FOR WHICH REACTOR VESSEL LEVEL MAYBE REQUIRED.
4. VERTICAL RUNS OF CAPILLARIES FOR EACH TRAIN TO BE IN ENCLOSED RACEWAYS FOR THERMOLAGGING & THERMAL UNIFORMITY AFFORDED UNDER ACCIDENT TRANSIENT CONDITIONS. ROUTING SHALL AVOID PROXIMITY TO HIGH ENERGY PIPING. ROUTE RTD CABLE WITH CAPILLARY TO CONVENIENT POINT OF SEPARATION FOR ELECTRICAL CONNECTIONS.
5. VENT SENSOR HOUSING FOR REFUELING ETC. REFILL PIPING TO R.V. HEAD BY PUMPING WATER INTO REACTOR VESSEL. (CAUTION REFUELING DISCONNECTIONS AND SENSORS TO BE SEALED PRIOR TO FILLING OF REFUELING CANAL TO PREVENT PARTICULATE FOULING).
6. MAINTAIN SMALL FLUID VOLUME THROUGH PENETRATIONS 1/4" CAPILLARY TUBING MAXIMUM.
7. TRANSMITTERS & CONNECTING TUBING TO BE LOCATED OR ROUTED TO MINIMIZE VARIATIONS IN AMBIENT TEMPERATURES. AVOID PROXIMITY TO PIPING SUBJECT TO TEMPERATURE VARIATIONS.

THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN CON EDISON AS:
"CLASS A" ITEMS
PER CI-240-1

FOREIGN PRINTS
WESTINGHOUSE DWG. 2325D82-SHT.1&4

COMPUTER GENERATED DRAWING NOT TO BE HAND REVISED

REVISION	DES / ENG		DATE	CKR.	SUPV.	DESIGN ENG.	DISC. ENG.	ENG. MGR.
	S.MISILO 11-01-99	THIS REVISION IS NON-CLASS PER CI-240-1 UPDATED DWG. PER CRS#199901119, FILE 0-2413 RELEASED FOR RECORD P.N. 69911-NP G.To.	4/15/86	K.E.	F.A. A.W. 11-8-88	W.J. KING 4-15-86 J.J. LOWNEY 4-15-86		
				REVIEW		APPROVALS		

TITLE: FLOW DIAGRAM
REACTOR VESSEL LEVEL
INSTRUMENTATION SYSTEM
- UFSAR FIGURE NO. 4.2-12

MOD. PROC. OR SPEC.
DRWN. BY K.EYMOLD | SCALE NONE | REC'D

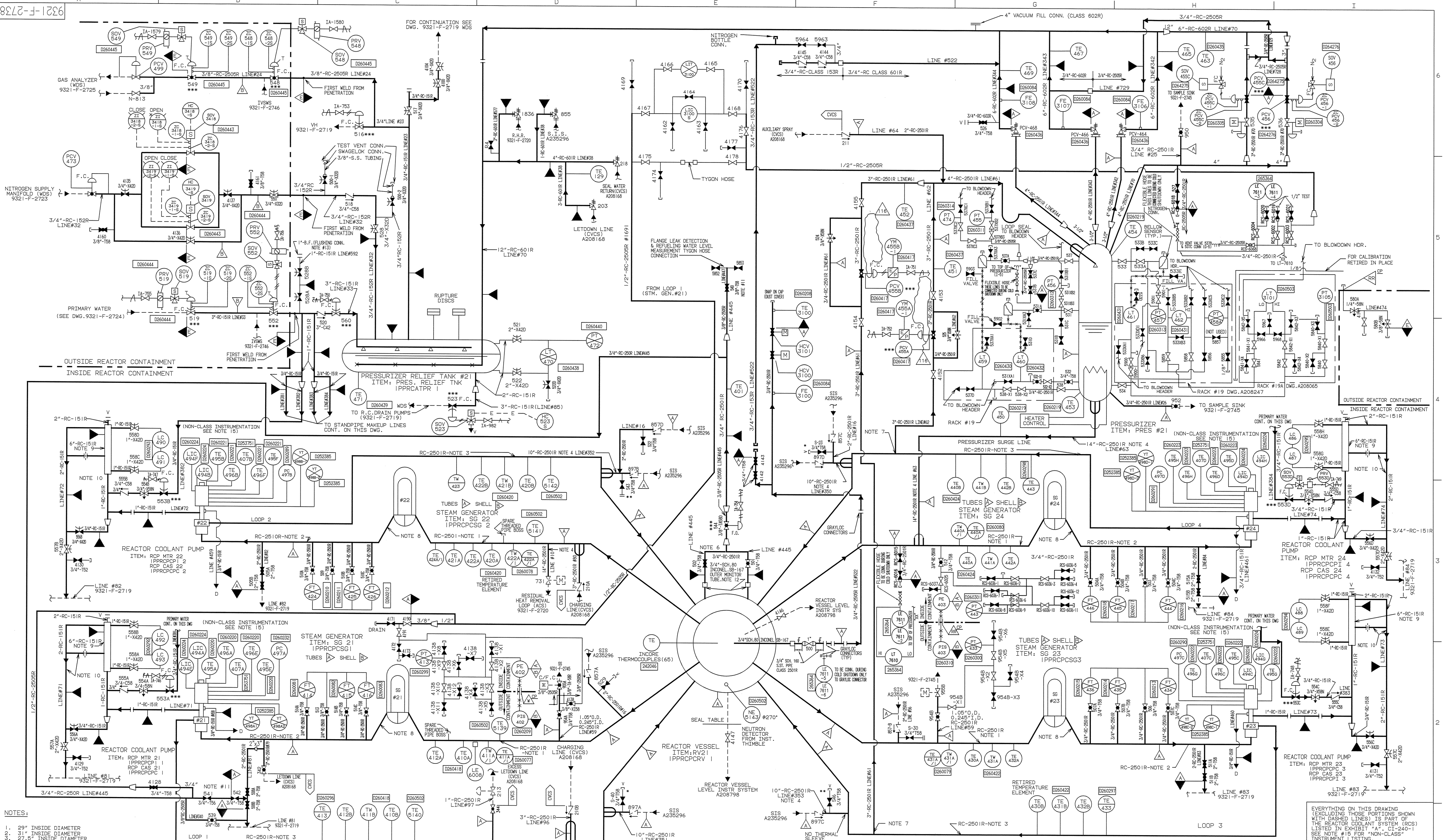
Con Edison

STATION
INDIAN POINT

DWG. NO. A208798-17 MY

MECH

9321-F-2738



EVERYTHING ON THIS DRAWING (EXCLUDING THOSE PORTIONS SHOWN WITH DASHED LINES) IS PART OF THE REACTOR COOLANT SYSTEM (RCS) LISTED IN EXHIBIT "A", CI-240-1 SEE NOTE #15 FOR "NON-CLASS" INSTRUMENT LISTING.

NOTES:

1. 29" INSIDE DIAMETER
2. 31" INSIDE DIAMETER
3. 27.5" INSIDE DIAMETER
4. SCHEDULE 140 PIPE
5. GLOBE VALVES NORMALLY INSTALLED WITH FLOW UNDER SEAT
6. LOCATE OUTSIDE SECONDARY SHIELD WALL VALVE NO.501 IS FROM INNER O-RING VALVE NO.502 IS FROM OUTER O-RING
7. SPRAY LINE SCOOP
8. ELBOW FLOW METERS
9. STAND PIPE IS SCH. 10S PIPE
10. PIPE SLOPED DOWNHILL TO DRAIN TANK
11. LOCATE VALVES 539, 540, 541 & 542 BELOW HOT LEG PIPING
12. TRANSITION WELD REQUIRED BETWEEN S.S. PIPING & INCONEL CONN. TO VESSEL
13. REFUELING CANAL FLUSH CONNECTION
14. *** INDICATES CONTROL VALVE HAS ADDITIONAL ASSOCIATED CONTROL EQUIPMENT & IS REPRESENTED ON CONTROL VALVE HOOK-UP DETAIL DWG. 9321-F-7056.
15. THE QUALITY GROUP A,B,C & SEISMIC BOUNDARIES EXTEND TO THE FIRST SEISMIC SUPPORT/RESTRAINT BEYOND THE BOUNDARIES SHOWN.

15. PER CLASSIFICATIONS 81-36,REV.1 & 89-090-CL,THE FOLLOWING INSTRUMENTATION IS "NON-CLASS."
16. ** SEE DETAIL #2 ON DWG NO. 206122
17. * SEE DETAIL #4 ON DWG. 206122
18. FOR LT-462 AND LT-3101: "HI" IS MEASURING (VARIABLE) LEG AND "LO" IS REFERENCE LEG.

RCP-#21	RCP-#22	RCP-#23	RCP-#24
LIC-494A	LIC-494B	LIC-494C	LIC-494D
LIC-494E	LIC-494F	LIC-494G	LIC-494H
PC-497A	PC-497B	PC-497C	PC-497D
TE-407A	TE-407B	TE-407C	TE-407D
TE-495A	TE-495B	TE-495C	TE-495D
TE-495E	TE-495F	TE-495G	TE-495H
TE-496A	TE-496B	TE-496C	TE-496D
TE-496E	TE-496F	TE-496G	TE-496H
YT-498A-1	YT-498B-1	YT-498C-1	YT-498D-1
YT-498A-2	YT-498B-2	YT-498C-2	YT-498D-2

REFERENCES:

1. PROCESS FLOW DIAGRAM (W) DWG.540-F-888
2. DEFINITION OF SYMBOLS
   (W) E SPEC. G675176 REV.2
3. INSTRUMENTATION AND CONTROL STANDARDS
   (W) SYMBOLS & APPLICATIONS FOR INSTRUMENT DIAGRAMS, SECTION 1.1, ISSUED AUG.12, 1966
   (W) INSTRUMENT INSTALLATION SECTION 3.0, ISSUED NOV.16, 1966
4. MATERIAL SPEC & FITTINGS
   (W) E. SPEC. G569866 REV.2
   (W) E. SPEC. G676398 REV.0

LEGEND

IVSWS - ISOLATION VALVE SEAL WATER SYS.
CVCS - CHEMICAL & VOLUME CONTROL SYS.
ACS - AUXILIARY COOLANT SYS.
SIS - SAFETY INJECTION SYS.
SS - SAMPLING SYS.
WDS - WASTE DISPOSAL SYS.

REFERENCE DWG.

FLOW DIAGRAM SYMBOLS---9321-C-2016
N_2 DWG.9321-F-2723
PIPING AT REACTOR COOLANT PUMP 9321-F-2734

REV	DESCRIPTION	DATE	BY	CHK'D	APP.
116	INCORPORATED DRN-05-05849 ER-05-2-114	5/15/06	DM	GH	N/A

REVISIONS

TITLE: FLOW DIAGRAM
REACTOR COOLANT
SYSTEM
- UFSAR FIGURE NO. 4.2-1

DRWN. BY GIBBS & HILL | SCALE NONE | REC'D

Entergy Nuclear Northeast

STATION INDIAN POINT

DWG. NO. 9321-F-2738-116 MY

CHAPTER 5
CONTAINMENT SYSTEMS

5.1 CONTAINMENT STRUCTURES

5.1.1 Design Basis

The reactor containment completely encloses the entire reactor and reactor coolant system and ensures 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 are designed to prevent any leakage through the containment. The structure provides biological shielding for both normal and accident situations.

The reactor containment is designed to safely withstand several conditions of loading and their credible combinations. The major loading conditions are:

1. Occurrence of a gross failure of the reactor coolant system, which creates a high-pressure and temperature condition within the containment.

2. Coincident failure of the reactor coolant system with an earthquake or wind.

5.1.1.1 Principal Design Criteria

5.1.1.1.1 Quality Standards

Criterion: Those systems and components of reactor facilities, which are essential to the prevention, or the mitigation of the consequences, of nuclear accidents, which could cause undue risk to the health and safety of the public shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes and standards pertaining to design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance criteria to be used shall be identified. An indication of the applicability of codes, standards, quality assurance programs, test procedures and inspection acceptance criteria used is required. Where such items are not covered by applicable codes and standards, a showing of adequacy is required. (GDC 1)

The containment system structure is of primary importance with respect to its safety function in protecting the health and safety of the public.

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–18-63. Further elaboration on quality standards of the reactor containment is given is Section 5.1.1.5.

5.1.1.1.2 Performance Standards

Criterion: Those systems and components of reactor facilities, which are essential to the prevention or to the mitigation of the consequences of nuclear accidents, which cause undue risk to the health and safety of the public shall be designed, fabricated, and erected to performance standards that enable such systems and components to withstand, without undue risk to the health and safety of the public, the forces that might reasonably be imposed by the occurrence of an extraordinary natural phenomenon such as earthquake, tornado, flooding condition, high wind or heavy ice. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been officially recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design. (GDC 2)

All components and supporting structures of the reactor containment are designed so that there is no loss of function of such equipment in the event of maximum potential ground acceleration acting in the horizontal and vertical directions simultaneously. The dynamic response of the structure to ground acceleration, based on the site characteristics and on the structural damping, is included in the design analysis. The reactor containment is defined as a Class I structure for purposes of seismic design (Section 1.11). Its structural members have sufficient capacity to accept, without exceeding specified stress limits, a combination of normal operating loads, functional loads due to a loss-of-coolant accident, and the loadings imposed by the maximum potential earthquake.

5.1.1.1.3 Fire Protection

Criterion: A reactor facility shall be designed to ensure that the probability of events such as fires and explosions and the potential consequences of such events will not result in undue risk to the health and safety of the public. Noncombustible and fire resistant materials shall be used throughout the facility wherever necessary to preclude such risk, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.(GDC 3)

Fire protection in all areas of the nuclear electric plant is provided by structure and component design that optimizes the containment of combustible materials and maintains exposed combustible material below the ignition temperature. The station is designed on the basis of limiting the use of combustible materials in construction by using fire-resistant materials to the greatest extent practical. Containment liner thermal insulation does not support combustion. The bearing oil systems for the reactor coolant pump motors are self-contained.

All oil-containing equipment associated with the reactor coolant pump motors is also completely enclosed by an oil-collecting system, which in the event of an oil leak, will contain and channel away the oil to remote storage containers.

5.1.1.1.4 Records Requirement

Criterion: The reactor licensee shall be responsible for assuring the maintenance throughout the life of the reactor of records of the design, fabrication, and

construction of major components of the plant essential to avoid undue risk to the health and safety of the public. (GDC 5)

Records of the design, fabrication, construction, and testing of the reactor containment are maintained throughout the life of the reactor.

5.1.1.1.5 Reactor Containment

Criterion: The containment structure shall be designed (a) to sustain, without undue risk to the health and safety of the public, the initial effects of gross equipment failures, such as a large reactor coolant pipe break, without loss of required integrity, and (b) together with other engineered safety features as may be necessary, to retain for as long as the situation requires, the functional capability of the containment to the extent necessary to avoid undue risk to the health and safety of the public. (GDC 10).

The design pressure and temperature of the containment exceeds the peak pressure and temperature occurring as the result of the complete blowdown of the reactor coolant through any rupture of the reactor coolant system up to and including the hypothetical double-ended severance of a reactor coolant pipe. Energy contribution from the steam system is included in the calculation of the containment pressure transient due to reverse heat transfer through the steam generator tubes. The supports for the reactor coolant system are designed to withstand the blowdown forces associated with the sudden severance of the reactor coolant piping so that the coincidental rupture of the steam system is not considered credible.

In 1989, the NRC approved changes to the design bases with respect to dynamic affects of postulated primary loop pipe ruptures, as discussed in Section 4.1.2.4.

The containment structure and all penetrations are designed to withstand, within design limits, the combined loadings of the design-basis accident and design and maximum potential seismic conditions.

All piping systems that penetrate the vapor barrier are anchored at the liner. The penetrations for the blowdown, and sample lines are designed so that the penetration is stronger than the piping system and so that the vapor barrier is not breached due to a hypothesized pipe rupture. The pipe rupture loads for the main steam and feedwater lines are resisted by the supports located away from their penetrations and do not affect the integrity of the penetrations for these lines. The pipe capacity in flexure is assumed to be limited to the plastic moment, based upon the yield strength of the pipe material. All lines (with the exception of sample tubing) connected to the primary coolant system that penetrate the vapor barrier are also restrained near the secondary shield walls (i.e., walls surrounding the steam generators and reactor coolant pumps) and are each provided with at least one valve between the shield wall and the reactor coolant system. These restraints are designed to withstand the thrust, moment, and torque resulting from a hypothesized rupture of the attached pipe.

All isolation valves are supported to withstand, without impairment of valve operability, the combined loadings of the design basis accident and design and maximum potential seismic conditions.

Section 5.1.5 includes a discussion of the details of the design of primary system supports. In addition, the design pressure will not be exceeded during any subsequent long-term pressure

transient determined by the combined effects of heat sources, such as residual heat and limited metal-water reactions, structural heat sinks, and the operation of the engineered safeguards, which uses only the emergency electric power supply.

5.1.1.1.6 Reactor Containment Design Basis

Criterion: The reactor containment structure, including openings and penetrations, and any necessary containment heat removal systems, shall be designed so that the leakage of radioactive materials from the containment structure under conditions of pressure and temperature resulting from the largest credible energy release following a loss-of-coolant accident, including the calculated energy from metal-water or other chemical reactions that could occur as a consequence of failure of any single active component in the emergency core cooling system, will not result in undue risk to the health and safety of the public. (GDC 49)

The following general criteria are followed to ensure conservatism in computing the required structural load capacity:

1. In calculating the containment pressure, rupture sizes up to and including a double-ended severance of reactor coolant pipe are considered.

2. In considering postaccident pressure effects, various malfunctions of the emergency systems are evaluated. Contingent mechanical or electrical failures are assumed to disable one of the diesel generators, two of the five fan-cooler units, and one of the two containment spray units. Equipment, which can be run from diesel power is described in Chapter 8.

3. The pressure and temperature loadings obtained by analyzing various loss-of-coolant accidents, when combined with operating loads and maximum wind or seismic forces, do not exceed the load-carrying capacity of the structure, its access opening or penetrations.

The most stringent case of these analyses is summarized in Section 14.3.5.3.7.

5.1.1.1.7 Nil-ductility Transition Temperature Requirement for Containment Material

Criterion: The selection and use of containment materials shall be in accordance with applicable engineering codes. (GDC 50).

The selection and use of containment materials comply with the applicable codes and standards tabulated in Section 5.1.1.5.

The concrete containment is not susceptible to a low-temperature brittle fracture.

The containment liner is enclosed within the containment and thus is not exposed to the temperature extremes of the environs. The containment ambient temperature during operation is between 50 and 130°F. This includes both hot operating and cold shutdown conditions. The Containment liner was specified for impact testing at a temperature of 30°F below the minimum service temperature of 50°F. The large Containment steel penetrations, equipment hatch and personnel lock, were specified for impact testing at -50°F which is more than 30°F below the

outside containment temperature of -5°F. These tests assure that the Nil Ductility Transition Criterion of GDC 50 is met.

5.1.1.2 Supplementary Accident Criteria

Systems relied upon to operate under postaccident conditions, which are located external to the containment and communicate directly with the containment, are considered to be extensions of the leakage boundary.

The pressure retaining components of the containment structure are designed for the maximum potential earthquake ground motion of the site combined with the simultaneous loads of the design basis accident as follows:

1. The liner is designed to ensure that no average strains greater than the strain at the guaranteed yield point occur at the factored loads. In regions of local stress concentrations or stresses due to localized secondary load effects, the liner is permitted to yield but the maximum liner strain is limited to 0.5-percent.

2. The mild steel reinforcement is designed to ensure that no strains greater than the strain at the guaranteed yield point occur at a cross section under the factored loads. The local yielding of reinforcing bars are permitted around the large openings for load combinations that include seismic loads.

The pressure-retaining components of containment subject to deterioration or corrosion in service are provided with appropriate protective means or devices (e.g., protective coatings).

5.1.1.3 Energy and Material Release

The design pressure is not exceeded during any subsequent long-term pressure transient determined by the combined effects of heat sources such as residual heat and metal-water reactions, structural heat sinks, and the operation of other engineered safety features utilizing only the emergency onsite electric power supply. The mass and energy releases to and the accident pressure and temperature effects on the containment structures, are those created by the hypothetical large break loss-of-coolant accident as presented in Section 14.3.5.

The following loadings are considered in the design of the containment in addition to the pressure and temperature conditions described above:

1. Structure dead load.
2. Live loads.
3. Equipment loads.
4. Internal test pressure.
5. Earthquake.
6. Wind.

5.1.1.4 Engineered Safety Features Contribution

Five types of engineered safety features are included in the design of this facility to ensure containment integrity. These systems are discussed in Chapter 6 and their effectiveness is analyzed in Chapter 14.

5.1.1.5 Codes and Standards

The design, materials, fabrication, inspection, and proof testing of the containment vessel complies with the applicable parts of the following codes and standards.

Code		Title
1.	ASTM A-333, Gr. 1	Specification for Seamless and Welded Steel Pipe for Low Temperature Service
2.	ASTM A-181	Forged or Rolled Steel Pipe Flanges, Forged Fittings, and Valves and Parts for General Service
3.	ASTM A-300, Cl. 1, Firebox	Specification for Notch Toughness Requirements for Normalized Steel Plates for Pressure Vessels
4.	ASTM A-201, Gr. B	Specification for Carbon Silicon Steel Plates of Intermediate Tensile Ranges for Fusion Welded Boilers and other Pressure Vessels
5.	ASTM A-36	Specification for Structural Steel
6.	ASTM A-131, Gr. C	Specification for Structural Steel for Ships
7.	ASTM A-240	Specification for Heat-Resisting Chromium and Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Fusion-Welded Unfired Pressure Vessels
8.	ASTM A-312	Specification for Seamless and Welded Austenitic Stainless Steel Pipe
9.	ASTM A442, Grade 60	Specifications for Pressure Vessel Plates, Carbon Steel, Improved Transition Properties
10.	ASME Boiler and Pressure Nuclear Vessels Vessel Code-Section III	Nuclear Vessels
11.	ASME Boiler and Pressure Unfired Pressure Vessels Vessel Code-Section VIII	Unfired Pressure Vessels
12.	ASME Boiler and Pressure Welding Qualifications Vessel Code-Section IX	Welding Qualifications
13.	ASTM C-33	Standard Specifications for Concrete Aggregates
14.	ASTM C-150	Standard Specifications for Portland Cement
15.	ASTM C-172	Standard Method of Sampling Fresh Concrete
16.	ASTM C-31	Standard Method of Making and Curing Concrete Compression and Flexure Test Specimens in the Field
17.	ASTM C-39	Standard Method of Test for Compressive Strength of Molded Concrete Cylinders
18.	ASTM-C-350	Specifications For Fly Ash For Use As AN Admixture in Portland Cement Concrete
19.	ASTM C-94	Specifications for Ready Mixed Concrete

20.	ASTM C-42	Standard Methods of Securing, Preparing, and Testing Specimens from Hardened Concrete for Compressive and Flexural Strengths
21.	ASTM C-494	Specifications for Chemical Admixtures for Concrete
22.	ASTM A-305	Specifications for Minimum Requirements for Deformations of Deformed Steel Bars for Concrete Reinforcement
23.	ASTM A-408	Specifications for Special Large Size Deformed Billet-Steel Bars for Concrete Reinforcement
24.	ASTM A-432	Specification for Deformed Billet Steel Bars for Concrete Reinforcement with 60,000 psi Minimum Yield Strength
25.	Research Council of Riveted and Bolted Structural Joints of the Engineering Foundation	Specification For Structural Joints Using ASTM A-325 Bolts
26.	ACI-613	Recommended Practice for Selecting Proportions for Concrete
27.	ACI-306	Recommended Practice for Winter Concreting
28.	ACI-318, Part IV-B	Structural Analysis and Proportioning of Members-Ultimate Strength Design
29.	ACI-318	Building Code Requirements for Reinforced Concrete
30.	ACI- 505	Specification for the Design and Construction of Reinforced Concrete Chimneys
31.	ACI-315	Manual of Standard Practice for Detailing Reinforced Concrete Structures
32.	ASA N6.2	Safety Standards for the Design, Fabrication and Maintenance of Steel Containment Structures for Stationary Nuclear Power Reactors
33.	ASA A58.1	American Standard Code Requirements for Minimum Design Loads in Buildings and Other Structures
34.		State Building and Construction Code for the State of New York
35.	SSPC-SP-6	Commercial Blast Cleaning

5.1.2 Containment Structure Design

5.1.2.1 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 contain all radioactive material, which might be released from the core following a loss-of-coolant accident. The structure serves as both a biological shield and a pressure container.

The structure, as shown on Plant Drawings 9321-2501, 9321-2502, 9321-2503, 9321-2506, 9321-2507, 9321-2508, [Formerly UFSAR Figures 5.1-2 through 5.1-7] and Figures 5.1-1 consists of side walls measuring 148-ft from the liner on the base to the springline of the dome, and has an inside diameter of 135-ft. The side walls for the cylinder and the dome are 4-ft 6-in. and 3-ft 6-in. thick respectively. The inside radius of the dome is equal to the inside radius of the cylinder so that the discontinuity at the springline due to the change in thickness is on the outer surface. The cylindrical part of the liner is substantially round. The difference between the minimum and maximum inside diameters at any selected cross section does not generally exceed 0.25-percent of the nominal diameter at that elevation. Between elevations 43-ft and 95-ft, the maximum diameter of any cross section is 135-ft 2-in., and the minimum diameter is 134-ft 10-in. except at the liner closing the temporary opening in the northwest quadrant where a minimum diameter of 134-ft 8-5/8-in. was measured. This portion of the liner was erected after all exterior concrete work was completed and is within the local buckle allowance of the liner plates. Above elevation 95 ft the tolerance on inside diameter does not exceed 0.50-percent of the nominal diameter of the selected cross section. The liner is erected true and plumb so that the deviation does not exceed 1/500 of the height at the selected cross section (allowing for 2-in. local buckling of the liner plates).

Particular care is taken in matching edges of cylindrical and hemispherical sections to ensure that all joints are properly aligned. Maximum permissible offset of completed joints is 25 percent of nominal plate thickness. Plates buckled beyond acceptable limits are cut out and replaced with new plates.

The flat concrete base mat is 9-ft thick with the bottom liner plate located on top of this mat. The bottom liner plate is covered with 3-ft of concrete, the top of which forms the floor of the containment.

Where uplift from pressure occurs at the outer areas of the mat, the 9-ft thick mat has sufficient flexural capacity to resist the uplift.

No hydraulic uplift exists since the bottom elevation of the mat is considerably higher than that of the high water level.

The large mass of the containment including interior concrete and equipment makes the structure inherently stable from overturning due to seismic motion.

In addition, keying action from the reactor pit and sumps, plus friction between the concrete and rock, prevents a sliding of the structure from horizontal ground motion.

The basic structural elements considered in the design of the containment structure are the base slab, side walls, and dome acting as one structure under all possible loading conditions. The liner is anchored to the concrete shell by means of stud anchors. The lower portions of the cylindrical liner are insulated to avoid thermal deformation of the liner under accident conditions.

The containment structure is inherently safe with regard to common hazards such as fire, flood, and electrical storm. The thick concrete walls are invulnerable to fire and only an insignificant

amount of combustible material, such as lubricating oil in pump and motor bearings, is present in the containment.

Internal structures consist of equipment supports, shielding, reactor cavity and canal for fuel transfer, and miscellaneous concrete and steel for floors and stairs. All internal structures are supported on the mat with the exception of equipment supports secured to the intermediate floors.

A 3-ft thick concrete ring wall serving as a missile and partial radiation shield surrounds the reactor coolant system components and supports the polar-type reactor containment crane. A 2-ft thick reinforced concrete floor covers the reactor coolant system with removable gratings in the floor provided for crane access to the reactor coolant pumps. The four steam generators, pressurizer, and various piping penetrate the floor. Spiral stairs provide access to the areas below the floor.

The refueling canal connects the reactor cavity with the fuel transport tube to the spent fuel pool. The floor and walls of the canal are concrete, with wall and shielding water providing the equivalent of 6-ft of concrete.

The refueling canal floor is 5-ft thick. The concrete walls and floor are lined with 0.25-in. thick stainless steel plate. The linings provide a leakproof membrane that is resistant to abrasion and damage during fuel handling operation.

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.

5.1.2.2 Design Load Criteria

The following loads are considered to act upon the containment structure creating stresses within the component parts.

1. Dead load consists of the weight of the concrete wall, dome, liner, insulation, base slab, and the internal concrete. Weights used for dead load calculations are as follows:

   a. Concrete — 150 lb/ft^3

   b. Reinforcing steel — 490 lb/ft^3 using nominal cross-sectional areas of reinforcing as defined in ASTM for bar sizes.

   c. Steel lining — 490 lb/ft^3 using nominal cross-sectional area.

   d. Insulation — 6 lb/ft^3 including stainless steel jacket.

2. Live load consists of snow and construction loads on the dome and major components of equipment in the containment. Snow and ice loads are assumed to be applied uniformly to the top surface of the dome at an estimated value of 20 lb/ft^2 of horizontal projection of the dome. This loading represents approximately

2-ft of snow, which is considered to be a conservative amount since the slope of the dome will tend to cause much of the snow that falls on it to slide off. A construction live load of 50 lb/ft^2 has been used on the dome, but will not be considered to act concurrently with the snow load. Equipment loads are considered as specified on the drawings supplied by the manufacturers of the various pieces of equipment.

Design live loads inside the containment building are as follows:

a. Elevation 68-ft-0-in. — 10-ft strip adjacent to crane wall = 600 psf Remaining strip = 100 psf

b. Elevation 95-ft-0-in. — Concrete slab = 500 psf Grating areas = 100 psf

3. The internal pressure transient used for the containment design and its variation with time is based on a postulated large break LOCA of 47 psig and liner temperature of 247°F. For the free volume of 2,610,000-ft^3 within the containment, the design pressure is 47 psig. This pressure transient is more severe than those calculated for various loss-of-coolant accidents, which are presented in Section 14.3.

4. Thermal expansion stresses due to an internal temperature increase caused by a loss-of-coolant accident is considered. The maximum temperature at the uninsulated section of the liner under accident conditions is 247°F. For the 1.25 times and 1.50 times design pressure loading conditions given in Section 5.1.2.4, the corresponding liner temperatures will be 285°F and 306°F respectively. The minimum external ambient design temperature, averaged over a 24 hour period, is 0°F. The liner maximum temperature following a loss-of-coolant accident with an outside air temperature of 0°F was calculated to be less than 247°F at the Stretch Power Uprate (SPU) power rating of 3216 MWt for the core. The initial containment air temperature in the SPU analysis for the liner temperature was set to 110°F, which is the maximum expected operating temperature at 100% power with an outside temperature of 0°F.

5. The ground acceleration for the design earthquake has been determined to be 0.1g applied horizontally and 0.05g applied vertically. These values have been resolved as conservative numbers based upon recommendations from Dr. Lynch, Director of Seismic Observatory, Fordham University.

   A dynamic analysis is used to arrive at equivalent design loads. Additionally, a hypothetical ground acceleration of 0.15 g horizontal and 0.10 g vertical is used to analyze for the no-loss-of-function. This is discussed in Section 5.1.3.11, Seismic Design.

   Due to symmetry of the containment structure, torsional loads generated by an earthquake are insignificant and have not been considered.

   Tornado loads have not been considered in the design of the Unit 2 containment; however, the seismic bars provide a more than adequate mechanism to

withstand the torsional effect if it were to occur. An evaluation of the effect of tornado loads on the containment structure is presented in Appendix B of the Containment Design Report.

6. The American Standards Association "American Standard Code Requirements for Minimum Design Loads in Buildings and Other Structures" (A58.1-1955) designates the site as being in a 25 psf zone for wind loads. In this code, for height zones between 100 and 499-ft, the recommended wind pressure on a flat surface is 40 psf. Correcting for the shape of the containment by using a shape factor of 0.60, the recommended pressure becomes 24 psf. The state building and construction code for the State of New York stipulates a wind pressure up to 30 psf on a flat surface for heights up to 300 feet. For design, a 30 psf basic wind load has been used from ground level up.

7. Internal pressure was applied to test the structural integrity of the containment shell up to 115-percent of the design pressure. For this structure, the test pressure is 54 psig. The containment is also structurally designed to withstand an external pressure 2.5 psig higher than the internal pressure.

5.1.2.3 Material Specifications

Basically five materials are used for the construction of the containment structure. These are:

1. Concrete.
2. Reinforcing steel.
3. Plate steel liner.
4. Insulation.
5. Protective Coating.

Basic specifications for these materials are as follows:

1. Concrete is a dense, durable mixture of sound coarse aggregate, fine aggregate, cement, and water. Cement conforms to ASTM, Specification C-150-65 "Standard Specification for Portland Cement," Type I (Normal), or Type II (moderate heat of hydration) requirements. Whenever high early strength is required, Type III Cement is used. Water is free from any injurious amounts of acid, alkali, salts, oil, sediment, or organic matter. The concrete has a minimum density of 150 lb/ft^3. The 28-day standard compressive strength of the concrete is 3000 psi. Adequate means of control are used in the manufacture of the concrete. To ensure the values of compressive strength are attained as a minimum, concrete samples are tested in accordance with the following ASTM Standards:

   ASTM C-172 - Standard Method of Sampling Fresh Concrete

   ASTM C-31 - Standard Method of Making and Curing Concrete Compression and Flexure Test Specimens in Field

   ASTM C-39 - Standard Method of Test for Compressive Strength of Molded Concrete Cylinders

All making and testing of concrete samples have been performed by Vacca Testing Laboratory and Research Company, Inc.

At certain specifically evaluated locations, non-structural surface type cracks and delaminations in the containment concrete have been repaired by injection of engineering approved epoxy grout. Although non-structural in nature, these repairs were performed in accordance with the requirements of IWL-4210 of the 1992 ASME Boiler and Pressure Vessel Code, Section XI, as applicable.

2. Reinforcing steel for the dome, cylindrical walls and base mat is high-strength, deformed billet steel bars conforming to ASTM Designation A432-65 "Specification for Deformed Billet Steel Bars for Concrete Reinforcement with 60,000 psi Minimum Yield Strength." This steel has a minimum yield strength of 60,000 psi, a minimum tensile strength of 90,000 psi, and a minimum elongation of 7-percent in an 8-in. specimen. Reinforcing bars No. 11 and smaller in diameter are lapped spliced in the mat for flexural loadings and spliced by the Cadweld process in the walls and dome for tension loading. Bars No. 14S and 18S are spliced by the Cadweld process only. A certification of physical properties and chemical content of each heat of reinforcing steel delivered to the job site has been issued from the steel supplier. The splices used to join reinforcing bars have been tested to ensure that they will develop at least 125-percent of the minimum yield point stress of the bar. The test program required cutting out, at random, approximately 3-percent, completed splices and testing to determine their breaking strength.

3. The plate steel liner is carbon steel conforming to ASTM Designation A442-65 "Standard Specification for Carbon Steel Plates with Improved Transition Properties," Grade 60. This steel has a minimum yield strength of 32,000 psi and a minimum tensile strength of 60,000 psi with an elongation of 22-percent in an 8-in. gauge length at failure.

   The liner is 0.25-in. thick at the bottom, 0.50-in. thick in the first three courses, except 0.75-in. thick at penetrations, a minimum of 0.34-in. in the general area at elevation 46-ft. due to past corrosion, and 0.375-in. thick for remaining portion of the cylindrical walls and 0.50-in. thick in the dome. The 0.34-in. minimum thickness affects the calculated stress levels presented in the Containment Design Report and the Containment Liner Stress Analysis Report. However, evaluation of the reduced minimum thickness has concluded that no design criteria are exceeded. The liner material has been tested to ensure an NDTT more than 30°F lower than the minimum operating temperature of the liner material.

   Impact testing has been done in accordance with Section N331 of Section III of the ASME Boiler and Pressure Vessel Code. A 100-percent visual inspection of liner anchors was made prior to pouring concrete.

4. The material used for the original insulation of the liner plate was polyvinylchloride with stainless steel jacket. This insulation has been selected to withstand the calculated temperature and pressure conditions associated with a postulated large break LOCA of 47 psig and liner temperature of 247°F. The carbon steel liner with an inorganic zinc protective coating makes contact with

the polyvinylchloride insulation, the stainless steel, and the sealant. However, these materials do not react with each other.

Because the insulation panels are jacketed with stainless steel and sealed at the joints, the insulation will not be subjected to the moisture and high humidity atmosphere of the containment during an accident.

Manufacturer's tests on the polyvinylchloride insulation indicated that the insulation was capable of withstanding periodic compression at 60 psig at temperatures from 40°F to 120°F and a single compression under accident conditions without any detriment or change to the insulation properties. The manufacturer's analog transient analysis indicated only a 5°F rise in liner temperature 1000 sec after an exposure to 310°F for the entire duration of the analysis. This provides a factor of safety of approximately 15 on specified tolerable temperature rise in the liner. A factor of safety of 2 is provided on specified insulation performance versus tolerable temperature rise in liner.

The maximum normal operating temperature of the containment was changed from 120°F to 130°F by Amendment 149 to the Facility Operating License DPR-26 for IP-2 dated March 27,1990. Evaluations performed show the insulation material used on the containment liner is adequate for use at the higher operating temperature.

For additional information on the liner insulation and the modifications made to it in 1973, see Section 5.1.7.

5. One 3 mil shop coat of Carbozinc No. 11 primer and one 4 mil minimum finish coat of Phenoline No. 305 as manufactured by the Carboline Company have been applied to the liner, as well as essentially all painted surfaces in containment, in accordance with the manufacturer's recommendations.

   The effect of the postaccident environment on protective coatings was conservatively evaluated for Indian Point Unit 2. The coatings showed no deterioration after a number of cycles. A more thorough discussion on the qualifications of the protective coatings applied during construction is presented in WCAP-7198-L.[1]

   In addition, various areas inside containment have been repaired and recoated with other DBA qualified coatings approved for use at Indian Point 2. Protective coatings used inside the containment are procured, applied, and maintained in compliance with Regulatory Guide 1.54 (June 1973), “Quality Assurance Requirements for Protective Coatings Applied to Water Cooled Nuclear Power Plants.” New quality requirements will be developed based on it’s provisions, but specific requirements, such as documented site meetings, field demonstrations, substrate priming, applicator reporting, inspection reporting and report forms will be considered on a job-by-job basis.

Quality of both materials and construction of the containment structure was ensured by a continuous program of quality control and inspection by Con Edison, and/or its field representatives, and Westinghouse Atomic Power Division, and United Engineers and Constructors Inc., as described in Section 5.1.2.6.

5.1.2.4 <u>Design Stress Criteria</u>

The design is based upon limiting load factors that are used as the ratio by which loads will be multiplied for design purposes to ensure that the loading deformation behavior of the structure is one of elastic, tolerable strain behavior. The load factor approach is being used in this design as a means of making a rational evaluation of the isolated factors, which must be considered in ensuring an adequate safety margin for the structure. This approach permits the designer to place the greatest conservatism on those loads most subject to variation and which most directly control the overall safety of the structure. In the case of the containment structure, therefore, this approach places minimum emphasis on the fixed gravity loads and maximum emphasis on accident and earthquake or wind loads. The loads utilized to determine the required limiting capacity of any structural element on the containment structure are computed as follows:

1. $C = 1.0D \pm 0.05D + 1.5\,P + 1.0\,(T + TL)$
2. $C = 1.0D \pm 0.05D + 1.25\,P + 1.0\,(T' + TL') + 1.25E$
3. $C = 1.0D \pm 0.05D + 1.0P + 1.0\,(T'' + TL'') + 1.0E'$

Symbols used in these formulae are defined as follows:

C	=	Required load capacity of section.
D	=	Dead load of structure and equipment loads.
P	=	Accident pressure load as shown on pressure-temperature transient curves.
T	=	Load due to maximum temperature gradient through the concrete shell and mat based upon temperature associated with 1.5 times accident pressure.
TL	=	Load exerted by the liner based upon temperatures associated with 1.5 times accident pressure.
T'	=	Load due to maximum temperature gradient through the concrete shell and mat based upon temperatures associated with 1.25 times accident pressure.
TL'	=	Load exerted by the liner based upon temperatures associated with 1.25 times accident pressure.
E	=	Load resulting from either design earthquake or wind, whichever is greater.
T"	=	Load due to maximum temperature gradient through the concrete shell and mat based upon temperatures associated with the accident pressure.
TL"	=	Load exerted by the liner based upon temperatures associated with the accident pressure.
E'	=	Load resulting from assumed hypothetical earthquake.

A chart for allowable versus actual stresses has been included in the Containment Design Report.

Load condition (1) indicates that the containment will have the capacity to withstand loadings at least 50-percent greater than those calculated for the postulated loss-of-coolant accident alone. Results of analysis using load condition (1) are shown in Figure 5.1-11.

Load condition (2) indicates that the containment will have the capacity to withstand loadings at least 25-percent greater than those calculated for the postulated loss-of-coolant accident with a coincident design earthquake. Results of analysis using load condition (2) are shown in Figure 5.1-12.

Load condition (3) indicates that the containment will have the capacity to withstand loads at least equal to those calculated for the postulated loss-of-coolant accident with a coincident hypothetical earthquake defined in Section 5.1.2.2. Results of analysis using load condition (3) are shown in Figure 5.1-13.

The mat has been analyzed using load conditions (1), (2) and (3) as shown in Figures 5.1-14 through 5.1-16 and also for loads occurring only at operating and test pressure conditions. For loads, see Table 5.1-1, Flooded Weights-Containment Building.

The loads resulting from wind on any portion of the structure do not exceed those resulting from earthquake.

The capacity of all structural components, with the minor exceptions of outer rebar at large containment openings addressed in Section 3.4.4 of the Containment Design Report, exceeds or is equal to the capacity required by the most severe loading combination. The loads resulting from the use of these equations will hereafter be termed "factored loads."

The load factors used in these equations are based upon the load factor concept employed in Part IV-B, "Structural Analysis and Proportioning of Members Ultimate Strength Design" of ACI –318-63. Because of the refinement of the analysis and the restrictions on construction procedure, the load factors in the design primarily provide for a safety margin on the load assumptions.

The design includes the consideration of both primary and secondary stresses. The design limit for tension member (i.e., the capacity required for the design load) is based upon the yield stress of the reinforcing steel.

The theoretical load carrying capacity of steel reinforced concrete cross-sections are reduced by a capacity reduction factor "∅", which provides for the possibility that small adverse variations in material strengths, workmanship, dimensions, and control, while individually within required tolerances and the limits of good practice, occasionally may combine to result in under-capacity. For tension members, the factor "∅" has been established as 0.95. The factor "∅" is 0.90 for flexure and 0.85 for diagonal tension, bond, and anchorage.

For principle compression and tension, the liner stresses are maintained below 0.95 specified minimum yield at normal operating temperature (i.e., ∅=0.95). For shear, the liner stresses are maintained below 0.6 yield.

The liner is designed to assure that no strains greater than the strain at the guaranteed yield point will occur at the factored loads. In regions of local stress concentrations or stresses due to localized secondary load effects, the liner is permitted to yield but the maximum liner strain is limited to 0.5-percent. Sufficient anchorage is provided to ensure elastic stability of the liner. The basic design concept for the liner stud anchorage is the ductility of the anchorage that assures stud failure due to shear, tension or bending stress without the stud connection causing failure or tear of the liner plate. References 2 and 3 provide information on design of stud connection. The studs in the 0.50-in. plate are installed on 24-in. horizontal and 28-in. vertical grid and in the 0.375-in. plate on a 24-in. horizontal and 14-in. vertical grid. Studs are centered between vertical bars. In the dome, 5-ft by 5-ft panels are anchored in the center by studs and by T–bars at the edges. The 0.50-in. diameter bent welding studs are 9-in. long minimum and 9.50-in. long maximum with a 2-in. 90 degree hook at the end. An arc stud welding process was used on all bent welding studs. The arc stud welding process produces a circular weld around the 0.50-in. diameter stud with a diameter (outside to outside of weld) equal to 0.678-in. and a height equal to 0.157-in. The design considers the possibility of daily stress reversals due to ambient temperature changes for the life of the plant, and fatigue limit of the studs exceeds the design requirements. However, to accommodate possible fatigue failure in the plate-to-stud weldment, the depth of penetration to the liner plate is controlled to avoid impairment of liner integrity.

The boundary conditions in the cylinder are determined by assuming a buckling model (shown in Figures 5.1-17 through 5.1-19) in which the studs form the low points and the center of the panels form the high points of a series of peaks and valleys thus forming a set of panels whose edges represent points of inflection. The analytical procedure used is a simply supported plate under biaxial compression. A Mohr's circle analysis is used to find the normal and shear stresses on this simply-supported plate. The critical buckling stress is derived considering a plate whose length is equal to one-half of the diagonal distance between studs. This critical buckling load is 38.1 ksi for the 0.375-in. liner and 38.4 ksi for the 0.50-in. liner, which is higher than the yield strength of the liner, 32 ksi; therefore, the liner plate will begin to yield before the critical buckling stress is reached, and buckling failure does not control the design. Since shear reduces the stability of a plate subjected to compressive stresses, critical shear is considered and it was found that critical buckling is controlled by normal stresses rather than shear stresses. This is determined by considering the magnitude of both the normal and the shear stresses on the panel. The magnitude of the shear is so low that it shows no effect on the previously stated critical buckling stresses.

In the dome the liner will be considered clamped at the stiffeners forming a 5-ft by 5-ft grid panel pattern. The center of each panel is fixed by a stud. Assuming points of inflection at the one-quarter point a distance of 1-ft 3-in. occurs between points of simple support. The critical buckling load is 58.1 ksi, which is also higher than the yield strength of the liner.

At maximum strain in the liner, the studs will not fail. This maximum strain due to an unbalanced load would occur in a panel adjacent to a buckled panel. Since this adjacent stud will not fail, no zipper effect will occur and massive buckling of the liner and mass failure of anchors is not credible.

The anchorages can fail by failure of the studs in shear or tension, by studs pulling out from the concrete, or by studs separating from the liner plate. The most likely mode of failure is by tensile failure of the stud. The anchors are designed so that failure occurs in the anchor rather

than the plate, thereby ensuring that the leaktight integrity of the containment liner will be maintained.

If failure should develop, it would be a random stud failure due to poor workmanship during stud attachment. This failure would not impair the liner integrity nor would it cause progressive failure.

The anchor must resist tensile and shearing loads. Tests have indicated that the lateral load needed to prevent column buckling is 1-percent of the axial yield load. Conservatively doubling this value to account for uncertain field conditions, a value of 2-percent is used.[4] The total load per plate would be 24-in. x 0.50-in. x 32,000 psi = 384,000 lb. Therefore, the tensile load per anchor is 384,000 lb x 0.02 = 7680 lb, which yields a stress of 7680/0.2 = 38,400 psi.

This compares with a yield value of 50,000 psi and a tensile strength of 60,000 psi in the studs. This does not consider the internal pressure, which provides further stability against buckling.

The shear load on the anchor is due to the strain in the liner. Assuming the liner approaches its yield strain of 0.1-percent, the anchor deflection would be 28-in. x .001 = .028-in. Tests on the stud anchor have shown a maximum deflection of about 0.1-in. can be tolerated before failure of the stud.

5.1.2.5 Missile Protection

Except for the upper portions of the steam generators and the Pressurizer, the high pressure reactor coolant system equipment is surrounded by the 3-ft concrete shield wall enclosing the reactor coolant loop and by the 2-ft concrete operating floor.

In 1989, the NRC approved changes to the design basis with respect to dynamic effects of postulated primary loop ruptures, as discussed in Section 4.1.2.4.

A structure is provided over the control rod drive mechanism to block any missiles generated from fracture of the mechanisms.

Systems containing hot pressurized fluids and that might affect the engineered safeguards components have been carefully checked against the possibility of being sources of missiles. The general criterion adopted has been to take provision, when necessary, against the generation of missiles rather than allow missile formation and try to contain their effects.

Once the design requirement that the above systems are not to be sources of missiles has been set forth, identification of potential deficiencies and generation of adequate fixes took place through the quality assurance program.

The following examples illustrate how this approach has been implemented.

5.1.2.5.1 Valves

Valves installed in the nuclear steam supply system have stems with back seat. This rules out the probability of ejecting valve stems; even if it were assumed that the stem threads fail, analysis shows that the back seat or the upset end cannot penetrate the bonnet and thereby become a missile. Additional interference is encountered with air- and motor-operated valves.

Valves with nominal diameter larger than 2-in. have been designed against bonnet-body connection failure and subsequent bonnet ejection by means of:

(1) following the EPRI recommendations[12] regarding bolting practices;
(2) using the design practice of ASME Section VIII for flange design; and
(3) by controlling the pre-load during the bonnet body connection stud tightening process.

The pressure-containing parts except the flange and studs are designed per criteria established by USAS B16.5. Flanges and studs are designed in accordance with ASME Section VIII. Piping and flange materials of construction are procured per ASTM A182, F316, or A351, GR CF8M.

Stud and nut material is ASTM A193-B7 and A194-2H. The proper stud torquing procedures and the use of a torque wrench, with indication of the applied torque, limit the stress of the studs to the allowable limits established in the EPRI Good Bolting Practices Reference Manual (NP-5067).[12] This stress level is far below the material yield, i.e., about 105,000 psi. The complete valves are hydro-tested per USAS B16.5 (1500 lb USAS valves are hydro-tested to 5400 psi). The cast stainless steel bodies and bonnets are radiographed and dye penetrant tested to verify soundness.

Valves with nominal diameter of 2-in. or smaller are forged and generally have screwed bonnet with canopy seal. The canopy seal is the pressure boundary while the bonnet threads are designed to withstand the hydrostatic end force. The pressure containing parts are designed per criteria established by the USAS B16.5 specification.

5.1.2.5.2 Reactor Coolant Pump Flywheel

The reactor coolant pump flywheel is not considered to be a credible source of missiles because of conservative design and care in manufacture and inspection. The flywheel material is ASTM A-533 having an nil-ductility transition temperature less than 10^{o}F. The design results in a primary stress less than 50-percent of the material yield strength at operating speed. The flywheel was subjected to 100-percent volumetric ultrasonic inspection, which is repeated at intervals during plant life. The finished machined bore is subjected to either magnetic particle or liquid penetrant examination. The design overspeed of the pump is 125-percent. The maximum pump overspeed on loss of external load is 112-percent. For an additional discussion of integrity of the reactor coolant pump flywheel, see Section 4.2.2.

5.1.2.6 Quality Control

To ensure a high degree of confidence in plant design, construction, workmanship, materials, and performance, a quality control program has been in effect for this project in which the following principal organizations have their respective responsibilities:

1. Consolidated Edison Company of New York, Inc. as initial owner and operator of the plant.

2. Westinghouse Electric Corporation as the turnkey plant contractor and supplier of major equipment.

3. United Engineers and Constructors Inc. as architect-engineer, construction manager, and constructor.

The function and responsibility in the quality control program of each of the above organizations is as follows:

5.1.2.6.1 Consolidated Edison Company of New York, Inc. (Con Edison) – Initial Licensee

A qualified field representative was assigned to the field during the construction period. His responsibilities included continuous inspection of the construction of the containment building to ensure that all materials used and work performed was strictly in accordance with the plans and specifications. The Con Edison representative, through instructions received from the home office, had the power to stop the construction until any discrepancies were corrected and the work once more was in compliance with the specifications and plans.

The Con Edison representative was in constant communication and consultation with the construction superintendent in matters regarding quality control. In addition, personnel from U.S. Testing Laboratories were assigned to this project to monitor the inspection of the construction and obtain samples of the materials for testing.

5.1.2.6.2 Westinghouse Electric Corporation

For the assurance of plant integrity and quality, Westinghouse performed the following functions regarding the containment building:

1. Reviewed and approved the containment design criteria, material specifications and detail design concepts before they were released for construction. This work was done by qualified structural engineers at the company's home office.

2. Reviewed the construction and inspection methods employed by United Engineers and Constructors Inc.

Westinghouse Pressurized Water Reactor Division, Nuclear Power Services Group had a field quality assurance representative in residence during the construction period. His function was the same as the Con Edison representative mentioned above. He reported discrepancies to the Westinghouse Construction and Services resident engineer who had the authority to stop the work until the discrepancy was resolved.

In addition to this, he audited the construction files, and verified that records were complete, accurate, and adequate for quality assurance.

Nuclear Power Service Headquarters quality assurance engineers also made trips to the site to audit, monitor, and review the project with regard to site quality assurance. Construction practices were observed for conformance to codes, specifications, and approved procedures.

5.1.2.6.3 United Engineers and Constructors Inc.

The responsibilities of United Engineers and Constructors Inc. in the quality control of the containment building were as follows:

1. They inspected all materials delivered to the job site, and examined the suppliers' certified test reports of physical and chemical properties for those components furnished by them.

2. They inspected fabrication of major components of the containment structure in the shop. Trip reports are available at the site.

3. They maintained an adequate force of qualified supervisory personnel at all times.

4. They supervised and were fully responsible for the quality of work performed by their subcontractors and for the craft labor employed and supervised by them.

5. They maintained as part of their field engineering force, qualified personnel who performed a thorough inspection of each construction operation.

No changes in design or specifications were allowed without the approval of the engineer in charge of design.

5.1.3 Containment Stress Analysis

5.1.3.1 General

The structural design of the containment meets the requirements established by 1961 edition of "The State Building and Construction Code for the State of New York" so far as these provisions are applicable. All concrete structures have been designed, detailed, and constructed in accordance with the provisions of "Building Code Requirements for Reinforced Concrete" (ACI 318-63) so far as these provisions are applicable.

5.1.3.2 Method of Analysis

Basically three separate structural components have been analyzed, each in equilibrium with loads applied to it and with constraints occurring at the juncture of the structures. The three components are:

1. The 135-ft ID hemispherical dome.
2. The 135-ft ID cylinder.
3. The base slab.

Mathematically, the dome and cylinder have been treated as thin-walled shell structures, which results in a membrane analysis. Since the thickness of the dome and cylinder is small in comparison with the radius of curvature (1/20 and 1/15) and there are no discontinuities such as sharp bends in the meridonal curves, the stresses due to pressure and wind or earthquake are calculated by assuming that they are uniformly distributed across the thickness.

Since the concrete is not assumed to resist any tensile or shear forces, radial shear reinforcing has been introduced in the lower portion of the wall in the form of hooked diagonal stirrups and diagonally bent bars as shown in Figure 5.1-1. Diagonal shear reinforcing, at 45° and 135° to the circumferential direction, are placed in the center of the cylinder wall for the full height of the wall and a distance above the springline into the dome to resist earthquake shears. The diagonal bars are discontinued in the upper area of the dome (beyond about 30 degrees above the springline), where the seismic shears are small and are carried by the dome reinforcing steel lying in the plane of principal tension.

The base slab has been treated as a flat circular plate supported on a rigid nonyielding foundation.

The limiting cases in the design of the wall for discontinuity moments and shears were considered. One case considered an uncracked wall and the other considered a cracked wall with the steel acting as a spring constant. The value of μ_c varied from zero in the cracked case to .14 in the uncracked case. In the uncracked case, variations in E_c will have no effect on the answer since E_c appears in both the numerator and the denominator of the stiffness formulation. For the above variation in E_c and μ_c, the values of discontinuity moment and shear vary by 14-percent and 7-percent respectively at the base. These are the maximum deviations of the wall forces since the wall will actually vary from uncracked to cracked with an increase in containment height rather than be cracked or uncracked for the total height.

In the area of thermal stress, the entire wall section will be cracked and no variation in E_c or μ_c need be considered. The liner stresses depend on the strains of the reinforcing steel and are not related to the concrete properties.

Shrinkage and creep effects will be relieved by cracking during the pressure test and will not be included in accident design considerations.

The finite element computer program has the capabilities of taking into account variations in μ_c and E_c and axisymmetric loads. However, it is not necessary to take into account the variations in μ_c and E_c for the reasons stated above.

The computer program used to study the general behavior of the structure and to generate boundary conditions was the axisymmetric shell structure program. This computer program, developed by Franklin Institute Research Laboratories, is designed to handle arbitrarily shaped shells of revolution subjected to axisymmetric as well as nonaxisymmetric loadings. The method of analysis consists of subdividing the shell into elements having continuous meridians with continuous first and second derivatives so that the first and second fundamental forms of the resulting shell elements are continuous throughout the element. By expanding the dependent variables in Fourier series in the circumferential direction, and assigning unspecified functions for the meridional variation, the independent variables are separated and a system of ordinary differential equations results for the dependent variables in terms of the meridional independent variable. Particular and complementary solutions of these ordinary differential equations are then found for each of the elements and each of the circumferential harmonics individually. The matching of the elements is achieved by writing the required boundary conditions.

The idealized section used with the axisymmetric shell structure program consists of five layers whose moment of inertia is equal to that of the actual section. The wall section is considered as cracked with the reinforcing carrying all loads.

A finite element program, with the capability to incorporate thermal loads, was used to analyze the containment shell considering the effect of the equipment hatch opening.

The shell was idealized into 10 layers with alternate layers of steel and concrete. Section 5.1.3.10 provides more information on the finite element analysis.

The computer program can handle the loads in the form of either surface traction or edge loads or both.

Analysis of the liner is presented in the Containment Liner Stress Analysis Report. The report also contains a description of analytical procedures arriving at forces, shears, and moments in the structural shell.

5.1.3.3 Dome Analysis

The analysis of the hemispherical dome has been performed by the super-position of membrane forces resulting from gravity, accident pressure, and accident thermal loads. In addition, earthquake or wind loading create both direct and shear stresses in the dome, and the operating temperature of the liner creates tension and compression. All of the combined direct stresses are developed in the reinforcing steel encased in the concrete. In the upper area of the dome (about 30 degrees above the springline), where the seismic shears are small, seismic shears are carried by dome reinforcing steel lying in the plane of the principal tension. The dome reinforcing is spliced to the vertical steel in the cylindrical concrete wall, so that a continuity between the dome and the cylinder is realized. See Figure 5.1-20 for a section of wall, dome and for reinforcing in the dome.

5.1.3.4 Cylinder Analysis

The analysis of the cylinder is by superposition of membrane forces resulting from gravity, pressure and thermal loads, overturning due to earthquake or wind and shears due to earthquake or wind. The concrete has been reinforced circumferentially using steel hoops and vertically by straight bars. Diagonal bars have been placed to resist the horizontal and vertical shears due to earthquake or wind. The required capacity of the diagonal bars has been designed so that the horizontal component per foot of the diagonals is equal to the maximum value of shear flow. A check was made to ensure that no net compressive force results in the diagonal bars because of the combination of seismic shear load and internal pressure load. Although, in the cylinder, the liner has some capacity available to resist the seismic shears, no credit is taken for this capacity.

For all of the cylinder and the lower areas of the dome, the diagonal reinforcing has been designed to accommodate all seismic shears. No credit has been taken for the dowel action of the vertical and horizontal bars in resisting seismic shear.

Only in the upper area of the dome (beyond about 30 degrees above the spring line) where the seismic shears are small is the liner counted on to resist shear. For all of the cylinder and the lower areas of the dome, the diagonal reinforcing has been designed to accommodate all seismic shears. No credit has been taken for the dowel action of the vertical and horizontal bars in resisting seismic shear.

5.1.3.5 Base Mat Analysis

The base slab was treated as a flat circular plate supported on a rigid non-yielding foundation. For loads applied uniformly around the slab, the analysis considers a 1-ft wide beam fixed at a point where the vertical shear is equal to zero. This is the point where the downward pressure on the mat and the dead weight overcome the uplift at the containment wall base mat juncture from pressure and earthquake loadings. Radial and circumferential reinforcing is provided at the top and bottom of the mat to resist moments in the areas where uplift occurs. Temperature

steel was added in other areas to meet requirements of the (ACI-318) Code. Diagonal tension reinforcement was added to meet requirements of ACI-318 Code. See Figure 5.1-23 for base slab reinforcing detail.

Moments and shears were calculated by writing equations for moment and shear in terms of X using the containment wall-base slab juncture as the origin with X increasing toward the center of the containment building. The point along the circumference of the containment wall chosen as the end of the beam is a point where the maximum tension from the earthquake will exist. Since the containment structure is considered a beam in all earthquake analyses, the maximum uplift for which the mat is designed will occur at only one point on the circumference and will represent the worst possible uplift on the mat.

All stresses were calculated using Part IV-B Structural Analysis and Proportioning of Members - Ultimate Strength Design of the Building Codes Requirements for Reinforced Concrete (ACI–318-63). No rebar stresses exceed 0.90 f_y.

A gradient with an operating temperature of 120°F inside the containment and a 50°F temperature at the mat-rock interface was considered and stresses were negligible. Ambient accident temperatures have no appreciable effect on the base slab. The maximum operating temperature of the containment is 130°F. The effect of elevated operating temperature on the structural elements was evaluated in 1987 and was found acceptable.

It is not possible to show that the design on nonyielding rock is more conservative than assuming the rock to be elastic. However, due to the installation of temperature reinforcing, the design is conservative. Reinforcing and concrete stresses are very low when considering the rock to be elastic.

To substantiate the above statement, the following studies were performed:

1. The foundation modulus were determined using the expression:[15]

$$k_Z = \frac{4Gr_O}{1-\mu}$$

where:

k_Z = The vertical spring constant of a circular base supported on an elastic foundation

$$G = \frac{E}{2\ (1+\mu)}$$

r_O = Radius of Foundation

μ = Poisson's Ratio

To obtain the foundation modulus, k_Z is divided by the area of the circular base to yield

$$k_o = \frac{k_z}{A} \times \frac{4\ G}{\pi r_o (1-\mu)}$$

Substituting for G

$$k_o = \frac{2E}{\pi r_o (1-\mu^2)}$$

2. The first case examined was that of a rectangular strip loaded with 1.5 times design accident pressure plus dead load using conservative properties for the Dolomitic limestone:[7,14]

$E = 6.0 \times 10^6$ psi

$\mu = 0$

Applying these values

$k_o = 4370$ lbs/in.^{3}

The "characteristic" λ is defined as:[6]

$$\lambda = \left[\frac{k}{4\ EI}\right]^{1/4}$$

Where:

E is the modulus of elasticity of the structural base (concrete),

I is the moment of inertia of the structural base,

$k = k_o b$, (b = width of base)

using base properties

$\lambda = 7.56 \times 10^{-3}$-in.$^{-1}$

Where $\lambda\ell > \pi$ beams may be considered as infinite in length.[6]

Taking the length of beam as being the base diameter

$\lambda\ell = 13.1 > \pi$

The beam was then analyzed as a beam of unlimited length loaded over an area equal to the base diameter with an 80 psi uniform load.

The solution to this problem gives

$$y_c = \frac{q}{2k}\left(2 - D_{\lambda a} - D_{\lambda b}\right)$$

$$M_c = \frac{q}{4\lambda^2}\left(B_{\lambda a} + B_{\lambda b}\right)$$

$$Qc = \frac{q}{4\lambda}\left(C_{\lambda a} - C_{\lambda b}\right)$$

where

y_C is deflection of point being considered

M_C is the moment at point being considered

Q_C is shear at point being considered

q is the uniform load

a is the distance from point under consideration to end of load

b is distance from point under consideration to other end of load.

$B_{\lambda x} = e^{-\lambda x} \sin \lambda x$

$C_{\lambda x} = e^{-\lambda x}(\cos \lambda x - \sin \lambda x)$

$D = e^{-\lambda x} \cos \lambda x$

Maximum moment occurs at mid-point of load and is equal to 352-in.-lbs/in.

For the area of the mat where there is only temperature reinforcing, the maximum moment would cause a stress of 30 psi in the reinforcing.

The maximum shear would occur at the ends and is equal to 2.64 kips/in. This shear would cause a shear stress in an unreinforced concrete section of 26.4 psi.

3. A second case examined was for the foundation material being less rigid than the concrete base. The model was the same for the first case:

Assumed $E_{rock} = 2.6 \times 10^6$ psi

$\mu = 0$

For this case, the following were determined:

$k_o = 1890$ lb/in.^{3}

$\lambda = 6.2 \times 10^{-3}$-in.$^{-1}$

M_{max} = 3.66-in.-kips/in.

Q_{max} = 3.23 kips/in.

S_{rebar} = 312 psi

ν_{conc} = 32.3 psi

As a final study, the maximum deflection as calculated in the first case was imposed as a settlement of the base mat for the outer portion and a section of the mat was analyzed for this settlement. A 30-ft section was used with fixity at the reactor pit, the remainder cantilevered from the pit.

The resulting moment and shear are as follows:

M = 142-in.-kips/in.

q = 396 lbs

resulting in a rebar stress of 12.2 ksi and a shear stress of 4.0 psi.

From the above, it can be seen that the assumption that a foundation on rock is a rigid unyielding foundation is a valid assumption and that temperature reinforcing provides much greater resistance than required to accommodate the effects of any elastic deformation of the subgrade.

5.1.3.6 Analysis of Liner and Reinforcing Steel

Approximately 67-percent of the inclined bars, provided to resist radial shear at the base of the containment wall, are secondary vertical bars, which are inside the primary vertical bars on the outside face and inside face of the wall. These bars are continuous and are bent across the wall where reinforcing is required to resist the radial shear. The remaining 33-percent of the required steel area is provided by stirrups that are hooked around the vertical bars by means of a 90 degree hook. Only one-third of the shear reinforcing at a particular elevation is made up of these hooked bars, which occur at four elevations up the wall. See Figure 4.16 of the Containment Design Report.

Since the stud anchors are hooked around reinforcing bars, concrete stresses for pull out loads are negligible. For high shear loads, which would be caused if a stud anchor should fail or be missing, local crushing of the concrete occurs; however, integrity of the anchor and liner plate is not impaired. See Figures 5.1-21 and 5.1-22.

The lowest elevation at which these hooked bars are used is at a point where only 65-percent of the maximum shear at the base is present. The remaining three levels are in regions where the shear is less than 25-percent of maximum base shear. Since the large majority of the shear is resisted by continuous vertical bars, a minimal amount of load must be transmitted to the vertical bars. The hooked stirrups will mechanically transmit the small amount of shear, which they carry. The main function of the stirrups is to contain the formation of the diagonal tension crack. The mechanical anchorage of the stirrups is sufficient for this purpose.

There are no significant structural loadings, which must be transferred through the liner such as those required for crane brackets or machinery equipment mounts. Miscellaneous spray system piping, instrumentation, conduit, and insulation, which are attached to the liner can be supported by the free-standing liner without inducing significant stresses in the liner or liner anchorage.

Liner stress is imposed on the cylindrical penetration as a circular uniform load acting around the circumference of the penetration. The liner plate is locally thickened at the penetrations to take care of additional stresses.

The liner can accommodate any shear it will see due to thermal expansion or earthquake.

An investigation was made on the thermal effects, based on the conservative assumptions that the base mat was fully fixed against any thermal movement thereby restraining the liner from movement. The 3-ft fill slab was then subjected to thermal growth. No excessive forces were introduced into the liner and the welds on the test channels were found to be sufficient to prevent any shear failure of the test channels from the liner due to movement of the 3-ft fill mat.

Seismic shear of the interior concrete is resisted by the keying action of the reactor pit and the sump for the recirculation pumps in addition to the weld channels. Considerable resistance is also provided by friction between the liner and the 3-ft slab.

Jet forces cannot remove the liner panels since the forces will be compressing the insulation panels against the liner and exterior wall. The panels are anchored to the liner with 3/16-in. diameter stainless steel studs. The consequence of an insulation panel being displaced from the liner during or as a consequence of an accident is that the exposed liner would tend to expand. The unequal strain between the exposed and unexposed portions of the liner causes a shear load on the liner anchor, and a local yielding in compression of the exposed portion of the liner. The liner anchor stud has the capacity to accommodate much greater strains than would be experienced at yield strain in the liner.

5.1.3.7 Containment Interior Structure

The interior structure may be separated into five main structural components. They are:

1. 3-ft thick fill slab.
2. 3-ft thick crane wall.
3. 4-ft to 6-ft thick refueling canal.
4. 2-ft thick operating floor slab.
5. Primary shield wall.

The method of design, stress analysis, critical stresses and locations are as follows:

1. 3-ft thick fill slab - The controlling loads on the 3-ft slab are the reactions are from the primary equipment supports due to various postulated pipe breaks. The slab was designed as a series of radial beams running under the equipment supports and spanning between the reactor support wall and the crane wall. Stresses in reinforcing were limited to 0.9 fy. Maximum stresses occur immediately below the primary equipment supports.

2. <u>3-ft thick crane wall</u> - The crane wall was designed for a 7 psi differential pressure occurring immediately after a primary pipe break and prior to pressure equalization.

   Although the stress levels associated with this pressure differential were sufficiently low to establish that the concrete could resist the pressure loading, sufficient reinforcing was provided to resist all membrane forces without any contribution from the concrete. Stresses were limited to 0.9 fy. The membrane hoop stress was 33 ksi and the axial vertical rebar stress was 14.3 ksi.

A two dimensional finite element analysis was performed to determine the effect of the jet forces associated with the pipe break on the crane wall.

The jet force associated with a pipe break has been based on the static force PA where P is the primary system operating pressure and A is the cross sectional area of the coolant pipe. The analysis indicated that in local areas (at the application of the force) yielding of the crane wall rebar will occur. The load was assumed to act at the mid-height of the wall, thus causing maximum bending moment. The ability of the wall to support the dead load of the crane was checked, considering the yielded area indicated by the computer analysis as unable to carry load. A beam 12-ft long and 5-ft deep (the underside of the operating floor to the top of the potential yield portion of the crane wall) was found to provide more than twice the ultimate capacity required. This analysis was very conservative for three reasons:

1. A jet force load at this location would cause little yielding since it is not located at mid span.
2. The haunch at the underside of the operating floor was not considered.
3. The membrane effect of the circular crane wall was not taken into account.

Further stability of the crane wall was demonstrated by determining the ultimate failure load by means of a yield line analysis. This analysis indicated that the structure has the capacity, through strain energy of structural response, to resist the uniform jet force load of 1500 kips or 975 kips with the 7 psi pressure differential without failure.

The containment internal concrete is essentially rigid; (fundamental frequency 18.6 cps) therefore, seismic loads were calculated using the maximum ground acceleration (0.15g).

The crane wall was initially considered as a cantilever beam with a frequency of approximately 13 cps and the base shear was determined by the response spectrum approach. The base shear was distributed to the individual nodes by the formula:

$$Fx = \frac{W_x h_x V}{\Sigma W h}$$

Where

V = base shear
W_x = weight of node under consideration
h_x = distance from base to section under consideration.
3 Wh = Summation of the product of weights and heights of all nodes

The moment at the base was determined and the uplift calculated by considering a circular ring of thickness equal to the area of steel per in. This maximum uplift, which occurs at one point at the base of the structure stresses the rebar to 5.2 ksi.

The crane wall was also designed to resist steam and feed water pipe break reactions of 340 kips and 200 kips where supports are connected to the wall. The extra steel provided for pipe break loads is available in the form of steel buttresses to resist pressure, jet force, and seismic loads; however, it was not considered in the analysis.

3. <u>4-ft to 6-ft thick refueling canal</u> - The refueling canal was designed for the 7 psi pressure differential. The wall resists the pressure by spanning vertically between the refueling floor and the operating floor. Stresses were limited to 0.9 fy.

An analysis was performed to check the effects of the jet force load the cross section was found to be sufficient to provide stability. A yield line analysis was performed and provided the basis for the above.

The seismic load was determined by the same procedure used for the crane wall. The average load in kips/ft was distributed over the wall and the vertical span was conservatively assumed to carry the entire load. The resulting bending moment was found to be well within the capacity of the wall.

4. <u>2-ft thick operating floor slab</u> - Because of the many openings in the floor for equipment, the floor was designed as a series of beams. Principal loadings were D.L. + 500 psf live load and 7 psi upward pressure differential + D.L. The first loading (D.L. + 500 psf live load) was designed in accordance with Part IV-B of ACI 318. Stresses for the pressure differential case were limited to 0.9 fy.

   The operating floor was investigated. There appears to be very little area of the operating floor, which could be reached by the expanding jet of water from a break in the reactor coolant system. The jet will be greatly dispersed in the distance between the primary coolant piping and the underside of the operating floor. The only area of the floor, which could be struck by a jet spans between areas of the floor heavily reinforced as beams. The span cross section consists of a T-beam with the 2-ft thick floor acting as the flange and the 7-ft high biological shielding wall as the web. This section can resist the jet force load within 0.9 fy stress limit on the rebar.

5. <u>Primary Shield Wall</u> - This was designed for two loading conditions due to a split in the reactor. The stress in the reinforcing was limited to the tensile strength of the bars. The first load considered was a 1-ft wide longitudinal split along the length of the reactor. The vessel is assumed accelerated through a 6-in. distance against the support wall by the jet force caused by a 2200 psi pressure acting through a 26.4-ft long by 1-ft wide longitudinal vessel rupture, which results in an impact load of 650 k/ft. This load is imposed by considering an impact factor of two. The maximum rebar stress is 69.5 ksi. The second load considered a pressure buildup of 1000 psi inside the pit due to release of reactor contents. This produces a rebar stress of 86 ksi. The rebar used is ASTM A 432 with specified yield of 60 ksi and ultimate tensile strength of 90 ksi.

To protect the containment base liner, an average of 2-ft of concrete above the containment liner plus a 1-in. liner plate embedded on top of the concrete was provided at the bottom of the containment reactor cavity pit. Below the containment liner plate is 4.5-ft of structural concrete poured on rock.

Temperature differential conditions as a result of a LOCA are considered to be of such short duration that the effects were not used in the design of interior structures for stress analysis. A sketch of the design conditions is given in Figure 5.1-24.

During normal operations, the only significant transient temperature gradients occur during startup. The minimum containment internal temperature is limited to 50°F. The maximum operating containment internal temperature is 130°F. Forced movement of containment air is used to limit the concrete temperature surrounding the reactor vessel. This forced air movement of the containment air as well as normal convection and radiation is expected to limit the concrete temperature differentials in the range of 5°F to 10°F. To demonstrate the large margin available in the concrete crane wall and the primary shield wall, a conservative assumption of a 30°F temperature gradient has been evaluated. The evaluation included the gradient effect through the crane wall, the 6-ft thick portion of the primary shield wall below the reactor coolant pipe nozzle, the 5-ft thick portion of the primary shield wall where the nozzles penetrate the wall, and the 4-ft thick wall above the shield wall.

The maximum rebar stress was found to be 4500 psi and occurs in the vertical rebar in the crane wall. The maximum compressive concrete stress was found to be 226 psi and occurs in the hoop direction in the 5-ft portion of the primary shield wall. These stresses are approximately 20-percent of the allowable working stress values and will have no significant effect on the design adequacy of the structures analyzed.

5.1.3.8 Pressure Stresses

5.1.3.8.1 Accident Pressure

Pressure effects on the containment structure may be divided into two types:
(1) membrane stresses and (2) discontinuity stresses.

1. For membrane stress analysis, the dome and cylinder are treated as thin-walled shell structures. (The thickness to radius ratio for the dome is 1/20 and the cylinder 1/15. These ratios are smaller than the 1/10 criterion for thin-walled shell analysis.[8] Membrane forces are resisted by steel reinforcing.

2. Discontinuity stresses occur at the juncture of the cylinder and the mat and the juncture of the cylinder and dome. Discontinuity effects are determined as follows:

    a. The radial growth of the shell is computed based on membrane stress in the reinforcing and liner.

b. The flexural rigidity of the meridional wall section is determined based on a cracked section analysis in accordance with conventional reinforced concrete design techniques.

c. Moments and shears are calculated based on having consistent deformation for the two elements at the point of discontinuity.

Discontinuity effects at the spring line are very slight due to the small difference in radial growth between the dome and cylinder. Since the circumferential reinforcing in the dome and cylinder vary, stresses and therefore deformations are essentially equal.

The mat is considered as offering complete fixity; no credit is taken for the liner at the base in resisting moments since at the point of maximum shear the bond between the liner and concrete is insufficient to transmit complementary beam shear. A slip surface between the concrete and liner is formed and the liner is subjected to membrane forces only.

The 9-ft thick mat is subjected to the following due to pressure inside the containment building:

1. Uplift at the juncture with the wall.
2. Moment and shear due to discontinuity effects with the wall.
3. Downward pressure loading due to internal pressure.

The 9-ft mat is designed to accommodate the flexural effects of these loads. At the crane wall, the mat is founded on the unyielding rock and further pressure loads are transmitted through bearing directly into the rock.

Resistance to these loads is based on a cracked concrete section. No credit is taken for the liner for the same reasons given for the wall.

Discontinuity shears in both the cylinder and mat are resisted by either bent bars or stirrups.

In the outer portions of the base mat, the slab is raised off of the rigid foundation under accident loadings; thus no frictional resistance can be offered by the rigid foundation. Where the uplift is overcome, the only load of any consequence, which must be resisted by the mat is the radial tension. The restraint, which is imposed by the rigid foundation on the bottom portion of the base mat, effectively eliminates all radial tension in the mat. However, for conservatism this restraint has been neglected in the analysis of the mat for radial tension. The hoop and radial reinforcing supplied as temperature reinforcing is more than adequate for this purpose.

5.1.3.8.2 Soil Pressure

Portions of the containment structure are subjected to the effects of backfill bearing against the containment wall. The effects on the structure are:

1. Shear and overturning effects due to seismic response and interaction between the soil and structure.

2. Discontinuity effects caused by the soil restraining deformation of the structure under accident pressures.

To determine the shear and overturning effects two limiting cases were investigated. The first was the case where the structure and soil move out of phase. It was assumed that the structure was subjected to the passive pressure of the soil with the mass of soil, within the shear failure envelope, accelerated against the structure with ground acceleration. In the second case the soil and structure move in phase. For this case it was assumed that the structure was subjected to the active pressure of the soil with the mass of soil, within the shear failure envelope, accelerated with the structure at ground acceleration.

These loads were then treated as external loads on the structure. See Section 3.1.5 of the Containment Design Report for additional information.

To determine the discontinuity effects caused by soil restraint, the structure was analyzed for the passive pressure case. The restraint of the deformation of the structure due to the soil was calculated. Vertical and circumferential bending moments due to this restraint were then determined. Reinforcing bar stresses were calculated and found to be minor. This analysis was then verified by a finite element analysis.

In this analysis, full contribution of the backfill was assumed. During the course of construction it became necessary to build a retaining wall in a substantial area of the backfill, to facilitate construction. The retaining wall extends over 50-ft in plan and includes all of the high fill points assumed in the analysis and design. It can therefore be concluded that the analysis was conservative in that the backfill effects on the completed structure would be only a fraction of that assumed in the original design.

5.1.3.9 Thermal Stresses

Temperature effects on the containment structure may be divided into two separate considerations: one effect is due to a thermal gradient through the wall, the other is caused by the rapid temperature rise of the liner under accident conditions. The reinforced-concrete wall restrains the liner from growing, resulting in compression in the liner and additional tension in the reinforcing.

1. Calculation of gradient stresses is based on method of analysis outlined in ACI 505-54, "Specification for the Design and Construction of Reinforced Concrete Chimneys."[9] The gradient used is linear with 120°F on the inside and 0°F exterior concrete temperature (-5°F ambient). The maximum operating temperature of the containment is 130°F. The effect of elevated operating temperature (up to 150°F) on the structural elements was evaluated in 1987 and was found to be acceptable.

   The ACI method assumes a cracked section in which the concrete carries no tension. The neutral surface (surface at which no thermal stress exists) is determined. Stresses in the liner and reinforcing are calculated based on the assumption that there is no distortion of the wall; i.e., variation of strain through the wall thickness is linear.

2. To determine the effects due to rapid rise in liner temperature, there are two basic assumptions made. The first is that the effects are internal in nature; i.e., the compressive force in the liner is balanced by a tensile force in the reinforcing. The second is that there is no distortion of the wall.

Because temperature effects are internal in nature and do not affect the overall tensile load carrying capability of the structure, local yielding of reinforcing under accident conditions is acceptable.

The temperature gradient through the wall is essentially linear on both the insulated and uninsulated portions and is a function of the operating temperature internally and the average ambient temperature externally. Accident temperatures mainly affect the liner, rather than the concrete and reinforcing bars, due to the insulating properties of the concrete. By the time the temperature of the concrete adjacent to the liner begins to rise significantly, the internal pressure and temperature in the containment shell due to maximum thermal gradient will not influence the capacity of the structure to resist the other forces. Temperature effects induce stresses in the structure, which are internal in nature; tension outside and compression in the inside of the shell such that the resultant force is zero. Loading combinations concurrent with these temperature effects may cause local stresses in the outside horizontal and vertical bars to reach yield; however, as local yielding is reached, any further load is transferred to the unyielded elements. At the full yield condition, the magnitude of final load resisted across a horizontal and vertical section remains identical to that which would be carried if the temperature effects were not considered. Thus, the overall carrying capacity of the structure and the factor of safety of the structural elements are not affected.

5.1.3.10 Analysis of Openings

The methods followed in design of large openings are described in Section 3.4 of the Containment Design Report (CDR). Included are descriptions of the safety factors used in design. Sample calculations are provided, listing all the criteria and analyzing the effects of all pertinent factors, such as cracking. Also addressed in the CDR is how the existence of biaxial tension in concrete (cracking) has been taken care of in the design, and how the normal and shear stresses due to axial load, two-directional bending, two-directional shear, and torsion are combined. Additionally, the criteria for the design of the thickened part of the wall around the openings is stated.

The methods used to check the design of the thickened stiff part of the shell around large openings and its effect on the shell, torsional stresses, and shrinkage considerations are also addressed in Section 3.4 of the Containment Design Report. This section also describes how deformations and forces are handled around the large openings and in the transition zones into the main portion of the structure.

In the cylindrical section of the containment, where there are large openings for access hatchways and penetrations, the reinforcing bars (hoop, vertical and diagonal) are continued without interruption around the openings.

No bar terminates at any openings as illustrated around the penetration in Figure 5.1-1. Also additional bars have been furnished locally to take the stresses developed around large openings. Concrete is locally thickened at the equipment access hatchway area to accommodate all the reinforcing bars required in this area.

A finite element analysis is performed on the large openings. Representation of the structure is by rectangular elements; each element consists of ten layers of orthotropic, elastic material to

represent the reinforcement, concrete and the liner. About 1000 degrees of freedom are considered in the model. This analysis is used as a check on the adequacy of the large openings. Results appear in the Containment Design Report.

A finite element analysis of the equipment hatch area indicated local liner plastic deformations during the pressure test. For the order of magnitude and location of these stresses, see Section 3.4 of the Containment Design Report. These deformations have no influence on the structure during the pressure test due to the ductility of the studs and liner plate.

The limiting elastic liner deformations during test pressure will be from tensile stresses. During an accident loading they will be from compressive stresses. Therefore, a relationship between the pressure and accident loads cannot be determined directly. However, the test pressure demonstrates the ductile behavior of the liner.

Since the containment is not subject to accident temperatures during the testing, no direct correlation between test and accident conditions can be made in evaluating thermal stresses at large openings.

The liner is stressed beyond the yield point in very local areas adjacent to the transition from the thickened equipment hatch boss to the cylinder wall. The maximum stress is equal to 39.28 ksi for the 1.5P loading condition. The strain corresponding to this stress (0.17-percent) is below the limits (0.5-percent) stated in Section 2.2.4 of the Containment Design Report. The average liner stress in the cylinder for the 1.5P load combination is approximately -15 ksi in the vertical direction and -2.0 ksi in the horizontal direction.

The maximum rebar stress associated with the 1.5P load combination is approximately 66 ksi in the 4'-6" portion of the containment wall cylinder.

For a complete discussion of liner stresses, see the Containment Liner Stress Analysis Report. For a detailed discussion of liner stresses in the equipment hatch area and further justification of the stresses noted above, see Section 3.4.4 of the Containment Design Report.

All reinforcing is continuous around penetrations. Steps have been taken to ensure that no local crushing of concrete will occur. From Reference 16, it has been determined that in order to prevent local crushing of the concrete, a minimum bend diameter of 31 times the bar diameter is required when the reinforcing is stressed to yield. The angle of bend in the rebar determines the force that will be transmitted to the concrete in the event the bar tries to straighten out due to tension. For this reason most bars are bent at 10 degrees except at large penetrations including the equipment hatch, personnel lock, main steam and feedwater, and air purge penetrations, where the deviation of the bar from its centerline is too large to permit a 10 degree bend. In these cases the bars have been bent at 30 degrees but a tie-back system is used, which prevents a buildup of forces. To prevent this buildup, (in all cases except the equipment hatch penetration), the line of force makes an angle of one-half of the angle of bend, from a horizontal line from the vertical bars and from a vertical line for the horizontal bars and is tangent to the outside of the penetration.

At the personnel and equipment hatches a large void will be carried since, due to the large offset of the bars from their centerline, it will take the bars longer to return to their centerline after passing the penetration. To prevent any cracking and spalling of concrete and to add lost strength to the cross-section, these voids have been filled with added rebar, which achieves bond by means of mechanical anchorage.

The same precautions mentioned above have been taken with the seismic bars. See Figure 5.1-25.

For penetrations between 9-in. and 18-in. in diameter, all the reinforcing bars including primary and secondary vertical bars and diagonal bars have been grouped around the penetrations. Due to the continuity of the bars and the relatively small opening size, no special provisions need be made to resist normal, shear, and bending stresses. The penetrations are keyed into the concrete, thus creating an edge loading, which will put torsion into the wall. The loads are small and the rebar will feel little effects from this torsional loading.

For penetrations greater than 18-in. up to 48-in. in diameter, the bars are continuous. Due to the large angle of bend of these bars, a tie-back system is used, which offers additional resisting strength to shear, bending, and torsional stresses.

5.1.3.11 Seismic and Wind Design

The design of the containment, which is a Class I structure (see Section 1.11), is based on a "response spectrum" approach in the analysis of the dynamic loads imparted by earthquake. The seismic design takes into account the acceleration response spectrum curves as developed by G. Housner. Seismic accelerations have been computed as outlined in TID-7024[10] and Portland Cement Association Publication.[11]

The following damping factors have been used:

	Component		Percent Critical Damping
1.	Containment structure		2.0
2.	Concrete support structure of reactor vessel		2.0
3.	Steel assemblies:		
	a.	Bolted or riveted	2.5
	b.	Welded	1.0
4.	Vital piping systems		0.5
5.	Concrete structures above ground:		
	a.	Shear wall	5.0
	b.	Rigid frame	5.0

As indicated in Section 5.1.2.2, ground accelerations used for design purposes are 0.1g applied horizontally and 0.05g applied vertically. The natural period of vibration is computed by the Rayleigh method; in this method, the containment structure is analyzed as a simple cantilever intimately associated with the rock base and with broad base sections of adequate strength to assure full and continued elastic response during seismic motions. Further, both bending and shear deformations are considered.

The structure is divided into sections of equal length and loaded laterally by dead weight of the section and any equipment and live load occurring at the section. Deflections caused by shear and moments are then determined, and the end deflection is given the value $\phi' = 1.0$ with corresponding values determined for other sections. The natural period of vibration for the structure is then determined by setting potential energy equal to kinetic energy and solving for the period.

$$T = 2\Pi \left[\frac{Y_0 \sum \varnothing^2 \, dm}{g \sum \varnothing \, dm} \right]^{1/2}$$

where

Y_0 = maximum actual deflection

$$\phi = \frac{\text{deflection of section under consideration}}{\text{maximum actual deflection}}$$

g = acceleration due to gravity

dm = weight of section under consideration

T = period in sec.

Based on an uncracked concrete section, the period is determined to be 0.241 sec. A more realistic calculation for a cracked section, using reinforcing steel and liner as the resisting elements, yields a period 0.936 sec.

Using the derived period and entering the acceleration spectral curves, Figures 1.11-1 and 1.11-2 of Section 1.11, and applying a 2-percent critical damping, a spectral acceleration for the containment was selected. This value was derived to determine the base shear. The distribution of base shear is a triangular loading assumption.

This assumption yields a load distribution pattern with zero loading at the base to a maximum loading at the spring line of the dome. Above this line, the loading decreases due to a change in section and consequently change in weight. This load distribution allows the determination of shears and moments at any critical section through the containment from which the appropriate unit stresses are obtained.

Seismic shears are resisted by diagonal reinforcing except in the upper areas of the dome. No credit is taken for the reinforcing in compression.

From 30 degrees above the springline, where the seismic shears are small, the shears are carried by dome reinforcing steel lying in the plane of principal tension

A finite element analysis was performed on the basemat using loads determined for the three basic loading conditions specified in the Containment Design Report. Maximum hoop moment caused by lack of symmetry of the seismic loading was found to be 454 in.–kips/in. This compares with a capacity of 690-in.-kips/in. for the in-place hoop reinforcing.

Tornado loads have not been considered in the design of the Indian Point Unit 2 Containment Building; however, similarity in design of Indian Point Unit 3 (where such loads are considered) indicates that seismic reinforcement bars provide a more than adequate mechanism to withstand the torsional effect of Tornado loads.

The torsional effect results from wind striking the containment building at an angle α from the normal, as shown in Figure 5.1-26. The torsional force is due to the component of the wind tangential to the surface of the containment building and is equal to:

$$F_t = AC_D\,(q)\,(\sin\alpha)$$

Where

A = surface area of the containment

C_D = 0.5 from A.S.C.E. Paper 3269 - "Transactions of the A.S.C.E.," Vol. 126 Part II 1961, p. 1165 (coefficient of drag)

$q = 0.002558\ V^2$ (wind pressure)

α = 45 degrees

This assumption is conservative in that the actual tangential force would be the result of skin friction and the effects would be negligible.

This component of torsional force is computed from a direct wind loading as based on A.S.C.E. Paper 3269.

Torsional shear is a maximum at the juncture of the walls and base slab and varies to zero at the top of the dome.

The torsional effect can be converted to a shear per lineal foot around the circumference of the containment by distributing the shear over the circumference of the seismic reinforcing.

The seismic bars provide a more than adequate mechanism to withstand this torsional effect. The maximum stress in the bars under this loading is 17 ksi. See Figure 5.1-26.

5.1.3.12 Cathodic Protection

During the initial Licensing process, a complete survey and tests to determine the need for cathodic protection on Indian Point Unit 2 was made by the A. V. Smith Engineering Company of Narberth, Pennsylvania. Electrical resistivity measurements and a visual inspection of the area away from the river, where the turbine generator building, reactor building, primary auxiliary building and associated facilities are located indicated that the environment is mostly rock with areas of dry sandy clay. The electrical resistivity of the soil ranged from 3,500 to 30,000 ohm-cm with the majority of the readings being above 10,000 ohm-cm. On this basis, it was determined that cathodic protection was not required on underground facilities in areas away from the river or the containment building liner, although a protective coating on pipes was recommended to eliminate any random localized corrosion attack. An analysis of Hudson River water data, obtained from the Con Edison plant chemist, showed the electrical resistivity of the water to vary over an extremely wide range due to salt intrusion from the ocean. The range of resistivity has been from 59 to 10,000 ohm-cm with a large number reading in the 300 ohm–cm area. This value was considered to be extremely corrosive and the following structures in the area near the river were placed under cathodic protection:

1. Circulating water lines.
2. Service water lines.
3. Bearing piles.
4. Sheet piling (earth and water side) and wing wall anchorage system.
5. Metallic structures inside intake structure (traveling screens, bar racks, circulating water pump suction, service water pump suction).

In 2008, a cathodic protection field survey and assessment of underground structures at Indian Point Unit 2 was performed by PCA Engineering of Pompton Lakes, New Jersey. A positive shift in pipe potential was found where the City Water supply piping from the City Water Tank crosses the Algonquin Gas pipes. As a result the City Water supply piping in the vicinity of the gas pipes was placed under cathodic protection.

The cathodic protection system for the Circulating Water lines and the Service Water lines were found not to be functional and the rectifiers were removed. In order to assure the lines will perform their functions, the buried pipes are inspected as part of the Buried Piping Program. Inspections of buried piping are initially performed using Guided Wave (GW) ultrasonic inspection techniques to locate potential areas of degradation. If significant degradation is detected during the GW inspections, excavation is performed to uncover the affected sections of piping and a direct visual inspection and UT thickness measurements are performed. Repairs and / or replacements are implemented as required to restore degraded piping sections to within the required structural margins of safety.

In addition to the inspections performed as part of the Buried Piping Program, the nuclear safety related portion of the service water piping is further subjected to pressure and / or flow testing as required by ASME XI, Subsection IWA-5244. Visual inspections on the inside surface of the SW piping are also performed under the GL 89-13, Service Water program. Based on the results of the inspections and testing, the Service Water system is structurally adequate to perform its required safety function.

The cathodic protection system for the Traveling Water Screens and Bar Racks were found not to be effective and the installed cathodic protective systems were retired. The original Traveling Water Screens which were carbon steel were upgraded to stainless steel frames, baskets, and chains. The splash housings are also stainless steel. The Bar Racks, replaced in the mid-1990’s are of carbon steel construction and are epoxy coated with a tar epoxy expected to provide corrosion protection over a 40-year period. The guides for the Screens and Racks are carbon steel channels mounted in a concrete through. The rate of corrosion is slow and the Screens and Racks are on a regular PM cycle that checks for degraded conditions.

The Service Water and Circulating Water pumps suction are not cathodically protected. Rather, the Service Water Pumps suction is inspected and refurbished as part of the Service Water Pumps Preventing Maintenance (PM) activities. The Circulating Water Pumps are inspected and refurbished according to Preventive Maintenance program requirements.

5.1.3.13 Containment - Shear Crack

The arrangement of reinforcing bars in the containment shell is such that a reinforcing bar crosses any potential crack plane. Any cracks resulting from diagonal tension caused by shearing forces will be carried by reinforcing bars, which span across the crack. Thus all shears will be carried by the reinforcing bars and none by the concrete.

The reinforcing bars are almost all continuous throughout the containment structure; however, where a bar terminates this is accomplished by means of a 180 degree hooked bar. In no case are bars simply terminated without providing means for additional anchorage.

Throughout the cylinder, the meridional reinforcing is continuous. Beyond the springline, the bars extend radially toward the center of dome. As the bars reach a 6-in. spacing, which is one-half the required spacing, alternate bars have been dropped off by means of reinforcing splice plates. The splice piece consists of a plate with two Cadweld sleeves welded on the incoming side and one sleeve welded on the outgoing side. Thus, the number of bars present is halved and the spacing is increased to the required 12-in.

This is repeated to the top of the dome where a three layered grid pattern has been used to maintain the continuity of the rebars. The bars in the grid pattern have been Cadwelded to the same type reinforcing splice plates described above, but the Cadweld is beveled to obtain the desired direction of the grid.

At the base in the area of high discontinuity stresses, additional No. 18S bars have been provided. At the point where they were no longer needed, they have been Cadwelded to a No. 11 bar, which is terminated with a 180 degree hook.

All seismic bars have been terminated in a 180 degree hook. In no case was a No. 18S bar terminated in this way since the minimum 180 degree hook could not be provided in a 4-ft 6-in. thick wall.

Radial shear reinforcing stirrups were terminated by hooking around vertical bars.

5.1.4 Containment Penetrations

5.1.4.1 General

In general, a penetration consists of a sleeve embedded in the concrete and welded to the containment liner. The weld to the liner is shrouded by a continuously pressurized channel, which is used to demonstrate the integrity of the penetration-to-liner weld joint. The pipe, electrical conductor cartridge, duct or equipment access hatch passes through the embedded sleeve and the ends of the resulting annulus are closed off, either by welded end plates, bolted flanges or a combination of these. (See Figures 5.1-27 through 5.1-31.)

Differential expansion between a sleeve and one or more hot pipes passing through it is accommodated by using a bellows type expansion joint between the outer end of the sleeve and the outer end plate, as shown on Figure 5.1-30. Pressurizing connections are provided to continuously demonstrate the integrity of the penetration assemblies.

5.1.4.2 Types of Penetrations

5.1.4.2.1 Electrical Penetrations

The electrical penetration system consists of 60 electrical penetrations including the following: 48 Crouse-Hinds, 1 Westinghouse, 10 Conax and 1 spare sleeve (below flood-up level.)

The Crouse-Hinds and Westinghouse types are identical in design (see Fig 5.1-27). This is because Westinghouse took over the Crouse-Hinds manufacturing facility and design after the

original plant penetrations were purchased. The design of this type of electrical penetration utilizes a single canister that is sealed at both ends by a combination of metal and ceramic seals. Epoxy layers on both ends provide a physical support for the conductors within the penetration canister. All of the Westinghouse and Crouse-Hinds penetrations are welded to the sleeve inside containment. The entire canister assembly is constantly pressurized by the weld channel pressurization system and monitored for any leakage.

The Conax penetrations are of a modular design consisting of a stainless steel header and 18 independently mounted conductor feedthrough modules (Figure 5.1-28 and 5.1-29), which can be individually removed and relocated. The header plate and the individual feedthrough modules are the pressure-retaining boundary. This type penetration does not have a sealed canister. The conductor modules are threaded into the header plate and the header plate itself is welded to the sleeve, which goes though the containment wall. Leakage monitoring of the Conax penetrations is accomplished by interconnecting ports machined in the header plate to each conductor feedthrough module. A small hole is provided on each conductor feedthrough module stainless steel tubular housing allowing the feedthrough module to be pressurized when the header plate's parts are pressurized. Metal compression fittings (swaging type) are used for mounting the conductor feedthrough modules to the header in a double seal manner. The individual conductors passing through the feedthrough module are surrounded by polysulfone and are sealed (swaged) at each end of the feedthrough housing. The length of the housing (feedthrough tube) is roughly 2-ft longer than the sleeve within, which the penetration is installed. Six of the Conax penetrations are welded to their sleeve outside containment and four are welded to their sleeve inside containment to accommodate differences in the sleeves into which they are welded.

Weld channel rings are used to create a double weld seal between the header plate and the containment sleeve. All of the weld joints necessary maintain containment integrity are monitored for leaks with the weld channel pressurization system.

If a minor leak should develop at any of the plant’s electrical penetrations, a release from inside containment to outside should not occur since each penetration is double sealed and pressurized to maintain a positive pressure (between 49 and 55 psi), which is higher than anticipated containment accident pressures.

5.1.4.2.2 Piping Penetrations

Double barrier piping penetrations are provided for all piping passing through the containment. The pipe is centered in the embedded sleeve, which is welded to the liner. End plates are welded to the pipe at both ends of the sleeve. Several pipes may pass through the same embedded sleeve to minimize the number of penetrations required. In this case, each pipe is welded to both end plates. A connection to the penetration sleeve is provided to allow continuous pressurization of the compartment formed between the piping and the embedded sleeve. . These penetrations are listed as “Hot” in Table 5.2-1. In the case of piping carrying hot fluid, the pipe is insulated and cooling is provided to reduce the concrete temperature adjoining the embedded sleeve. Local areas are allowed to have increased temperatures not to exceed 250ºF. Cooling is provided for hot penetrations through the use of air-to-air heat exchangers. These are made in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII, by welding together one flat sheet and one embossed sheet of 10 gauge carbon steel material, the embossment forming coolant passages. The unit is rolled into the form of a cylinder with an outside diameter slightly smaller than the respective inside diameter of the penetration sleeve. The exchanger is placed inside the sleeve and outside the pipe insulation,

with the inlet and outlet coolant connections penetrating the sleeve between the outside concrete wall surface and the bellows expansion joint.

The coolant to be used is ambient air fed by a rotary blower, which is backed up with a full sized spare. The isolation features and criteria for piping penetrations are given in Chapter 6. Figure 5.1-30 shows typical hot and cold pipe penetrations.

A total of 107 pipes pass through 53 penetration sleeves, 23 of which are considered thermally hot. In addition, two spare penetration sleeves (capped and pressurized) are available for the possible future addition of piping.

All piping penetrations are designed for normal loads within the stress limits of the ASME Code, Section VIII.

All piping penetrations except main steam and feedwater are designed as anchors for the pipes passing through them and will transmit piping loads to the reinforced concrete wall. The anchorage strength exceeds the maximum combined forces imposed by the effects on the piping penetration of dead load, loads induced from a loss of coolant accident, thermal expansion of the pipe, penetration air pressure, and earthquake loads. The piping penetrations are designed to transmit the above combined loadings to the concrete structure without exceeding the yield strength of penetration steel.

In addition, each piping penetration is designed to withstand, within emergency load criteria, the effect of the rupture of a pipe passing through that penetration at or near the penetration.

The main steam and feedwater penetrations are designed so that the pipes themselves are effectively enclosed for blowdown just inside and just outside the wall. These anchors are designed to prevent a main steam or feedwater pipe rupture from causing a breach of containment at the penetrations. The anchors are designed to 90-percent of yield strength.

All piping penetrating the containment is designed to meet the requirements of USAS B31.1 (1955) Power Piping Code.

Pipes that penetrate the containment building wall and that are subject to machinery-originated vibratory loadings, such as from the reactor coolant pumps, have their supports spaced in such a manner that the natural frequency of the piping system immediately adjacent to the penetrations is greater than the dominant frequencies of the pump. Pipeline vibration was checked during preliminary plant operation and where necessary, vibration dampers were fitted. This checking and fitting effectively eliminates vibrating loads as a design consideration.

5.1.4.2.3 Equipment and Personnel Access Hatches

An equipment hatch has been provided. It is fabricated from welded steel and furnished with a double-gasketed flange and a bolted, dished door. The hatch barrel is embedded in the containment wall and welded to the liner. Provision is made to continuously pressurize the space between the double gaskets of the door flanges, and the weld seam channels at the liner joint, hatch flanges and dished door. Pressure is relieved from the double gasket spaces prior to opening the joints. The personnel hatch is a double door, mechanically latched, welded steel assembly. A quick-acting type equalizing valve connects the personnel hatch with the interior of the containment vessel for the purposes of equalizing pressure in the two systems when entering or leaving the containment. Two spring-loaded check valves in series are installed to

allow pressure relief inside the air locks to the containment interior. The 16-ft diameter equipment hatch opening and the 8-ft 6-in. diameter personnel hatch are the only openings, which require special design consideration. The personnel hatch doors are interlocked to prevent both being opened simultaneously and to ensure that one door is completely closed before the opposite door can be opened.

Remote indicating lights and annunciators situated in the control room indicate the door operational status. An emergency lighting and communications system operating from an external emergency supply is provided in the lock interior. Emergency access to either the inner door, from the containment interior, or the outer door, from outside, is possible by the use special door unlatching tools. The design is in accordance with Section VIII of the ASME Code.

[Deleted]

The design basis Fuel Handling Accident (FHA) analysis does not credit accident mitigation via Containment isolation subsequent to a postulated fuel assembly drop. However, Containment closure after a FHA event is an option that can reduce total exposure and is a good ALARA practice. The roll-up door serves as a mechanism that will support rapid closure of Containment in the event a radiation release occurs during fuel handling. Containment closure subsequent to a total loss of Residual Heat Removal (RHR) cooling in MODE 6 is accomplished through installation of the equipment hatch or temporary closure plate. The roll-up door is currently not considered a suitable device that can be credited for Containment closure subsequent to a loss of RHR, pending NRC approval of an outstanding license Amendment request.

5.1.4.2.4 Special Penetrations

1. Fuel Transfer Penetration - A fuel transfer penetration is provided for fuel movement between the refueling transfer canal in the reactor containment and the spent fuel pit. The penetration consists of a 20-in. stainless steel pipe installed inside a 24-in. pipe. The inner pipe acts as the transfer tube and is fitted with a pressurized double-gasketed blind flange in the refueling canal and a standard gate valve in the spent fuel pit. This arrangement prevents leakage through the transfer tube in the event of an accident. The outer pipe is welded to the containment liner and provision is made by use of a special seal ring for pressurizing all welds essential to the integrity of the penetration during plant operation. Bellows expansion joints are provided on the pipes to compensate for any differential movement between the two pipes or other structures. Figure 5.1-31 shows a sketch of the fuel transfer tube.

2. Containment Supply and Exhaust Purge Ducts - The ventilation system purge ducts are each equipped with two quick-acting, tight-sealing valves (one inside and one outside of the containment) to be used for isolation purposes. The valves are manually opened for containment purging, but are automatically closed upon receipt of a safety injection signal or high-containment radiation signal. The space between the valves is pressurized above design pressure while the valves are normally closed during plant operation. See Section 5.3, Containment Ventilation System, and Section 6.4, Containment Air Recirculation Cooling System.

   Seismic Class I debris screens inside the primary containment protect the primary containment isolation valves in the containment purge and pressure relief

exhaust ducts from debris that may inhibit their correct operation. The screens are stainless-steel wire mesh and are mounted over the exhaust ducts.

Two solenoid-controlled, pneumatically operated butterfly valves are provided for each purge penetration, one on each side of the containment building wall. Two penetrations, one supply and one exhaust, are required. Valves are spring-loaded to fail closed.

The space between the valves is pressurized from the pressurization system through an electrically operated three-way solenoid valve. This pressure is maintained only when valves are closed and must be relieved before butterfly valves can be opened. Failure to release this pressure will prevent the inside containment valves from opening. By procedure the outside containment valves are opened after the inside containment valves are open.

Failure of any of the valves to open will prevent the containment building purge supply fan from running. Tripping of the containment building purge supply fan will automatically close the inside containment butterfly valves. By procedure the outside containment butterfly valves must then be closed. When these valves are closed the space between the valves is automatically pressurized. Failure of any valves to close will prevent the adjacent space from being pressurized and will sound the loss of pressurization alarm. Loss of pressure for either zone will be displayed by individual indicating lights at the main control board.

The valve control solenoids for the inside containment isolation valves FCV-1170 and FCV-1172 and pressurization solenoids are controlled from a single control switch on the fan room control panel. The valve control solenoids for outside containment isolation valves FCV-1171 and FCV-1173 are controlled from a switch in the control room. The cycle is initiated by setting the fan room control switch to the "open" position. This will energize the pressurization alarm.

When the pressure between the valves has been relieved, the valve control solenoids for the inside containment isolation valves are energized and these two valves are opened. If for any reason, either of the two inside containment isolation valves fail to open within a given time after the cycle is initiated, both of these valves will close and pressure will be restored. The circuit is interlocked to prevent inadvertent opening of the valves during a safety injection condition.

Once the inside containment purge valves have been opened, the operator has a predetermined time to place the control switch for the outside containment purge valves to the "open" position and once opened to start the purge supply fan. Failure to do so will cause the inside containment purge valves to close.

Position indicating lights for each of the four valves are provided on the fan room control panel and the main control board.

3. Sump Penetrations - The piping penetration in the containment sump area is not of the typical sleeve-to-liner design. In this case, the pipe is welded directly to the base liner. The weld to the liner is shrouded by a test channel, which is used to demonstrate the integrity of the liner.

5.1.4.3 Design of Containment Penetrations

5.1.4.3.1 Criteria

The liner is basically not a load-carrying member. Because it is subjected to strains imposed by the reinforced concrete, the liner has been reinforced at each penetration in accordance with the ASME Code Section VIII. The weldments of liner to penetration sleeve are of sufficient strength to accommodate stress concentrations and adhere strictly to ASME Code Section VIII requirements for both type and strength. The penetration sleeves and plates are designed to accommodate all loads imposed on them under operating conditions (thermal effects and internal penetrations and test pressures) and accident conditions (loads resulting from all strains, internal pressures, and seismic movements). All reinforcing bars except stirrups and facing bars that are not counted on to carry any load are continuous around the openings.

Liner stress is imposed on the cylindrical penetration as a circular uniform load acting around the circumference of the penetration. The liner plate is locally thickened at the penetrations to take care of additional stresses.

5.1.4.3.2 Materials

The materials for penetrations, including the personnel and equipment access hatches, together with the mechanical and electrical penetrations, are carbon steel, conforming with the requirements of the ASME Nuclear Vessels Code and exhibiting ductility and welding characteristics compatible with the main liner material. As required by the Nuclear Vessels Code, the penetration materials meet the necessary Charpy V-notch impact values at a temperature 30°F below lowest service metal temperature, which is 50°F within containment and -5°F outside the containment.

The stainless steel bellows of the hot penetration expansion joints were protected from damage in transit and during construction by sheet metal covers fastened in place at the fabricator's shop.

1. Piping Penetrations: Materials

Piping Penetration Material	Specification
Penetration Sleeve - 12-in. dia. and under	ASTM-A333, Gr. 1
Over 12-in. dia.	ASTM-A201, Gr. B
(see exception below)	normalized to A300 CL. 1, Firebox
- 22-in. dia. containment	ASTM-A53, Gr. B sump suction
- Rolled shapes	ASTM-A36, A131, Gr. C

2. Electrical Penetrations: Materials

   The penetration sleeves to accommodate the electrical penetration assembly cartridges are schedule 80 carbon steel in accordance with ASTM-A333, Gr. 1, except where otherwise noted. The electrical cartridges have been secured to the penetration sleeve so that all possible leak paths between the cartridge and sleeve will be blocked by a pressurized zone.

3. Access Penetrations: Materials

The equipment and personnel access hatch material is as follows:

Item	Material Specification
Equipment hatch insert	ASTM A516, Gr. 60 normalized to ASTM A300, CL. 1, Firebox
Equipment hatch flanges	ASTM A516, Gr. 60 normalized to ASTM A300, CL. 1, Firebox
Equipment hatch head	ASTM A516, Gr. 60 normalized to ASTM A300, CL. 1, Firebox
Personnel hatch	ASTM A516, Gr. 60 normalized to ASTM A300, CL. 1, Firebox

5.1.4.4 Leak Testing of Penetration Assemblies

A preoperational proof test was applied to each penetration by pressurizing the necessary areas to 54 psig. This pressure was maintained for a sufficient time to allow soap bubble and Freon sniff tests of all welds and mating surfaces. Any leaks found were repaired and retested; this procedure was repeated until no leaks existed.

5.1.4.5 Construction

The qualification of welding procedures and welders has been in accordance with Section IX, "Welding Qualifications," of the ASME Boiler and Pressure Vessel Code. The repair of defective welds has been in accordance with Paragraph UW-38 of Section VIII, "Unfired Pressure Vessels."

5.1.4.6 Testability of Penetrations and Weld Seams

All penetrations, the personnel air lock, and the equipment hatches are designed with double seals, which will be normally pressurized at or above the containment design pressure. Individual testing at 115-percent of containment design pressure is also possible.

The containment ventilation purge ducts are equipped with double isolation valves and the space between the valves is permanently piped into the penetration pressurization system. The space can be pressurized to 115-percent of design pressure when the isolation valves are closed. The purge valves fail in the closed position upon loss of power (electric or air).

All welded joints in the liner have steel channels welded over them on the inside of the vessel. During construction, the channel welds were tested by means of pressurizing sections with Freon gas and checking for leaks by means of a Freon sniffer. These welds were also then continuously pressurized at 50 psig.

5.1.4.7 Accessibility Criteria

Limited access to the containment through personnel air locks is possible with the reactor at power or with the primary system at design pressure and temperature at hot shutdown. After shutdown, the containment vessel is purged to reduce the concentration of radioactive gases and airborne particulates. This purge system has been designed to reduce the radioactivity level to doses defined by 10 CFR Part 20 for a 40-hr occupational work week within 2 to 6 hr after plant shutdown. Since negligible fuel defects are expected for this reactor, much less than the 1-percent fuel rod defects used for design and purging of the containment is normally accomplished in less than 2 hr. To ensure removal of particulate matter and radioactive gases, the purge air is passed through a high efficiency and charcoal filters before being released to the atmosphere through the purge vent. The primary reactor shield has been designed so that access to the primary equipment is limited by the activity of the primary system equipment and not the reactor.

5.1.4.8 Penetration Design – Computations

The penetration sleeves and end plates are designed to accommodate all loads imposed on them. The sleeve and end plate loads include the effects of internal pressure; concentrated loads imposed by the sleeve anchors on the concrete as the anchors strain in conjunction with wall movement under both operating and accident conditions; thermal effects due to both gradient and thermal reactions of the particular item passing through the sleeve; shear, bending, and compression due to accident end pressures; and shear and bending due to seismic movements of the particular item passing through the penetration. The sleeve and expansion joint are designed to remain within ASME Code Section VIII stress limitations with small strains under all or any combinations of loadings mentioned above.

For design computations of penetrations and the shell adjacent to them, see Figures 5.1-32 and 5.1-33. In Section 5.1.4.8.1, the formula for radial deformation of a hole in a plate subjected to biaxial stresses is determined by performing an integration of the tangential strains around the periphery of the hole.

In Section 5.1.4.8.2, the relationship between the deflection determined from above to the final plate and penetration sleeve deformations is developed and the formulas for stress in the liner and the stress in the penetration sleeve are developed.

Section 5.1.4.8.3 shows a summary of the liner and penetration stresses and states the assumptions made in the analysis.

In addition, thermal loads have been investigated for their effect on the shell adjacent to the penetration sleeve and found to be insignificant (38 psi bearing stress on the concrete is the maximum stress on the concrete shell).

5.1.4.8.1 Radial Deformation of a Hole in a Plate

From Reference 17, page 81

$$\sigma o = S - 2 S \cos 2\theta + [S' - 2S' \cos(2\theta - \pi)]$$

where

S = Horizontal stress in liner
S' = Vertical stress in liner

$$\delta D = \frac{1}{E}\int_{o}^{\pi} \left(S - 2S\cos 2\theta + \left[S' - 2S'\cos\left(2\theta - \pi\right)\right]\right) r\sin\theta d\theta$$

$$\delta D = \frac{r}{E}\left[\int_{o}^{\pi} S\sin\theta d\theta - 2S\int_{o}^{\pi}\cos 2\theta\sin\theta d\theta + S'\int_{o}^{\pi}\sin\theta d\theta - 2S'\int_{o}^{\pi}\cos\left(2\theta - \pi\right)\sin\theta d\theta\right]$$

$$\int \cos\left(2\theta - \pi\right)\sin\theta d\theta = -\int \cos 2\theta\sin\theta d\theta$$

$$= -\int\left(1 - 2\sin^2\theta\right)\left(\sin\theta\right)d\theta$$

$$= -\int\left(\sin\theta - 2\sin^3\theta\right)d\theta$$

$$= -\left[\left(-\cos\theta\right) - 2\frac{\sin^2\theta\cos\theta}{3} + \frac{2}{3}\int\sin\theta d\theta)\right]$$

$$= -\left(-\cos\theta + 2/3\sin^2\theta\cos\theta + \frac{4}{3}\cos\theta\right)$$

therefore

$$\int \cos(2\theta - \pi)\sin\theta d\theta = \frac{-\cos\theta}{3} - \frac{2}{3}\sin^2\theta\cos\theta$$

$$\delta = \frac{r}{E}\left[-S\cos\theta - 2S\left(\frac{\cos\theta}{3} + \frac{2}{3}\sin^2\theta\cos\theta\right) - S'\cos\theta - 2S'\left(\frac{-\cos\theta}{3} - \frac{2}{3}\sin^2\theta\cos\theta\right)\right]_{o}^{\pi}$$

$$\delta = \frac{r}{E}\left[\left(S + \frac{2}{3}S + S' - \frac{2}{3}S'\right) - \left(-S - \frac{2}{3}S - S' + \frac{2}{3}S'\right)\right]$$

$$\delta = \frac{r}{E}\left[2S + \frac{4}{3}S + 2S' - \frac{4}{3}S'\right]$$

$$\delta = \frac{r}{E}\left[\frac{10}{3}S + \frac{2}{3}S'\right]$$

$$\delta = \frac{2}{3}\frac{r}{E}\left[5S + S'\right]$$ (for stresses in the same direction)

$$\delta = \frac{2}{3}\frac{r}{E}\left[5S - S'\right]$$ (for stresses in the opposite direction)

5.1.4.8.2 Plate and Sleeve Deformation

$$\Delta_{UN} = \Delta_{Pl\,Res.} + \Delta_{Sleeve}$$

$$\Delta_{UN} = \frac{S_1}{E}(1-\nu)R + \frac{S_l\left(t_{pl}\right)R^2\lambda}{2Et_{sleeve}}^{*}$$

$$\Delta_{UN} = \frac{S_1}{E}\left[R(1-\nu) + \frac{t_{pl}\ R^2\lambda}{2t_{sleeve}}\right]$$

$$S_1 = \frac{\Delta_{UN}E}{R\left[(1-\nu) + \frac{t_{pl}\ R\lambda}{2t_{sleeve}}\right]}$$

$$S_{sleeve} = \frac{S_1\left(t_{pl}\right)R\lambda}{2t_{sleeve}}$$

$$S_{sleeve} = \frac{\Delta_{UN}E\ \ t_{p1}\ R\lambda}{R\left[(1-\nu) + \frac{t_{p1}\ R\lambda}{2t_{sleeve}}\right]2t_{sleeve}}$$

where:

$$\lambda = \left[\frac{3(1-\nu^2)}{R^2 t^2_{sleeve}}\right]^{\frac{1}{4}}$$

*S_1 = Stress in Liner
t_{pl} = plate thickness, in.

R = radius, in.
ν = poisson ratio
E = modulus of elasticity

5.1.4.8.3 Summary

Penetration	Stress in Sleeve (ksi)	Stress in Liner (ksi)
Air purge	-23.8	-19.5
Main steam	-33.4	-27.94
Typical mech. Penetration	-31.0	-31.1
Electrical penetration		
A) C and T[1]	-22.5	-29.5
B) T and T[1]	+18.2	+19.7
Fuel transfer		
A) C and T[1]	-25.7	-25.6
B) T and T[1]	+20.8	+16.6

[*Note – 1. First letter represents direction of vertical liner stress; second letter represents direction of horizontal liner stress, C signifies compression and T signifies tension.*]

A) Ignores effects of insulation in the vertical direction.

B) Considers effects of insulation.

Conservative Assumptions

1. The weld pressurization channel stiffens the area.

2. The liner alone was designed for stress concentration effects while the cracked concrete was ignored.

3. The unrestrained growth is based on maximum growth from a stress concentration consideration.

4. The main steam and mechanical penetrations have been considered in a noninsulated zone when they are just inside the insulated zone. The compression in the hoop direction will be greatly reduced or perhaps go into tension, thus reducing the stresses.

5. The allowable stress in the sleeve is 56,700 psi except for the stainless steel fuel transfer penetration, which is 49,500 psi. These values come from Table N-421 and Figure N-414 of the ASME Nuclear Vessel Code Section III.

5.1.5 <u>Primary System Supports</u>

In 1989, the NRC approved changes to the design bases with respect to dynamic effects of postulated primary loop pipe ruptures, as discussed in Section 4.1.2.4.

In 2000, an analysis (Reference 19) of the reactor coolant loop and its component supports, which incorporates the NRC approved changes, was performed to reflect the replacement steam generator and removal of sixteen of the original twenty-four steam generator support frame hydraulic snubbers. The analysis also reflected the de-activation of the original horizontal and vertical pipe rupture restraints, located on the cross over legs at the steam generator end. Based on this revised analysis, it was concluded that the Unit 2 reactor coolant system can withstand the combination of blowdown and seismic loads within acceptable stress limits. By reducing the number of snubber and de-activating the rupture restraints the extent of maintenance, inspection and testing requirements is reduced and the reliability of the Reactor Coolant System is enhanced by reducing the possibility of equipment malfunction. In 2003, the reactor coolant loop and its component supports were re-analzed due to a power uprate. This latest analysis does not consider the coincident combination of blowdown and seismic loads.

The original design basis is described in the following paragraphs.

The primary system supports, steam generator, reactor coolant pump, pressurizer, and reactor vessel were designed to withstand pipe break or seismic acceleration based on the following:

1. The break is either a circumferential or longitudinal pipe rupture of area equivalent to the pipe cross section occurring anywhere in the system piping. The longitudinal rupture occurs at any point 360 degrees around the pipe. The support system is designed to withstand the steady thrust equivalent to the product of system operating pressure and pipe rupture area without exceeding yield stress in the support members. The stress limits on the vessels and piping are tabulated in Section 1.11. The component supports prevent rupture of reactor coolant piping in the remaining intact loops which could result from an assumed rupture in any one loop, thereby ensuring that the path for safety injection flow to the core is available. Additionally, the supports are designed to prevent secondary piping rupture as a result of rupture in the primary loop and vice versa.

2. The nuclear steam supply system and its support system are designed such that the nuclear steam supply system is capable of continued safe operation for the combination of normal loads and the design earthquake loading. The equipment and supports operate within normal design limits for the design earthquake. The system and its supports are also designed to withstand the maximum potential earthquake without loss of function. The seismic response curves for both the design and maximum potential earthquake and the stress limits are presented in Section 1.11. Component loads are obtained from the curve using the appropriate period and damping.

3. The primary system supports were not originally designed to resist combined seismic and accident loads. They were designed as statically uncoupled component supports.

   A complete reactor coolant system loop, including the steam generator and the reactor coolant pump supports, has been analyzed for combined dead, seismic, and blowdown loads. Stresses were determined by means of the three-dimensional frame computer program, STRUDL. The dead load assumed is the flooded weight of the component. The seismic load considered is 0.6g horizontal acceleration times the flooded mass of the component at the center of gravity of the component acting in the N-S, E-W, NW-SE and SW-NE directions analyzed

separately. The horizontal earthquake component acting on the steam generator is assumed to be carried by the upper steam generator. The vertical component of earthquake is assumed equal to 0.4g acting simultaneously with the horizontal load at the center of gravity of both the pump and steam generator. The system was analyzed for each separate accident or pipe rupture resulting in a jet load equal to 1500 kips as shown in Figure 5.1-34.

The combined dead plus seismic plus accident maximum resultant member axial stress and axial plus bending stress (in parentheses) for the steam generator and pump supports is shown in Figures 5.1-35 through 5.1-42 (stresses are expressed in ksi.) The section views of the support shown can be identified by the isometric views of the pump and steam generator supports shown in Figures 5.1-43 and 5.1-44. Negative values indicate compression and positive values indicate tensile stress. Since response of the primary systems is elastic, deformations are very small and were not considered as design parameters required to verify the design adequacy of the supports.

It should be noted the stresses shown are not for a particular combined blowdown or seismic load case but rather the worst combination for a given member; hence, the values shown are upper limits for each member and could not in fact actually occur in the combination shown. It should also be noted that the primary support structures are designed as trusses rather than frames hence the bending stresses indicated are secondary in nature.

5.1.5.1 <u>Steam Generator</u>

The steam generators are supported within a caged structural system, consisting of four connected trusses, all welded together, fabricated of carbon steel members, with provisions for limited movement of the structure in a horizontal direction to accommodate piping expansion with a system of "Lubrite" plates, hydraulic snubbers, guides, and stops. The "Lubrite" plates, hydraulic snubbers, guides, and stops were originally designed as a rigid support to resist the action of seismic and pipe break loads. In 2000, the number of hydraulic snubbers supporting the steam generator frame in the direction of the hot leg, has been reduced from the original six down to two per steam generator. The two remaining snubbers are located at the upper support point of the frame at Elevation 92'-0". The analysis of the reactor coolant loop and of the steam generator support structure accounts for the replacement steam generator and for the reduced number of hydraulic snubbers. The following are loading conditions that the structure was originally designed to resist:

1. Vertical dead weight of pipe and vessel, flooded = 1,000 kips

2. Seismic loads:

   a. Horizontal load of 474 kips acting at the centroid of the steam generator vessel, located near the top of the support structure, which is directly transferred to the hydraulic snubbers, guides, and stops, and in turn to the bottom of the 2-ft thick concrete operating floor slab at elevation 93-ft.

   b. Vertical load of 320 kips transferred as axial load to the base plates and anchor bolts at elevation 46-ft.

3. Primary system - longitudinal pipe rupture:

   a. Reaction at the nozzle of the steam generator from the pipe between the reactor and the steam generator elbow, produces a force of 1090 kips in any direction and an overturning moment or torsional moment of (1090 kips x 4.25-ft) 4632-ft-kips. Overturning and torsional moments are resisted by the support system at elevation 46-ft and horizontal forces are distributed, through the truss action, to elevations 46-ft and 93-ft.

   b. Reactions at the nozzle of the steam generator from the pipe between the steam generator elbow and reactor coolant pump elbow, produces a force of 850 kips in any direction and a torsional moment or overturning moment of (850 kips x 5.0-ft) 4250-ft-kips. Overturning and torsional moments are resisted by the support system at elevation 46-ft, and horizontal forces are distributed, through the truss action, to elevations 46-ft and 93-ft.

4 Primary system - circumferential break:

   a. Reactions at the nozzle of the steam generator from the pipe between the reactor and steam generator produces a horizontal force of 1490 kips. This force is transferred through the vessel support to the two vertical trusses of the structural system, which in turn, transmits it as horizontal reactions at the slabs at elevations 46-ft and 93-ft. The moment produced by this force is (1490 kips x 2-ft) 2980-ft-kips and is less than the dead load resisting moment (500 kips x 10-ft) 5000-ft-kips, and the vertical forces at elevation 46-ft are all compressive, no uplift.

   b. Reactions at the nozzle of the steam generator from a pipe between the steam generator and the reactor coolant pump produces a horizontal force of 1700 kips plus an overturning moment of (1700 kips x 4.25-ft) 7225-ft-kips, or a vertical force of 1700 kips and an overturning moment of (1700 kips x 5.33-ft) 9061-ft-kips. The horizontal force and moments are transferred to the structural system and the reactions are resisted at the slabs at elevations 46-ft and 93-ft, or the vertical force and moment are resisted at elevation 46 ft.

5. Secondary system - longitudinal rupture in steam pipe:
   Reactions at the nozzle of the steam generator from the steam pipe longitudinal rupture at the top of the vessel produce:

   a. Horizontal force of 600 kips and a torsional moment of 2400-ft-kips. Horizontal force is transferred through the vessel to the structural support system, which in turn transmits it as horizontal reactions to the slabs at elevations 46-ft and 93-ft. The torsional moment is transferred through the vessel to the structural system, which in turn, transmits it to the base at elevation 46-ft.

   b. Vertical upward or downward force of 600 kips and an overturning moment of 2400-ft-kips. Upward forces are overcome by the operating weight of the steam generator. Downward force is added to the operating weight and transferred to the base at elevation 46-ft. Overturning moment is transferred

through the vessel supports to the structural system, which in turn, transmits it as vertical reactions at the base, elevation 46-ft.

6. Secondary system - circumferential break in steam pipe:

   Reaction at the nozzle of the steam generator from the steam pipe guillotine break at the top of the vessel produces a horizontal force of 600 kips. This force is transferred through the vessel to the structural system, which in turn transmits it as horizontal reactions of 1085 kips at elevation 93-ft and 485 kips at elevation 46-ft.

7. Secondary system - feedwater pipe breaks:

   The reactions from circumferential and longitudinal pipe breaks in the feedwater system are resisted in a manner similar to steam pipe breaks listed under preceding sections (5) and (6), but are much smaller in magnitude. Maximum longitudinal 1600-ft-kips, maximum circumferential 200 kips.

5.1.5.2 Reactor Coolant Pump

The reactor coolant pump is supported on a three-legged structural system consisting of three connected trusses fabricated of carbon steel members, structural sections and pipe, supported from elevation 48-ft-6-in. Provisions for limited movement of the structure in any horizontal direction to accommodate piping expansion is accomplished with a sliding "Lubrite" base plate arrangement and a system of tie rods and anchor bolts that restrain the structure from movement beyond the calculated limits. To improve the ability of the reactor coolant pumps to meet combined LOCA and seismic loads, two of the reactor coolant pump holddown bolts have been replaced with higher strength ASTM A540 steel bolts.

The following are loading conditions that the structure was originally designed to resist:

1. Vertical dead weight of pipe and pump flooded = 206 kips.

2. Seismic:
   a. Horizontal load of approximately 117 kips acting at the centroid of the pump assembly, which is transferred by the structural system and piping to the tie rods and base of the supporting structure at elevation 48-ft-6-in. This load includes the seismic effect of the support self-weight.

   b. Vertical seismic load of approximately 78 kips transferred directly as axial load to the base plates and anchor bolts. This load includes the seismic effect of the support self-weight.

3. Primary system - longitudinal rupture:
   a. Reaction at the nozzle of the pump from a pipe break in the pipe between the steam generator elbow and pump elbow produces a torsional moment of 3825-ft-kips, together with a horizontal force of 850 kips or an overturning moment of 3825-ft-kips, together with a vertical up or down force of 850 kips. Torsional forces are resisted by the structural stability of the primary piping connected to the pump.

Reactions from horizontal forces are resisted by the tie rods connected to the steam generator and reactor support structures. Forces caused by an overturning moment are resolved into horizontal and vertical components, which are resisted by tension in the anchor bolts, axial load on the foundations, and tension in the tie rods.

b. Reaction at the nozzle of the pump from a pipe break in the pipe between the pump and the reactor, produces a torsional moment of 6880-ft-kips, together with a horizontal force of 1165 kips, or an overturning moment of 6880-ft-kips, together with a vertical up or down force of 1165 kips.

Torsional forces are resisted by the structural stability of the primary piping connected to the pump. Reactions from the horizontal forces are resisted by the tie rods connected to the walls.

Forces caused by an overturning moment are resolved into horizontal and vertical components, which are resisted by:

(1) Tension in the anchor bolts.
(2) Axial load on the foundations.
(3) Tension in the tie rods.

4. Primary system - circumferential break:

a. Reactions at the nozzle of the pump from a pipe break in the pipe between the steam generator and pump produces a horizontal force on the structure of 1700 kips. This force is resisted directly by the bumper located against the elbow of the pipe. Components of the force are then transferred to the base of the structure and the tie rods connecting the pump support to the steam generator support system.

b. Reactions at the nozzle of the pump from a pipe break in the pipe between the pump and the reactor produces a torsional moment of 3240-ft-kips and a horizontal force of 1340 kips on the structure.

Torsional forces are resisted by the structural stability of the remaining primary piping connected to the pump.

Reactions from the horizontal forces are resisted by tie rods connected to the walls.

5.1.5.3 <u>Pressurizer</u>

Pressurizer is supported on a free-standing structural system, consisting of six connected trusses fabricated of carbon steel members, all welded together and secured at the base by anchor bolts at elevation 46-ft.

The following are loading conditions that the structure has been designed to resist:

1. Vertical dead weight of pipe and vessel flooded is 360 kips. The self-weight of the support is 21 kips.

2. Seismic:

   a. Horizontal seismic load of 174 kips acting at the centroid of the pressurizer vessel, which coincides in elevation with the slab at elevation 95-ft, is directly transferred through the concrete embedded guides to the slab. This load excludes the seismic effect of the support self-weight.

   b. Vertical seismic load of 123 kips transferred through the structural system as axial forces to the base plates and anchor bolts at elevation 46-ft. This load excludes the seismic effect of the support self-weight.

3. Longitudinal pipe rupture

   a. Reaction at the surge pipe nozzle of the pressurizer produces either a torsional moment of 734-ft-kips and a horizontal force of 234 kips or an overturning moment of 734-ft-kips and a horizontal or vertical force of 234 kips.

      These moments and forces are resisted by the structural system and transferred to the base at elevation 46-ft.

4. Circumferential pipe break:

   a. Reaction at the surge pipe nozzle of the pressurizer produces a horizontal force of 234 kips and an overturning moment of 734-ft-kips.

      These moments and forces are resisted by the structural system and transferred to the base at elevation 46-ft.

5.1.5.4 <u>Reactor Vessel Support Girder</u>

The reactor vessel is supported on four cooling plates that are fastened to the top flange of a circular box section ring girder, fabricated of carbon steel plates. The bottom flange of the girder is in continuous contact with a nonyielding concrete foundation.

In addition to the reactor vessel weight and piping reactions of the girder has been designed to support the conditions of loading for pipe break and seismic forces as outlined in Figure 5.1-45.

5.1.5.5 <u>Reactor Vessel Rupture</u>

The reactor pressure vessel is enclosed by a 6-ft thick circular reinforced concrete shield wall that is designed to sustain the internal pressure and provide missile protection for the containment liner in the highly unlikely failure of the reactor vessel due to a longitudinal split. All stresses will be maintained within specified minimum ultimate rebar tensile stress.

In the event of a circumferential reactor break, the 0.25-in. basemat liner plate at the bottom of the containment reactor cavity pit directly under the reactor vessel is protected by 2-ft of concrete with a 1-in. steel liner plate embedded on top of the concrete. Directly below the reactor cavity pit containment basemat liner plate, 4.5-ft of concrete is poured on rock. Refer to Figures 5.1-46 through 5.1-51.

As discussed in Section 5.1.3.7, in the event of reactor vessel failure, a pressure build up of 1000 psi and rebar stresses of 86 ksi (assuming all concrete is cracked) inside the pit due to release of reactor contents is assumed. Since the integrity of the wall is not jeopardized, the integrity of the vessel support that is supported on the wall will not be jeopardized. Deflection of the shield wall will not cause large stresses in the vessel support since a lubricated surface is provided on the shoes, allowing the vessel support to slide.

5.1.5.6 Circumferential Cracking

The worst circumferential crack location from the standpoint of downward missiles is just below the reactor coolant system piping nozzles. As the following calculations show, this missile will not violate the containment structure and liner integrity.

As a consequence of this circumferential crack, the downward missile represented by bottom vessel head has the following characteristics at the time of impact on the cavity floor:

1.	Weight:	381,000 lb
2.	Cross sectional area of crater:	63-ft^2
3.	Downward velocity:	213-ft/sec
4.	Concrete crushing strength:	4000 psi

The depth of penetration has been calculated by using the Petri formula for penetration into an infinite, thick concrete slab, as reported in Nav. Docket P-51:

$$D = K\frac{W}{A} \log_{10} \left(1 + \frac{V^2}{215{,}000}\right)$$

where:

D = depth of penetration, ft
K = penetration coefficient for 4000 psi concrete
W = missile weight, lb
A = missile area, ft^2
V = missile velocity, ft/sec

The following parameters have been used:

$K = 2.8 \times 10^{-3}$
W = 381,000 lb
A = 63-ft^2
V = 213 ft/sec

The result is a depth of penetration of 1.4-ft.

As mentioned above, the 0.25-in. basemat liner is covered by 2-ft of concrete with a 1-in. steel plate on top. As it can be readily seen, even neglecting the 1-in. steel plate in the penetration calculations, the containment liner will not be reached.

5.1.5.7 Longitudinal Splitting

The cavity wall is designed to withstand the forces and internal pressurization associated with a longitudinal split without gross damage. See Section 5.1.3.7 for a discussion of the analysis of this assumed accident condition.

5.1.6 Containment Structure Design Evaluation

5.1.6.1 Reliance On Interconnected Systems

The containment leakage limiting boundary is provided in the form of a single, carbon steel liner on the vessel having double barrier weld channels and penetrations. Each system whose piping penetrates this boundary is designed to maintain isolation of the containment from the outside environment. Provision is made to continuously pressurize penetrations and weld channels and to monitor leakage from this pressurization.

5.1.6.2 System Integrity and Safety Factors

Pipe Rupture - Penetration Integrity - The penetrations for the blowdown and sample lines are designed so that the penetrations are stronger than the piping system and so that the vapor barrier will not be breached due to a hypothesized pipe rupture. The pipe rupture loads for the main steam and feedwater lines are resisted by the supports located away from their penetrations and do not affect the integrity of the penetrations for these lines.

Major Component Support Structures - The support structures for the major components are designed to resist all thrust forces, moments and torques associated with either a reactor coolant system or main steam pipe break. All primary structural steel elements are designed for stresses not exceeding yield stress due to these forces.

Containment Structure Components Analyses - The details of radial, longitudinal, and horizontal shear analyses for the containment reinforced concrete are given in Section 5.1.3.

5.1.6.3 Performance Capability Margin

The containment structure is designed based upon limiting load factors, which are used as the ratio by which accident and earthquake loads are multiplied for design purposes to ensure that the load/deformation behavior of the structure is one of elastic, low strain behavior. This approach places minimum emphasis on fixed gravity loads and maximum emphasis on accident and earthquake loads. Because of the refinement of the analysis and the restrictions on construction procedures, the load factors primarily provide for a safety margin on the load assumptions. Load combinations and load factors used in the design, which provide an estimate of the margin with respect to all loads, are tabulated in Section 5.1.2.

5.1.7 Liner Insulation

Insulation is provided on approximately the first 43-ft of the containment liner to limit the temperature rise in the liner under accident conditions to 80°F above ambient and thereby avoid excessive liner compressive stress during the accident. The first 18-ft (elev. 46-ft to 64-ft, except in the piping penetration area in the southeast quadrant where the insulation rose only 16-ft to the 62-ft elevation) was installed as part of the original containment design. In 1973 an additional 25-ft (elev. 64-ft to 89-ft) was added. The first 18-ft (elev. 64-ft to 82-ft) covers the

entire circumference of the liner while the upper 7-ft (elev. 82-ft to 89-ft) only covers part of the circumferential area in the north and south-southwest quadrants where the main steam and feedwater lines extend up along the crane wall. The insulation panels are attached to the steel containment liner by means of 3/16-in. diameter stainless steel studs welded to the liner on the basis of six per panel. The insulation panels are protected by stainless steel jacketing on the exposed faces and sealed at the joints. Details of the insulation installation are given in Table 5.1-2.

The insulation has been designed to meet the following operational requirements:

1. Normal operating temperature of 120°F. (The maximum normal operating temperature of the containment was changed from 120°F to 130°F by Amendment 149 to the Facility Operating License DPR-26 for IP-2 dated March 27, 1990. Evaluations performed show the insulation material used on the containment liner is adequate for use at the higher operating temperature.)

2. Under accident conditions, the rise in liner temperature not to exceed 80°F. The analyses performed to support the Stretch Power Uprate (SPU) also performed analyses of the containment liner under the most limiting conditions for liner stress and showed a temperature rise well under allowed 80°F.

3. Insulation panels to be rated nonburning in accordance with ASTM procedure D–1692.

4. To be removable by sections for inspection of the containment liner.

5.1.8 Minimum Operating Conditions (For Containment Integrity)

Containment integrity internal pressure limitations and leakage rate requirements are established in the facility Technical Specifications.

5.1.9 Containment Structure-Inspection And Testing

5.1.9.1 Initial Containment Leakage Rate Testing

Criterion: Containment shall be designed so that integrated leakage rate testing can be conducted at the peak pressure calculated to result from the design basis accident after completion and installation of all penetrations and the leakage rate shall be measured over a sufficient period of time to verify its conformance with required performance. (GDC 54)

After completion of the containment structure and installation of all penetrations and weld channels, an initial integrated leakage rate test was conducted at the containment design pressure (47 psig), maintained for a minimum of 24 hr, verifying that the leakage rate is no greater than 0.1-percent by weight of the containment volume per day at design basis accident conditions. This leakage rate test was performed using the absolute method. In addition, a reduced pressure integrated leakage rate test was conducted at a pressure not less than 50 percent of the containment design pressure and maintained for a minimum of 24 hr.

5.1.9.2 <u>Periodic Containment Leakage Rate Testing</u>

Criterion: The containment shall be designed so that an integrated leakage rate can be periodically determined by test during plant lifetime. (GDC 55)

The containment is tested in accordance with 10 CFR 50 Appendix J as discussed in section 5.1.12.

A leak rate test at the containment design pressure using the same method as the initial leak rate test can be performed at any time during the operational life of the plant, provided the plant is not in operation and precautions are taken to protect instruments and equipment from damage.

5.1.9.3 <u>Provisions for Testing of Penetrations</u>

Criterion: Provisions shall be made to the extent practical for periodically testing penetrations, which have resilient seals or expansion bellows to permit leak tightness to be demonstrated at the peak pressure calculated to result from occurrence of the design basis accident. (GDC 56)

Penetrations are designed with double seals, which are continuously pressurized above accident pressure. The large access openings such as the equipment hatch and personnel air locks are equipped with double gasketed doors and flanges with the space between the gaskets connected to the pressurization system. The system uses a supply of clean, dry, compressed air that will place the penetrations under an internal pressure above the peak calculated accident pressure.

A permanently piped monitoring system is provided to continuously measure leakage from all penetrations.

Leakage from the monitoring system is checked by continuous measurement of the integrated makeup air flow. In the event excessive leakage is discovered, each penetration can then be checked separately at any time.

5.1.9.4 <u>Provisions for Testing of Isolation Valves</u>

Criterion: Capability shall be provided to the extent practical for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits. (GDC 57)

Capability is provided to the extent practical for testing the functional operability of valves and associated apparatus during periods of reactor shutdown.

Initiation of containment isolation employs coincidence circuit, which allow checking of the operability and calibration of one channel at a time. Removal or bypass of one signal channel place that circuit in the half-tripped mode.

Hydrostatic tests of isolation valves in series are performed by first testing the upstream valve with the second valve open, then opening the upstream valve and closing the second valve, so that each valve will have an independent test.

The main steam and feedwater barriers and isolation valves in systems that connect to the reactor coolant system are hydrostatically tested to measure leakage.

Valves in the residual heat removal system are not considered to be isolating valves in the usual sense inasmuch as the system would be in operation under accident conditions.

Field and operational inspection and testing have been divided into three phases:

1. Construction tests; those taking place during erection of the containment building liner.

2. Preoperational tests; those taking place after the containment structure was erected and all penetrations were complete and installed.

3. Postoperational tests; monitoring during reactor operation.

5.1.10 Construction Tests

During erection of the liner, the following inspection and tests were performed.

5.1.10.1 Bottom Liner Plates

All liner plate welds are tested for leaktightness by vacuum box. The box is evacuated to at least a 5 psi pressure differential with the atmospheric pressure.

After completion of a successful leak test, the welds were covered by channels. A strength test was performed by applying 54 psig air pressure to the channels in the zone for a period of 15 min.

The zone of channel covered welds was pressurized to 47 psig with a 20-percent by weight of freon-air mixture. The entire run of the channel-to-plate welds was then traversed with a halogen leak detector.

The sensitivity of the leak detector was 1 x 10^{-9} standard cc per second. The sniffer was held approximately 0.5-in. from the weld and traversed at a rate of about 0.5-in./sec. The detection of any amount of halogen indicated a leak requiring weld repairs and retesting.

After the halogen test was completed, all liner welds not accessible for radiography were pressurized with air to 47 psig and soap-tested. Any leaks indicated by bubbles were repaired and retested. Where leaks occurred, welds were removed by arc gouging, grinding, chipping, and/or machining before rewelding. In addition, the zone of channels was held at the 47 psig air pressure for a period of at least 2 hr. The drop in pressure did not exceed the equivalent of a leakage of 0.05-percent of the containment building volume per day. Compensation for change in ambient air temperature was made.

5.1.10.2 Vertical Cylindrical Walls and Dome

For the liner, a complete radiograph was made of the first 10-ft of full penetration weld made by each welder or welding operation. A minimum of a 12-in. film "spot" radiograph was made every 50-ft of weld thereafter on the side walls and dome, except where backup plates were

used. The radiograph films were reviewed by United Engineers and Constructors. When a spot radiograph showed defects that required repair, two adjacent spots were radiographed. If defects requiring repair were shown in either of these, all of the welding performed by the responsible operator or welder was 100-percent radiographed to determine the area of defect.

The performance and acceptance standards for all radiography were ASME Section VIII, Paragraph UW51.

The liner plate-to-plate welds were tested for leaktightness by vacuum box techniques. After successful completion of the spot radiography and vacuum box tests and subsequent repair of all defects, the channels were welded in place over all seam welds in a predetermined zone. A strength test was performed on the liner plate weld and the channel weld by pressurizing the channel with air at 54 psig for 15 min. In addition, each zone of channel covered weld was leak-tested using the freon-air mixture at 47 psig.

In locations where radiography was not possible, such as the lower courses of shell plates where backup plates were used, and where liner bottom welds and floor plate welds were made to angles and tees, the liner fabricator welded on a 2-in. long overrun coupon. The overrun coupon was chipped off, marked for location and given to United Engineers and Constructors for testing. These welds were also vacuum box tested.

Welded studs were visually inspected, and at least one at the beginning of each day's work and another at approximately mid-day were bend-tested to 45 degrees for each welder. Studs failing visual or bend-testing were removed.

While the liner is not a pressure vessel, industry experience has shown that leaks in pressure vessels normally occur at joints. For this reason, and following current liner fabrication practice, there was no radiographic or other nondestructive examination of liner plate.

5.1.10.3 Penetrations

Strength and leak tests of individual penetration internals and closures and sleeve weld channels were performed in a similar manner to the above and all leaks repaired and the penetration or weld channel retested until no further leaks were found. See Figures 5.1-53 through 5.1-56 for the areas of the containment and liner, which were instrumented for the strength test.

5.1.11 Preoperational Tests

All penetrations and the welds joining these penetrations to the containment liner and the liner seam welds were designed to provide a double barrier, which can be continuously pressurized at a pressure higher than the design pressure of the containment. This blocks all of these potential sources of leakage with a pressurized zone and at the same time provides a means of monitoring the leakage status of the containment, which is more sensitive to changes in the leakage characteristics of these potential leakage sources.

After the containment building was complete with liner, concrete structures, and all electrical and piping penetrations, equipment hatch and personnel locks were in place, the following tests were performed.

5.1.11.1 Strength Test

A pressure test was made on the completed building using air at 54 psig. This pressure was maintained on the building for a period of at least 1 hr. During this test, measurements and observations were made to verify the adequacy of the structural design. For a description of observations, cracks, strain gauges, etc., refer to Reference 18.

5.1.11.2 Integrated Leakage Rate Test: (Type A)

The integrated leakage rate tests were performed on the containment building at 47 psig using the absolute method. This leakage test was performed with the double penetration and weld channel zones open to the containment atmosphere. The leakage rate demonstrated by this test was equal to or less than 0.1-percent of the containment free volume per day at design basis accident conditions. After it was assured that there were no defects remaining from construction, a sensitive leak rate test was conducted.

5.1.11.3 Sensitive Leak Rate Test: (Type B)

The sensitive leak rate test included only the volume of the weld channels and double penetrations. This test was considered more sensitive than the integrated leakage rate test, as the instrumentation used permitted a direct measurement of leakage from the pressurized zones. The sensitive leak rate test was conducted with the penetrations and weld channels at 50 psig and with the containment building at atmospheric pressure. The leak rate for the double penetrations and weld channel zones was equal to or less than 0.2-percent of the containment free volume per day.

5.1.11.4 Containment Isolation Valve Test: (Type C)

These tests were conducted to detect leaks through certain containment isolation valves.

5.1.12 Postoperational Tests

Containment testing is conducted in accordance with the Technical Specifications and 10 CFR Appendix J, including integrated leakage rate tests at the containment design pressure, In 1997, the Technical Specifications were amended to allow the use of 10 CFR 50 Appendix J, Option B (as modified by approved exemptions) and NRC Regulatory Guide 1.163 dated September 1995 for integrated leakage rate tests, air lock tests, and containment isolation valve operability tests.

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TABLE 5.1-1
Flooded Weights - Containment Building

Item	Flooded Operating Weight, lbs
Pressurizer -1	346,000
Steam generators - 4	3,746,000
Reactor - 1	
(a) Vessel	868,000
(b) Internals	420,000
(c) Piping	1,000,000
Reactor pumps - 4	824,000
Accumulator tanks - 4	529,000
175-ton polar crane - 1	650,000
Ventilation fans - 5	656,000
Reactor coolant drain tank - 1	20,000
Pressure relief tank - 1	100,000
Other miscellaneous equipment	100,000
Total	9,259,000

TABLE 5.1-2
Containment Liner Insulation Properties

1. Elevation 46-ft to 64-ft liner insulation:

- 1-1/4-in. polyvinylchloride insulation, Vinylcel, as manufactured by Johns-Mansville and 1-1/2 in. Pittsburgh Corning Foamglass Insulation.

- 0.019-in. thick stainless steel jacket (exposed side) except for areas using Pittsburgh Corning Foamglass Insulation in which a jacket thickness of 0.024" is used.

- Insulation adhesive is Johns-Manville Dutch Brand FN12 or an approved equal.

2. Elevation 64-ft to 89-ft liner insulation:[1]

- 1.5-in. thick FOAMGLAS[R] with density of 8.5 to 9 lb/ft^3, as manufactured by Pittsburgh Corning Corporation. This insulation has a thermal conductivity of 0.5 - 0.525 BTU-in/hr-ft^2-°F and a specific heat (Cp) of 0.18 BTU/lb-°F.

- 1/16-in. commercial grade pure asbestos paper backing adjacent to the liner plate on the unexposed face.

- The adhesive bonding the FOAMGLAS[R] to the asbestos paper is Cadoprene No. 434 and bonding the stainless steel jacket to the FOAMGLAS[R] is Cadoseal No. 700 by Epolux Manufacturing Corporation.

Note:
[1] Insulation from Elevation 82-ft to 89-ft only covers part of the circumferential area in the north and south-southwest quadrants.

5.1 FIGURES

5.2 CONTAINMENT ISOLATION SYSTEM

5.2.1 Design Basis

Each system whose piping penetrates the containment leakage limiting boundary is designed to maintain or establish isolation of the containment from the outside environment under the following postulated conditions:

1. Any accident for which isolation is required (severely faulted conditions) with

2. A coincident independent single failure or malfunction (expected fault condition) occurring in any active system component within the isolated bounds.

Piping penetrating the containment is designed for pressures at least equal to the containment design pressure. Containment isolation valves are provided as necessary in lines penetrating the containment to ensure that no unrestricted release of radioactivity can occur. Such releases might be due to rupture of a line within the containment concurrent with a loss-of-coolant accident, or due to rupture of a line outside the containment that connects to a source of radioactive fluid within the containment.

In general, isolation of a line outside the containment protects against rupture of the line inside concurrent with a loss-of-coolant accident, or closes off a line, which communicates with the containment atmosphere in the event of a loss-of-coolant accident.

Isolation of a line inside the containment prevents flow from the reactor coolant system or any other large source of radioactive fluid in the event that a piping rupture outside the containment occurs. A piping rupture outside the containment at the same time as a loss-of-coolant accident is not considered credible, as the penetrating lines are seismic Class I design up to and including the second isolation barrier and are assumed to be an extension of containment.

The isolation valve arrangement provides two barriers between the reactor coolant system or containment atmosphere and the environment.

System design is such that failure of one valve to close will not prevent isolation, and no manual operation is required for immediate isolation. Automatic isolation is initiated by a containment ventilation isolation signal, a Phase A isolation signal ("T" signal), a Phase B isolation signal ("P" signal), or manually. See Section 5.2.4 or Chapter 7.0 for further details.

The containment isolation valves have been examined to ensure that they are capable of withstanding the maximum potential seismic loads.

To ensure their adequacy in this respect:

1. Valves are located in a manner to reduce the accelerations on the valves. Valves suspended on piping spans are reviewed for adequacy for the loads to which the span would be subjected. Valves are mounted in the position recommended by the manufacturer.

2. Valve yokes are reviewed for adequacy and strengthened as required for the response of the valve operator to seismic loads.

3. Where valves are required to operate during seismic loading, the operator forces are reviewed to ensure that system function is preserved. Seismic forces on the operating parts of the valve are small compared to the other forces present.

4. Control wires and piping to the valve operators are designed and installed to ensure that the flexure of the line does not endanger the control system. Appendages to the valve, such as position indicators and operators, are checked for structural adequacy.

Isolation valves are provided as necessary for all fluid system lines penetrating the containment to ensure at least two barriers for redundancy against leakage of radioactive fluids to the environment in the event of a loss-of-coolant accident. These barriers, in the form of isolation valves or closed systems, are defined on an individual line basis. In addition to satisfying containment isolation criteria, the valving is designed to facilitate normal operation and maintenance of the systems and to ensure reliable operation of other engineered safeguards systems.

With respect to numbers and locations of isolation valves, the criteria applied are generally those outlined by the seven classes described in Section 5.2.2 below.

5.2.2 System Design

The seven classes listed below are general categories into which lines penetrating containment may be classified. The seal water referred to in the listing of categories is provided by the isolation valve seal water system described in Section 6.5. The following notes apply to these classifications.

1. The "not-missile-protected" designation refers to lines that are not protected throughout their length inside containment against missiles generated as the result of a loss of coolant accident. These lines, therefore, are not assumed invulnerable to rupture as a result of a loss of coolant.

2. In order to qualify for containment isolation, valves inside the containment must be located behind the missile barrier for protection against loss of function following an accident.

3. Manual isolation valves that are locked closed or otherwise closed and under administrative control during power operation qualify as automatic trip valves.

4. A check valve qualifies as an automatic trip valve in certain incoming lines not requiring seal water injection.

5. The double disc type of gate valve is used to isolate certain lines. When sealed by water injection, this valve provides two barriers against leakage of radioactive liquids or containment atmosphere.

6. In lines isolated by globe valves and provided with seal water injection, the valves are generally installed so that the seal water wets the stem packing.

7. Loss of seal water through those isolation valves closed only by a containment isolation phase B signal is prevented by solenoid operated valves in the seal water injection lines. Excessive loss of seal water through motor operated isolation valves that could fail to close in response to a containment isolation signal is limited by flow restrictive orifices installed in the seal water lines. A water seal at the failed valve is ensured by proper slope of the protected line, or a loop seal, or by additional valves on the side of the isolation valves away from the containment.

8. Isolated lines between the containment and the second outside isolation valve are designed to the same seismic criteria as the containment vessel and are considered to be an extension of containment.

5.2.2.1 Class 1, Outgoing Lines, Reactor Coolant System

Outgoing lines connected to the reactor coolant system that are normally or intermittently open during reactor operation are provided with at least two automatic trip valves in series located outside the containment. Automatic seal water injection is provided for lines in this classification.

An exception to the general classification is the residual heat removal loop's reactor coolant system suction line, which has two barriers established by normally closed valves located outside containment.

5.2.2.2 Class 2, Outgoing Lines

Outgoing lines not connected to the reactor coolant system that are normally or intermittently open during operation and not missile protected or that can otherwise communicate with the containment atmosphere following an accident are provided at a minimum with two automatic trip valves in series or a single automatic double-disc gate valve outside containment. Automatic seal water injection is provided for lines in this classification. Most of these lines are not vital to plant operation following an accident.

5.2.2.3 Class 3, Incoming Lines

Incoming lines connected to open systems (i.e., systems that are in some way connected to the containment environment) outside containment, and not missile protected or that can otherwise communicate with the containment atmosphere following an accident are provided with one of the following arrangements outside containment:

1. Two automatic trip valves in series, with automatic seal water injection. This arrangement is provided for lines that are not necessary to plant operation after an accident.

2. Two manual isolation valves in series, with manual seal water injection. This arrangement is provided for lines that remain in service for a time, or are used periodically, subsequent to an accident.

Incoming lines connected to closed systems outside containment, and not missile protected or that can otherwise communicate with the containment atmosphere, are provided either with two isolation valves in series outside containment with seal water injection, or at a minimum, with

one check valve or normally closed isolation valve located either inside or outside containment. The closed piping system outside containment provides the necessary isolation redundancy for lines that contain only one isolation valve.

Exception is the containment spray headers, for which valving is based on safeguards requirements.

5.2.2.4 Class 4, Missile Protected Lines

Incoming and outgoing lines that penetrate the containment and that are normally or intermittently open during reactor operation and are connected to closed systems inside the containment and protected from missiles throughout their length, are provided with at least one manual isolation valve located outside the containment. Seal water injection is not required for this class of penetration. An exception is the residual heat exchanger cooling water lines for which design is based on safeguards requirements.

5.2.2.5 Class 5, Normally Closed Lines Penetrating the Containment

Lines that penetrate the containment and that can be opened to the containment atmosphere but that are normally closed during reactor operation are provided with two isolation valves in series or one isolation valve and one blind flange.

5.2.2.6 Class 6, Special Service Lines

There are a number of special groups of penetrating lines and containment access openings. These are discussed below.

Each ventilation purge duct penetration is provided with two tight-closing butterfly valves, which are normally closed during reactor power operation and are actuated to the closed position automatically upon a containment isolation or a containment high radiation signal. One valve is located inside and one valve is located outside the containment at each penetration. The space between valves is pressurized by air from the weld channel and penetration pressurization system whenever they are closed. Blind flanges can also be used for containment isolation in place of automatic purge isolation valves, provided they meet the same design criteria as the isolation valves.

The containment pressure relief line is similarly protected. However, because the line is periodically opened during reactor power operation, three tight closing butterfly valves in series are provided, one inside and two outside the containment. These valves also are actuated to the closed position upon a containment isolation or containment high radiation signal. The two intravalve spaces are pressurized by air from the weld channel and penetration pressurization system whenever they are closed.

The equipment access closure is a bolted, gasketed closure that is air sealed during reactor operation. The personnel air locks consist of two doors in series with mechanical interlocks to ensure that one door is closed at all times. Each air lock door and the equipment closure are provided with double gaskets to permit pressurization between the gaskets by the weld channel and penetration pressurization system. (See Section 6.6.)

The fuel transfer tube penetration inside the containment is designed to present a missile-protected and pressurized double barrier between the containment atmosphere and the atmosphere outside the containment. The penetration closure is treated in a manner similar to

the equipment access hatch. A positive pressure is maintained between the double gaskets of the tube cover flange to establish the double barrier between the containment atmosphere and the inside of the fuel transfer tube. The interior of the fuel transfer tube is not pressurized. Seal water injection is not required for this penetration.

The following lines would be subjected to pressure in excess of the isolation valve seal water system pressure (~52 psig) in the event of an accident, due to operation of the safety injection system recirculation pumps:

1. Residual heat removal loop inlet line.
2. Residual heat removal loop outlet line.
3. Bypass line from residual heat exchanger outlet to safety injection pumps suction.
4. Residual heat removal pumps mini-flow line.
5. Residual heat removal loop sample line.
6. Recirculation pump discharge sample line.

These lines are isolated by double disk gate valves or double valves, which can be sealed by nitrogen gas from the high pressure nitrogen supply of the isolation valve seal water system. A self-contained pressure regulator operates to maintain the nitrogen injection pressure slightly higher than the maximum expected line pressure. These valves are closed or intermittently operated during reactor operation, and the nitrogen gas injection is manually initiated.

Lines, which are capable of communicating with the containment atmosphere (normally filled with air or vapor) include:

1. Steam jet air ejector return line to containment.
2. Containment radiation monitor inlet and outlet lines.

In an accident condition the space between the two containment isolation valves in each line are sealed by pressurizing with air from the weld channel and penetration pressurization system. The air is introduced into each space at approximately 2 psi above the containment design pressure through a separate line from the weld channel and penetration pressurization system. Parallel, redundant, fail-open valves in each injection line open on the appropriate containment isolation signal to provide a reliable supply of pressurizing air. A flow-limiting orifice in each injection line prevents excessive air consumption if one of these valves spuriously fails to open, or if one of the containment isolation valves fails to respond to the "trip" signal.

5.2.2.7 Class 7, Steam and Feedwater Lines

These lines and the shell side of the steam generator are considered basically as an extension of the containment boundary and as such must not be damaged as a consequence of reactor coolant system damage. This requires that the steam generator shell, feed and steam lines within the containment are to be classified and designed for the reactor coolant system missile-protected category. The reverse is also true in that a steam break is not to cause damage to the reactor coolant system.

5.2.3 Isolation Valves And Instrumentation Diagrams

Figures 5.2-1 through 5.2-28 show all valves in lines leading to the atmosphere or to closed systems on both sides of the containment barrier, valve actuation and preferential failure

modes, the application of "trip" (containment isolation) signals, relative location of the valves with respect to missile barriers, and the boundaries of seismic Class I designed lines. The item numbers in these figures align with the item numbers in Table 5.2-1. Figure 5.2-29 defines the nomenclature and symbols used.

5.2.4 Valve Parameters Tabulation

A summary of the fluid systems lines penetrating containment and the valves and closed systems employed for containment isolation is presented in Table 5.2-1. Each valve is described as to type, operator, position indication and open or closed status during normal operation, shutdown and accident conditions. Information is also presented on valve preferential failure mode, automatic trip by the containment isolation signal, and the fluid carried by the line.

Containment isolation valves are provided with actuation and control equipment appropriate to the valve type. For example, air-operated globe and diaphragm (Saunders Patent) valves are generally equipped with air diaphragm operators, with fail-safe operation provided by the control devices in the instrument air supply to the valve. Motor-operated gate valves are capable of being supplied from reliable onsite emergency power as well as their normal power source. Manual and check valves, of course, do not require actuation or control systems.

The containment isolation system is brought into service by one of three conditions: phase A isolation signal, phase B isolation signal, or containment ventilation isolation signal.

The automatically tripped isolation valves are actuated to the closed position by any of these isolation signals. The first of these signals is derived in conjunction with safety injection actuation, and trips the majority of the automatic isolation valves. These are valves in the so-called "nonessential" [***Note*** *- "Nonessential" process lines are defined as those, which do not increase the potential for damage to in-containment equipment when isolated. "Essential" process lines are those providing cooling water and seal water flow for the reactor coolant pumps. These services should not be interrupted unless absolutely necessary while the reactor coolant pumps are operating.*] process lines penetrating the containment. This is defined as "phase A" isolation and the trip valves are designated by the letter "T" in the isolation diagrams, Figures 5.2-1 through 5.2-29. This signal also initiates automatic seal water injection (See Section 6.5). The second, or "phase B", containment isolation signal is derived upon actuation of the containment spray system, and trips the automatic isolation valves in the so-called "essential"* process lines penetrating the containment. These trip valves are designated by the letter "P" in the isolation diagrams. Containment ventilation isolation represents closing of the three ventilation lines to the containment and will be automatically activated by high containment radioactivity, a phase A isolation signal, or automatic containment spray (and associated phase B) actuation; see Section 5.3 for further information on the containment, heating, cooling and ventilation system.

A manual containment isolation signal can be generated from the control room for either phase A or phase B isolation. These signals perform the same functions as the automatically derived signals. The containment ventilation isolation signal can be manually activated by a manual safety injection signal, a manual phase A containment isolation signal, or a manual containment spray signal.

Nonautomatic isolation valves, i.e., remote stop valves and manual valves, are used in lines, which must remain in service, at least for a time, following an accident. These are closed manually if and when the lines are taken out of service.

Standard closing times available with commercial valve models are adequate for the sizes of containment isolation valves used. Valves equipped with air-diaphragm operators generally close in approximately 2 sec. The typical closing time available for large motor-operated gate valves is 10 sec. Closing times of greater than 10 seconds are permitted on a case by case basis if properly justified by an individual valve evaluation.

The large butterfly valves used to isolate the containment ventilation purge ducts are equipped with air piston operators and spring returns capable of closing the valves. The butterfly valves used to isolate the 10-in. pressure relief line are equipped with air piston operators each with a separate accumulator air supply on each valve capable of closing the valves. These valves all fail to the closed position on loss of control signal or instrument air. Allowable closure time for these valves is less than or equal to 3 seconds.

5.2.5 Valve Operability

All containment isolation valves, actuators, and controls are located so as to be protected against missiles that could be generated as the result of a loss-of-coolant accident. Only valves so protected are considered to qualify as containment isolation valves.

Only isolation valves located inside containment are subject to the high-pressure, high-temperature, steam-laden atmosphere resulting from an accident. Operability of these valves in the accident environment is ensured by proper design, construction, and installation, as reflected by the following considerations:

1. All components in the valve installation, including valve bodies, trim and moving parts, actuators, instrument air and control and power wiring, are constructed of materials sufficiently temperature resistant to be unaffected by the accident environment. Special attention is given to electrical insulation, air operator diaphragms and stem packing material.

2. In addition to normal pressures, the valves are designed to with-stand maximum pressure differentials in the reverse direction imposed by the accident conditions.

This criterion is particularly applicable to the butterfly-type isolation valves used in the containment purge lines. Valve actuators are installed on these butterfly valves and travel is limited to a maximum of 60 degrees to ensure that the valves will be able to close against the maximum calculated design-basis accident pressure of 47 psig. An adjustable position setting on the actuators allows the valves to be opened to a full 90-degree position when containment integrity is not required.

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 1 of 10)

Item No.	Penetration and System	Dia-gram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Testing/ Sealing Method[1]	Used After Accid.	Fluid Gas or Water	Penetration Hot/Cold	Remarks
1	Pressurizer relief tank to gas analyzer RCS	5.2-1	549 548	Globe Globe	Air Air	Yes Yes	Op/Cl Op/Cl	Op/Cl Op/Cl	Closed Closed	FC FC	T T	Water (A) Water (A)	No	G	Hot	
2.	Pressurizer relief tank N_2 supply tank RCS	5.2-1	518 3418 3419 4136	Check Globe Globe Dia.	- Sole. Sole. Manual	No No No No	Closed Open Open Closed/OI	Closed Open Open Closed/OI	Closed Op/Cl Op/Cl Op/Cl	- FC FC -	- - - -	- - - -	Yes/No	G	Cold	
3.	Pressurizer relief tank makeup - RCS	5.2-1	552 519	Dia Dia	Air Air	Yes Yes	Closed Closed	Closed Closed	Closed Closed	FC FC	T T	Water (A) Water (A)	No	W	Cold	
4.	Residual heat removal return - ACS/SIS	5.2-2	$741A_{1A}$ 744_{1A}	Check DDV	- Motor	No Yes	Closed Open	Open Open	Open Op/CL	- FAI	- -	RHR Nit (M)	Yes	W	Hot	May be closed depending on accident condition
5.	Resid. Heat removal loop to - S.I.pumps - ACS/SIS	5.2-2	888A 888B	DDV DDV	Motor Motor	Yes Yes	Closed/OI Closed/OI	Closed Closed	Op/Cl Op/Cl	FAI FAI	- -	Nit (M) Nit (M)	Yes	W	Hot**	
	To sampling system ACS/SS	5.2-2	958 959 990D	Globe Globe Globe	Motor Motor Manual	No No No	Closed/OI Closed/OI LC/OI	Closed Closed LC	Closed Closed LC	FAI FAI -	- - -	Nit (M) Nit (M) Nit (M)	Yes Yes No	W W -	Hot -	May be used during shutdown and after accident
	RHR pump mini-flow line	5.2-2	1870 743	Globe Globe	Motor Motor	Yes Yes	Open Open	Open Open	Op/Cl Op/Cl	FAI FAI	- -	Nit (M) Nit (M)	Yes/No Yes/No	W W	Hot	

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 2 of 10)

Item No.	Penetration and System	Dia-gram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Testing/ Sealing Method$_1$	Used After Accid.	Fluid Gas or Water	Penetration Hot/Cold	Remarks
6.	Residual heat removal loop-out - ACS	5.2-2	732_{1A}	DDV	Manual	No	LC/OI	Open	Closed	-	-	Nit (M)	No	W	Hot	
7.	Containment sump recirculation - ACS/SIS	5.2-2	885A 885B	DDV_{2A} DDV_{2A}	Motor Motor	Yes Yes	Closed/OI Closed/OI	Closed Closed	$Closed_2$ $Closed_2$	FAI FAI	- -	RHR RHR	No_2	W	Cold	2. Normally closed but may be opened after accident if normal recirculation path from recirculation pump not available 2A. The upstream disc (nearest containment) of 885A and the downstream disc (RHR Loop side) of 885B have a 3/16" hole to prevent pressure locking
8.	Letdown line - CVCS	5.2-3	201 202	Globe Globe	Air Air	Yes Yes	Open Open	Open Open	Closed Closed	FC FC	T T	Water (A) Water (A)	No	W	Hot	
9.	Charging line - CVCS	5.2-3	205 226 227	Gate Globe Globe	Motor Motor Motor	No No No	Open Open Closed/OI	Open Open Closed	Op/Cl Op/Cl Op/Cl	FAI FAI FAI	- - -	Water (M) Water (M) Water (M)	Yes* Yes* Yes*	W	Cold	* May be used depending on accident.
10.	Reactor coolant pump seal-water supply lines (4) - CVCS	5.2-4	250ABCD 4925, 4926, 4927, 4928	Globe Globe	Motor Motor	No No	Open Open	Open Open	Op/Cl Op/Cl	FAI FAI	- -	Water (M) Water (M)	Yes_3 Yes_3	W W	Cold Cold	3. Manual isolation if and when pumps are stopped.
11.	Reactor coolant pump seal water return - CVCS	5.2-4	222	DDV	Motor	Yes	Open	Open	Closed	FAI	P	Water (A)	No	W	Cold	
12.	Reactor coolant sample line - SS	5.2-5	956E 956F	Globe Globe	Motor Motor	Yes Yes	Op/Cl Op/Cl	Op/Cl Op/Cl	Closed Closed	FAI FAI	T T	Water (A) Water (A)	Yes_4	W	Hot	4. Used to take postaccident RCS samples
13.	Fuel transfer tube - FHS	5.2-5	A	Blind flange	-	No	Closed	-	-	-	-	Air_5	No	W	Cold	Flange is double gasketed in refuel-ing canal (missile protected). 5. Normally seal with air (WCPPS)

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 3 of 10)

Item No.	Penetration and System	Dia-gram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Posit. Isolation Trip	Testing/ Sealing Method[1]	Used After Accid.	Fluid Gas or Water	Penetration Hot/Cold	Remarks
14.	Containment spray headers (2) - SIS	5.2-6	869A,B	DDV	Motor	No	Open	Op/Cl	Op/Cl	FAI	-	Water (M)	Yes	W	Cold	
			867A,B	Check	-	No	Closed	Closed	Op/Cl	-	-	-				
			878A	Globe	Manual	No	LC/OI	Closed	Closed	-	-	-				
15.	Safety injection headers (2) - SIS	5.2-7	850A	DDV	Motor	No	Open	Open	Op/Cl	FAI	-	Water(M)	Yes	W	Hot**	
			851A	DDV	Motor	Yes	Open	Open	Op/Cl	FAI	-	Water(M)	Yes	W	Hot**	
			851B	DDV	Motor	Yes	Open	Open	Op/Cl	FAI	-	Water(M)	Yes	W	Hot**	
			850B	DDV	Motor	No	Open	Open	Op/Cl	FAI	-	Water(M)	Yes	W	Hot**	
16.	Safety injection test line - SIS	5.2-7	859A	Globe	Manual	No.	LC/OI	Closed	Closed	-	-	Water (A)	No	W	Cold	
			859C	Globe	Manual	No	LC/OI	Closed	Closed	-	-	Water (A)				
17.	Accumulator/ OPS N_2 supply - SIS	5.2-8	4312	Check	-	No	Closed	Closed	Closed[6]	-	-	-	No	G	Cold	6. Could be opened depending on type of accident
			863	Globe	Air	Yes	Op/Cl	Op/Cl	Closed[6]	FC	T	-	No			
18.	Accumulator sample - SS	5.2-8	956G	Globe	Air	Yes	Op/Cl	Op/Cl	Closed	FC	T	Water (A)	No	W	Cold	Valves A and B opened intermittently to take sample
			956H	Globe	Air	Yes	Op/Cl	Op/Cl	Closed	FC	T	Water (A)				
19.	Primary system vent header and N_2 supply line - WDS	5.2-9	1786	Dia	Air	Yes	Open	Closed	Closed	FC	T	Water (A)	No	G	Hot	
			1787	Dia	Air	Yes	Open	Closed	Closed	FC	T	Water (A)				
			3416	Globe	Sole.	No	Open	Open	Op/Cl	FC	-	-	Yes/No	G	Hot	
			3417	Globe	Sole.	No	Open	Open	Op/Cl	FC	-	-				
			5459	Dia	Manual	No	Closed/OI	Closed	Op/Cl	-	-	-				
			1616	Check	-	No	Closed	Closed	Op/Cl	-	-	-				
20.	Reactor coolant drain tank to gas analyzer - WDS	5.2-9	1788	Dia	Air	Yes	Op/Cl	Op/Cl	Closed	FC	T	Water (A)	No	G	Hot	Valves opened intermittently
			1789	Dia	Air	Yes	Op/Cl	Op/Cl	Closed	FC	T	Water (A)				
21.	RCDT pump discharge – WDS	5.2-9	1702	Dia	Air	Yes	Open	Op/Cl	Closed	FC	T	Water (A)	No	W	Cold	Valves open intermittently
			1705	Dia	Air	Yes	Open	Op/Cl	Closed	FC	T	Water (A)				

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 4 of 10)

Item No.	Penetration and System	Dia-gram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Testing/ Sealing Method$_1$	Used After Accid.	Fluid Gas or Water	Penetration Hot/Cold	Remarks
22.	Reactor coolant pump cooling water in - ACS	5.2-10	797	DDV	Motor	Yes	Open	Op/Cl	Closed	FAI	P	Water (A)	No_7	W	Cold	7. Could be used depending on the type of accident
23.	Reactor coolant pump water out (6") - ACS	5.2-10	784	DDV	Motor	Yes	Open	Op/Cl	Closed	FAI	P	Water (A)	No_8	W	Cold	8. Could be used depending on the type of accident
24.	Reactor coolant pump water out (3") - ACS	5.2-10	FCV-625	DDV	Motor	Yes	Open	Op/Cl	Closed	FAI	P	Water (A)	No_9	W	Cold	9. Could be used depending on the type of accident
25.	Resid. Heat exch. Cooling water in - ACS	5.2-11	CS	-	-	-	-	-	-	-	-	-	Yes	W	Hot	Residual heat exchanger and associated component cooling lines are a missile protected closed system
			CS	-	-	-	-	-	-	-	-	-	Yes	W	Hot	Component cooling system closed
26.	Resid. Heat exch. Cooling water return - ACS	5.2-11	$822A_{1B}$	Gate	Motor	Yes	Closed	Open	Open	FAI	-	-	Yes	W	Cold	
			$822B_{1B}$	Gate	Motor	Yes	Closed	Open	Open	FAI	-	-	Yes	W	Cold	
			CS	-	-	-	-	-	-	-	-	-	Yes	W	Cold	Component cooling system closed
27.	Recir. Pump cooling water supply - ACS	5.2-12	$753H_{1B}$	Gate	Manual	No	Open	Open	Op/Cl	-	-	-	Yes	W	Cold	May be closed depending on accident condition
			CS	-	-	-	-	-	-	-	-	-	Yes	W	Cold	Component cooling system closed
28.	Recir. Pump cooling heater return - ACS	5.2-12	$753G_{1B}$	Gate	Manual	No	Open	Open	Op/Cl	-	-	-	Yes	W	Cold	May be closed depending on accident condition
			CS	-	-	-	-	-	-	-	-	-	Yes	W	Cold	Component cooling system closed
29.	Excess letdown heat exchanger cooling water in – ACS	5.2-13	791	Dia	Air	Yes	Closed	Closed	Closed	FC	T	Water (A)	No	W	Cold	
			798	Dia	Air	Yes	Closed	Closed	Closed	FC	T	Water (A)				

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 5 of 10)

Item No.	Penetration and System	Dia-gram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Testing/ Sealing Method$_1$	Used After Accid.	Fluid Gas or Water	Penetration Hot/Cold	Remarks
30.	Excess letdown heat exchanger cooling water out - ACS	5.2-13	796 793	Globe Dia	Air Air	Yes Yes	Closed Closed	Closed Closed	Closed Closed	FC FC	T T	Water (A) Water (A)	No	W	Cold	
31.	Containment sump pump discharge – WDS	5.2-13	1728 1723	Dia Dia	Air Air	Yes Yes	Open Open	Open Open	Closed Closed	FC FC	T T	Water (A) Water (A)	No	W	Cold	
31a.	Sampling system return - WDS	5.2-13	5132 4399	Globe Globe	Motor Motor	Yes Yes	Closed Closed	Closed Closed	Cl/Op Cl/Op	FAI FAI	T T	Water (A) Water (A)	No/Yes$_{10}$ No/Yes$_{10}$	W W	Cold Cold	10. Can be used to return highly radioactive water to containment after post-accident analysis of sampling system.
32.	Containment air sample in - rad. mon.	5.2-14	PCV-1234 PCV-1235	Dia Dia	Air Air	Yes Yes	Open Open	Open Open	Closed Closed	FC FC	T T	Air (A) Air (A)	No$_{11}$ No$_{11}$	G G	Cold Cold	11. May be opened for air sampling following accident when the containment pressure is below 5 psig
33.	Containment air sample out - rad. mon.	5.2-14	PCV-1236 PCV-1237	Dia. Dia.	Air Air	Yes Yes	Open Open	Open Open	Closed Closed	FC FC	T T	Air (A) Air (A)	No$_{12}$ No$_{12}$	G G	Cold Cold	12. May be opened for air sampling following accident when the containment pressure is below 5 psig
34.	Air ejector discharge to containment sec sys	5.2-14	PCV-1229 PCV-1230	Globe Globe	Air Air	Yes Yes	Closed Closed	Closed Closed	Closed Closed	FC FC	T T	Air (A) Air (A)	No No	G G	Cold Cold	
35.	Main steam headers$_{13}$	-	CS	-	-	-	-	-	-	-	-	-	-	-	Hot	Steam generators 13. (four penetrations)

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 6 of 10)

Item No.	Penetration and System	Dia-gram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Testing/ Sealing $Method_1$	Used After Acid.	Fluid Gas or Water	Penetration Hot/Cold	Remarks
36.	Main feedwater headers	-	CS	-	-	-	-	-	-	-	-	-	-	-	Hot	Steam generators (four penetrations)
37.	Steam generator blowdown/sample sec. sys.	5.2-15	PCV1214 PCV1215 PCV1216 PCV1217	Globe	Air	Yes	Open	Op/Cl	Closed	FC	T	Water (A)	No	W	Hot	*(four penetrations)
			PCV1214A PCV1215A PCV1216A PCV1217A	Globe	Air	Yes	Open	Op/Cl	Closed	FC	T	Water (A)	No	W	Hot	
38.	S.G. blowdown sample															System deleted
39.	Ventilation system water cooling water in – SWS_{14}	5.2-16	SWN-41 SWN-42 SWN-43	BV Relief Gate(2) Globe(3)	Motor - Manual	No No No	Open Closed LC/OI	Open Closed Closed	Op/Cl Closed Op/Cl	FAI - -	- - -	SWS SWS SWS	Yes - Yes	W - -	Cold - -	Fan cooler units - missile protected, closed system 14. (five penetrations)
40.	Ventilation system motor cooling water out -	5.2-16	SWN-44	BV	Motor	No	Open	Open	Op/Cl	FAI	-	SWS	Yes	W	Cold	Fan cooler units - missile protected, closed system
	SWS_{15}	5–2–16	SWN-51	Globe	Motor	No	Open	Open	Op/Cl	FAI	-	SWS	Yes	W	Cold	15. (Five penetrations)
40a.	Ventilation system motor cooling water out - SWS_{16}	5–2–16	SWN-71	Globe	Motor	No	Open	Open	Op/Cl	FAI	-	SWS	Yes	W	Cold	16. Five penetrations
41.	Service air	5.2-17	SA-24 SA-24-1	Dia Dia	Manual Manual	No No	LC/OI LC/OI	LC LC	LC LC	- -	- -	Water (A) Water (A)	No	G	Cold	
42.	Not assigned															

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 7 of 10)

Item No.	Penetration and System	Dia-gram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Testing/ Sealing Method$_1$	Used After Accid.	Fluid Gas or Water	Penetration Hot/Cold	Remarks
43.	Weld channel pressurization air supply (PPS)$_{17}$	5.2-17	PCV1111-1$_{1B}$ PCV1111-2$_{1B}$ CS	Ball Ball -	Manual - -	No - -	Open	Open	Op/Cl	- -	- -	- -	Yes -	G -	Cold -	17. Two penetrations penetration press system
44.	Spare	5.2-17	580A 580B	Needle Needle	Manual Manual	No No	LC/OI LC/OI	LC LC	LC LC	- -	- -	- -	No	G	Cold	Penetration capped inside containment and outside containment downstream of valve 580B
45.	Auxiliary steam supply	5.2-18	UH-43	DDV	Manual	No	LC/OI	Closed$_{18}$	LC	-	-	Water (A)	No	G	Hot	18. May be opened during shutdown for cont. heating
46.	Auxiliary steam supply condensate return	5.2-18	UH-44	DDV	Manual	No	LC/OI	Closed$_{19}$	LC	-	-	Water (A)	No	W	Hot	19. May be opened during shutdown for cont. heating
47.	City water	5.2-18	MW-17 MW-17-1	Gate Gate	Manual Manual	No No	LC/OI LC/OI	Closed$_{20}$ Closed$_{20}$	LC LC	- -	- -	Water (A) Water (A)	No	W	Cold	20. May be opened during shutdown for maintenance or fire protection purposes
48.	Purge supply duct in - vent. sys.	5.2-19	FCV-1170 FCV-1171	BV BV	Air Air	Yes Yes	Closed$_{21}$ Closed$_{21}$	Open Open	Closed Closed	FC FC	CVI CVI	Air (A) Air (A)	No	G	Cold	21. May be open for safety related purging, or to facilitate safety related surveillance or maintenance.
49.	Purge exhaust duct out - vent. sys.	5.2-19	FCV-1172 FCV-1173	BV BV	Air Air	Yes Yes	Closed$_{22}$ Closed$_{22}$	Open Open	Closed Closed	FC FC	CVI CVI	Air (A) Air (A)	No	G	Cold	22. May be open for safety related purging, or to facilitate safety related surveillance or maintenance.
50.	Containment pressure relief - vent	5.2-19	PCV-1190 PCV-1191 PCV-1192	BV BV BV	Air Air Air	Yes Yes Yes	Closed$_{23}$ Closed$_{23}$ Closed$_{23}$	Closed Closed Closed	Closed Closed Closed	FC FC FC	CVI CVI CVI	Air (A) Air (A) Air (A)	No	G	Cold	23. Opened intermittently for pressure relief.
51.	Recirculation pump discharge sample line	5.2-20	990A 990B	Globe Globe	Motor Motor	Yes Yes	Closed Closed	Closed Closed	Op/Cl Op/Cl	FAI FAI	T T	Nit (M) Nit (M)	No/Yes$_{24}$	W	Cold	24. Used periodically after accident to sample recirculation fluid.

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 8 of 10)

Item No.	Penetration and System	Dia-gram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Testing/ Sealing Method$_1$	Used After Accid.	Fluid Gas or Water	Penetration Hot/Cold	Remarks
52.	Pressurizer steam space sample line	5.2-20	956A 956B	Globe Globe	Air Air	Yes Yes	Op/Cl OP/Cl	Op/Cl OP/Cl	Closed Closed	FC FC	T T	Water (A) Water (A)	No/Yes$_{25}$ No/Yes$_{25}$	W W	Hot Hot	25. Could be used for taking postaccident samples.
53.	Pressurizer liquid space sample line	5.2-20	956C 956D	Globe Globe	Air Air	Yes Yes	Op/Cl Op/Cl	Op/Cl Op/Cl	Closed Closed	FC FC	T T	Water (A) Water (A)	No/Yes$_{26}$ No/Yes$_{26}$	W W	Hot Hot	26. Could be used for taking postaccident samples.
54. 55. 56.	Containment Pressure Instrumentation	5.2-21	1814A 1814B 1814C	Globe	Manual	No	LO	Open	Op/Cl	-	-	-	Yes	G	Cold	
			CS	-	-	-	-	-	-	-	-	-	-	-	-	
57.	Postaccident containment sampling system supply and return lines (7)	5.2-22	SOV-5018 SOV-5020 SOV-5022 SOV-5024	Globe Globe Globe Globe	Sole. Sole. Sole. Sole.	Yes	Closed	Closed	Both$_{26a}$	FC	-	-	Yes	G	Cold	26a. Isolation valves are opened intermittently after an accident.
			SOV-5019 SOV-5021 SOV-5023 SOV-5025	Globe Globe Globe Globe	Sole. Sole. Sole. Sole.	Yes	Closed	Closed	Both$_{26a}$	FC	-	-	Yes	G	Cold	
58.	Spare	5.2-23														
59.	Spare	5.2-24														
60.	Spare															
61.	Spare															
62.	Spare															
63.	Not assigned															

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 9 of 10)

Item No.	Penetration and System	Dia-gram	Valve No. or Closed System	Valve Type	Oper. Type	Posit. Indic. In Cont. Room	Normal Posit.	Posit. During Shutdown	Posit. After Accident	Posit. On Power Fail	Cont. Isolation Trip	Testing/ Sealing Method$_1$	Used After Accid.	Fluid Gas or Water	Penetration Hot/Cold	Remarks
64.	Instrument/air postaccident venting supply	5.2-25	IA-39	Check	-	No	Open	Open	Both	-	-	-	Yes/No$_{27}$	G	Cold	27. Could be used to resupply instrument air to containment post-accident.
			PCV1228	Dia	Air	Yes	Open	Open	Both	FC	T	-	Yes/No$_{27}$	G	Cold	
65.	Postaccident venting exhaust line	5.2-25	E-2	Dia	Air	No	Closed/OI	Closed	Both	FC	-	Air (A)	Yes/No$_{28}$	G	Cold	28. Could be used after accident if containment venting were deemed necessary.
			E-1	Dia	Air	No	Closed/OI	Closed	Both	FC	-	Air (A)	Yes/No$_{28}$	G	Cold	
			E-3	Dia	Air	No	Closed/OI	Closed	Both	FC	-	Air (A)	Yes/No$_{28}$	G	Cold	
			E-5	Dia	Air	No	Closed/OI	Closed	Both	FC	-	Air (A)	Yes/No$_{28}$	G	Cold	
66.	Deleted															
67.	Containment leak test air line$_{30}$	5.2-26	A	Blind Flange	-	No	Closed	Closed	Closed	-	-	-	No.	Gas	Cold	30. Two penetrations
			B	Blind Flange W/Test Conn.	-	No	Closed	Closed	Closed	-	-	-	No	Gas	Cold	
68.	Equipment access	-	CS	-	-	-	-	-	-	-	-	Air (A)	No	-	-	
69.	Personnel air lock (2)	5.2-27	85A, 95A$_{31}$	Ball	Interlk w/door	Yes	Closed	Closed	Closed	-	-	Air (A)	No	Gas	Cold	31. 85A & 95A may be open when 85B & 95B are closed. 85B & 95B may be open when 85A & 95A are closed.
			85B, 95B$_{31}$	Ball								Air (A)				
			85C, 95C	Spring check	-	No	Closed	Closed	Closed	-	-	-	No	Gas	Cold	
			85D, 95D	Spring check	-	No	Closed	Closed	Closed	-	-	-	No	Gas	Cold	
70.	Steam generator level, pressurizer level, and pressure pneumatic indication lines (4)	5.2-28	IIP-500	Globe	Manual	No	Closed/OI	Closed	Both	-	-	-	Yes$_{32}$	-	Cold	32. Depending on accident type
			IIP-501	Globe	Manual	No	Closed/OI	Closed	Both	-	-	-	Yes$_{32}$	-	Cold	
			IIP-502	Globe	Manual	No	Closed/OI	Closed	Both	-	-	-	Yes$_{32}$	-	Cold	
			IIP-503	Globe	Manual	No	Closed/OI	Closed	Both	-	-	-	Yes$_{32}$	-	Cold	
			IIP-504	Globe	Manual	No	Closed/OI	Closed	Both	-	-	-	Yes$_{32}$	-	Cold	
			IIP-505	Globe	Manual	No	Closed/OI	Closed	Both	-	-	-	Yes$_{32}$	-	Cold	
			IIP-506	Globe	Manual	No	Closed/OI	Closed	Both	-	-	-	Yes$_{32}$	-	Cold	
			IIP-507	Globe	Manual	No	Closed/OI	Closed	Both	-	-	-	Yes$_{32}$	-	Cold	

TABLE 5.2-1
Containment Piping Penetrations and Valving
(Sheet 10 of 10)

Key:

FAI	- fail as is	RCS	- reactor coolant system	Hot – insulated & cooled penetration
FC	- fail closed	ACS	- auxiliary coolant system	Cold – standard piping or equipment penetration
FO	- fail open	WDS	- waste disposal system	Hot** - insulated non cooled penetration
LC	- locked closed	SIS	- safety injection system	IVSWS -isolation valve seal water system
LO	- locked open	SS	- sampling system	WCPPS -weld channel pressurization system
BV	- butterfly valve	CVCS	- chemical and volume control system	RHR -residual heat removal system
DDV	- double disk gate valve	Vent	- ventilation system	
Dia	- diaphragm valve	SWS	- service water system	
T	- containment isolation signal - phase A	FH	- fuel handling	
P	- containment isolation signal - phase B	PPS	- penetration pressurization system	
A	- automatic	CVI	- containment ventilation isolation signal	
M	- manual	CS	- closed system	
Op/Cl	- open/closed	Nit	- nitrogen	
OI	- may be opened intermittently to support plant operations			

Notes:

1. Sealing Methods and Type C Testing:
   - For valves sealed by IVSWS water (designated “Water (A)” or “Water (M)”), minimum Type C test pressure is 52 psig.
   - For valves sealed by IVSWS nitrogen (designated “Nit (M)”), minimum Type C test pressure is 47 psig.
   - For valves sealed by WCPPS (designated “Air (A)”), minimum Type C test pressure is 47 psig.
   - For valves sealed by RHR system fluid (designated “RHR”), minimum Type C test pressure is 52 psig (valves 741A,885A,885B).
   - For valves sealed by service water system (designated “SWS”), minimum Type C test pressure is 52 psig (valve series SWN-41, SWN-42, SWN-43, SWN-44, SWN-51, SWN-71). Either the “A” or “B” valve(s) may serve as the required containment isolation valve(s) for the SWN-41, SWN-44 and SWN-71 series. Designation of the “B” valve(s) in the SWN-44 series requires the codesignation of the SWN-51 valves associated with the penetration(s) as an additional required containment isolation valve(s) (see Figure 5.2-16).
   - For all other isolation valves not sealed by a system, gas (ie. Nitrogen or air) is the test medium at a minimum Type C pressure of 47 psig.

1A. These valves testable only when at cold shutdown (741A, 744, 732).

1B. These valves are excluded from Type C testing per License Amendment N0. 63, dated August 28, 1980 (822A, 822B, 753G, 753H, PCV-1111-1, PCV-1111-2).

5.2 FIGURES

5.3 CONTAINMENT HEATING, COOLING AND VENTILATION SYSTEM

5.3.1 Design Basis

5.3.1.1 Performance Objectives

The containment heating, cooling and ventilation systems are designed to accomplish the following:

1. Remove the normal heat loss from all equipment and piping in the reactor containment during plant operation and to maintain a normal ambient temperature of 130°F or less.

2. Provide sufficient air circulation and filtering of iodine throughout all containment areas to permit safe and continuous access to the reactor containment within two hours after reactor shutdown assuming defects exist in 1-percent of the fuel rods.

3. Provide for positive circulation of air across the refueling water surface to assure personnel access and safety during shutdown.

4. Provide containment heating, if required, to assure a minimum containment ambient temperature of 50°F before the reactor is taken above the cold shutdown condition.

5. Provide for purging of the containment vessel to the plant vent for dispersion to the environment. The rate of release does not permit offsite dose to exceed Offsite Dose Calculation Manual (ODCM).

6. Provide for depressurization of the containment vessel following an accident. The postaccident design and operating criteria are detailed in Section 6.4.

7. Provide for continuous pressure relief via an exhaust system.

In order to accomplish these objectives the following systems are provided:

1. Containment recirculation cooling system
2. Control rod drive mechanism cooling system
3. Containment purge and pressure relief system
4. Containment auxiliary charcoal filter system
5. Steam heating system

5.3.1.2 Design Characteristics-Sizing

The design characteristics of the equipment required in the containment for cooling, filtration and heating to handle the normal thermal and air cleaning loads during normal plant operation are presented in Table 5.3-1. In certain cases where engineered safeguards functions also are served by the equipment, component sizing is determined from the heavier duty specifications associated with the design basis accident detailed further in Section 6.4.

5.3.2 System Design

5.3.2.1 Piping and Instrumentation Diagram

The containment ventilation, purging, and recirculation cooling and filtration systems flow diagram is shown in Plant Drawing 9321-4022 [Formerly UFSAR Figure 5.3-1]. The containment ventilation systems and main plant vent are designed as Class I structures.

5.3.2.2 Containment Cooling and Ventilation System

Air recirculation cooling during normal operation is accomplished using air handling units discharged into a common header ductwork distribution system to ensure adequate flow of cooled air throughout the containment. The cooling coils in each air handling unit transfer up to 61.7×10^6 Btu/hr in the event of an accident when supplied with approximately 1600 gpm cooling water at 95°F inlet temperature and steam-air flowrate of 64,500 cfm.

Each air-handling unit consists of the following equipment arranged so that, during normal and accident operation, air flows through the unit in the following sequence: cooling coils, moisture separators (demisters), centrifugal fan with direct-drive motor, and distribution header. The fans and motors of these units are equipped with vibration sensors to detect abnormal operating conditions in the early stages of the disturbance. The normal air flow rate per air-handling unit is approximately 70,000 cfm. Section 6.4.2 provides additional information on the operation of this system.

The following additional systems supplement the main containment recirculation system:

1. Control rod drive cooling system consisting of fans and ductwork to circulate air through the control rod drive mechanism shroud and discharge it to the main containment volume. Four direct driven axial flow fans are provided for use. There are two power supplies for each fan.

2. Two unit steam heaters are located in containment to provide additional area heating as required. The containment purge supply is also provided with steam pre-heating to supplement containment heating as required.

5.3.2.3 Containment Purge System

The containment purge system is independent of the primary auxiliary building exhaust system, (except for the common exhaust fans) and includes provisions for both supply and exhaust air. The supply system includes roughing filters, heating coils, fan, supply penetration with two butterfly valves for bubble tight shutoff, and a purge supply distribution header inside containment. The exhaust system includes exhaust penetration with two butterfly valves identical to those above, exhaust ductwork, filter bank with roughing, HEPA and charcoal filters, fans and exhaust vent. The purge supply and exhaust flow rates are nominally 23,000 cfm and 25,000 cfm respectively. The quick closing purge isolation valves close upon receipt of an accident signal. Allowable closure times for these valves are specified in Section 5.2.4.

During power operation, the purge system is routinely not operated. Prior to purging the containment air, particulate and gas monitor indications of the closed containment activity levels will be used to guide routine releases from the containment. During power operation, the containment air particulate and gas monitor indications will help determine desirability of using

either one or both of two auxiliary particulate and charcoal filter units installed in the containment primarily for preaccess cleanup.

When containment purging is in progress for access following reactor shutdown, releases from the plant vent are continuously monitored with a gas monitor, as described in Section 11.2.

5.3.2.4 Purge System Isolation Valves

The purge supply and exhaust duct butterfly valves, both inside and outside the containment, are normally closed during power operation. They may be opened for safety related reasons, i.e., pressure control or to facilitate safety related surveillance or maintenance. The opening angle is limited during operation so that the valves can close against a differential pressure (see Section 5.2.5). The spaces between the closed valves are pressurized with air by the weld channel and penetration pressurization system. The valves are designed for rapid automatic closing by the containment isolation signal (derived from any safety injection signal), upon a signal of high activity level within the containment in the event of a radioactivity release when the purge line is open, or upon a manually initiated signal. To ensure optimum sealing of the resilient valve seats, the two valves located outside containment are enclosed and a minimum ambient temperature is maintained.

5.3.2.5 Containment Pressure Relief Line

The normal pressure changes in the containment during reactor power operation, and during plant cooldown if the containment purge system is not operating, will be handled by the containment pressure relief line. This line is equipped with three quick-closing butterfly type isolation valves, one inside and two outside the containment. The valves will be automatically actuated to the closed position by the containment isolation signal, by a containment high radioactivity signal, or upon a manually initiated signal. Each of these air operated valves is equipped with an accumulator to assure each can close even if the air supply is lost. The two intra-valve spaces are pressurized with air by the Weld Channel and Penetration Pressurization System when the valves are closed. The pressure relief line discharges to the plant vent. The opening angle of the pressure relief valves is limited during operation so the valves can close against a differential pressure.

5.3.2.6 Containment Purge and Pressure Relief Isolation Reset

Opening of the purge and pressure relief isolation valves following an isolation signal requires deliberate operator action by resetting all isolation signals and depressing both Containment Ventilation Isolation reset push buttons. Further, in order to reset the Containment Ventilation Isolation signal for Train B, the control switches for the purge and pressure relief isolation valves in Train B must first be placed in the closed position. In addition, guard plates are placed over the reset buttons. These three features preclude the possibility of inadvertently opening these valves.

REFERENCES FOR SECTION 5.3

1. Deleted

TABLE 5.3-1 (Sheet 1 of 2)
Containment Cooling and Ventilation System - Principal Component Data Summary

System	Units Installed	Unit Capacity	Units Required for Normal Operation
Containment cooling and recirculation			
Demister	5	72,500 cfm	5_2
Cooling coils - normal	5	2.20×10^6 Btu/hr_1	5_2
Cooling coils - DBA	5	61.7×10^6 Btu/hr_3	
Fans	5	72,500 cfm	5_2
Fan pressure	-	7.21-in. H_2O (Note 7)	
Fan motors (440 V, 3-phase)	5	350 hp	5_2
Control rod drive mechanism cooling			
Fans, standard conditions	4	15,000 cfm	3
Fan pressure	-	5.5-in. H_2O	
Fan motors	4	30 hp	3
Reactor compartment cooling			
Part of containment recirculation system	-	12,000 cfm	
Refueling canal air sweep			
Part of containment recirculation system	-	17,500 cfm	
Purge supply	1	23,000 cfm_6	Optional
Fan pressure	-	2.5-in. H_20	
Fan motors	1	40 hp	
25 psig steam preheat coils	1 set	3×10^6 Btu/hr	Optional
Air filters, roughing	-	-	1

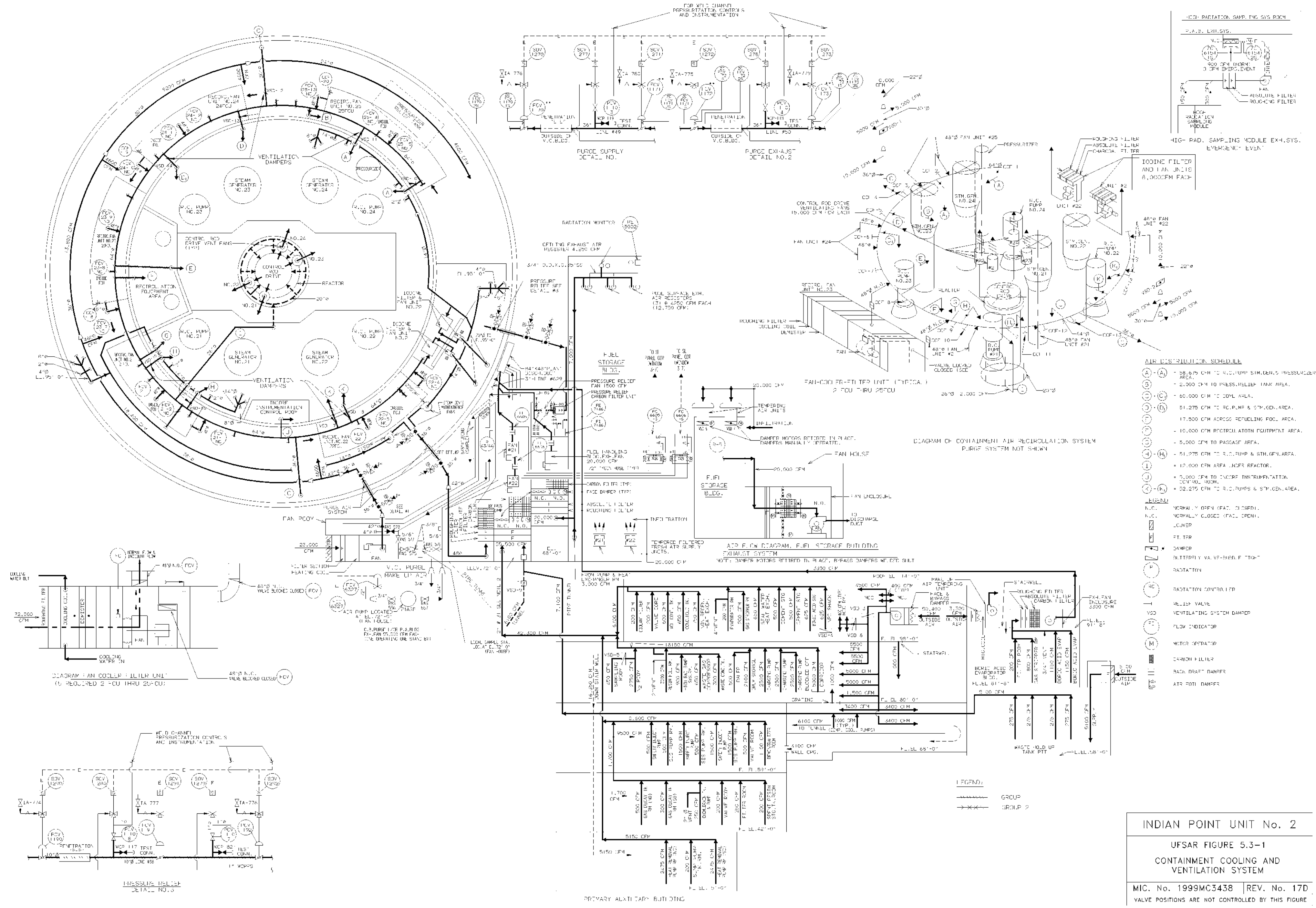
TABLE 5.3-1 (Sheet 2 of 2)
Containment Cooling and Ventilation System - Principal Component Data Summary

System	Units Installed	Unit Capacity	Units Required for Normal Operation
Purge exhaust			
Fans, standard conditions	2[4]	55,500 cfm[6]	Optional
Fan pressure	-	10.3-in. H_2O	Optional
Fan motors	2	125 hp	Optional
Plenums	1	-[5]	Optional
HEPA filters	-	-	Optional
HECA filters (charcoal adsorbers)	1	-	Optional
Containment auxiliary charcoal filter			
Fans, standard conditions	2	8,000 cfm	Optional
Fan pressure	-	5.0-in. H_20	Optional
Fan motors	2	10 hp	Optional
Filters and charcoal filters; roughing, HEPA	2	8,000 cfm	Optional
Steam heating			
Heaters, 25 psig steam	2	400,000 Btu/hr each	Optional

Notes:
1. This value reflects the increase in air side flow rate due to removal of the original plant HEPA Filters.
2. Depends on time of year and containment atmospheric temperature.
3. Based on minimum assumed performance at 271°F containment temp 95°F Service Water temp, 1600 gpm Service Water flow and 64,500 cfm air flow rate.
4. The two exhaust fans are used interchangeably or as backup for:
    1. Ventilation of primary auxiliary building.
    2. Containment building purge system.
5. Purge (25,000 cfm) and primary auxiliary building exhaust (55,500 cfm) are fed into a common plenum.
6. Purge supply (23,000 cfm) and purge exhaust (25,000 cfm) are the nominal, as built, flow rates for the purge system (± 10%).
7. At 72,500 cfm flow rate.

5.3 FIGURES

Figure No.	Title
Figure 5.3-1	Containment Cooling and Ventilation System [Replaced with Plant Drawing 9321-4022]



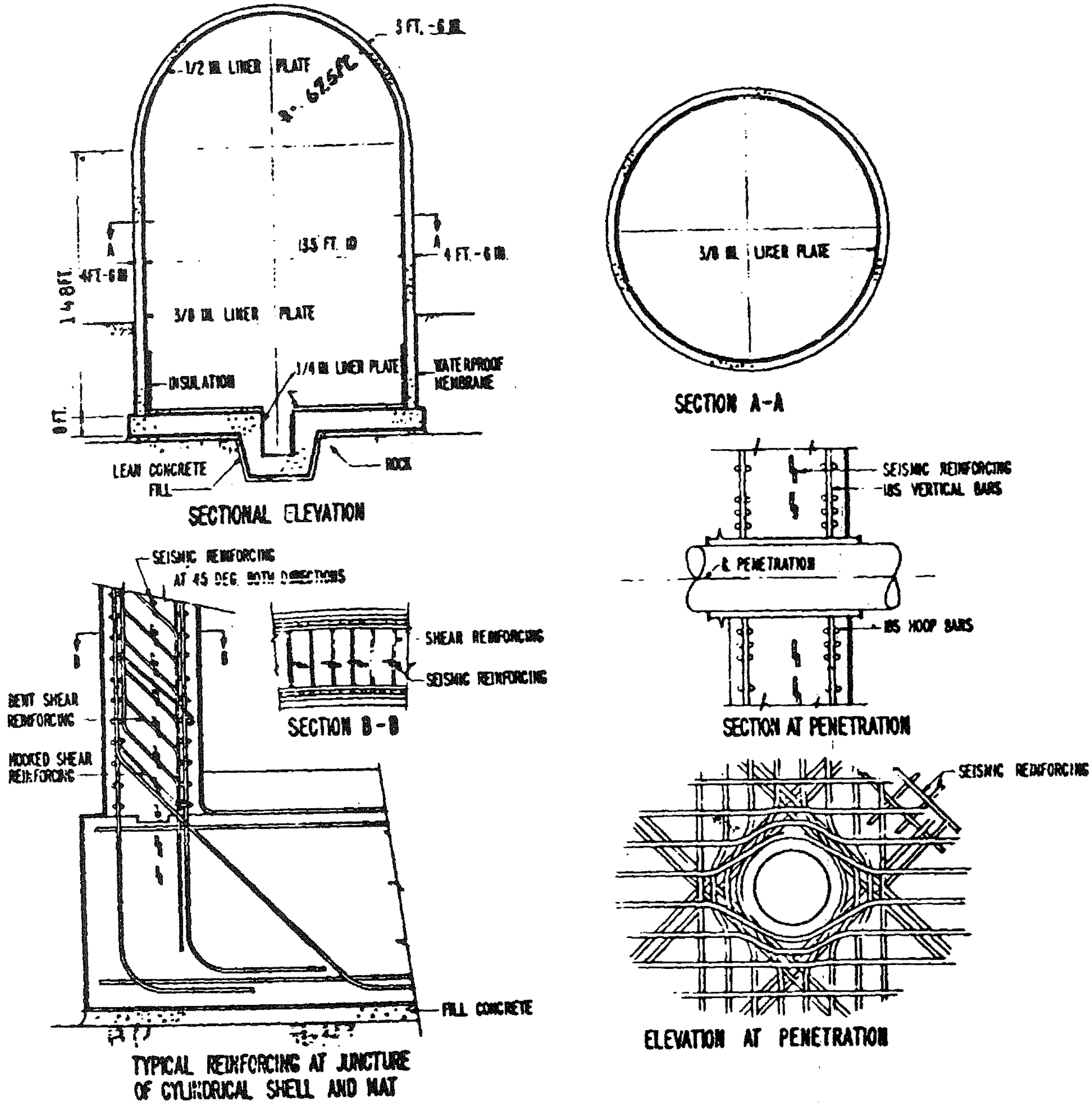
INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.3-1

CONTAINMENT COOLING AND VENTILATION SYSTEM

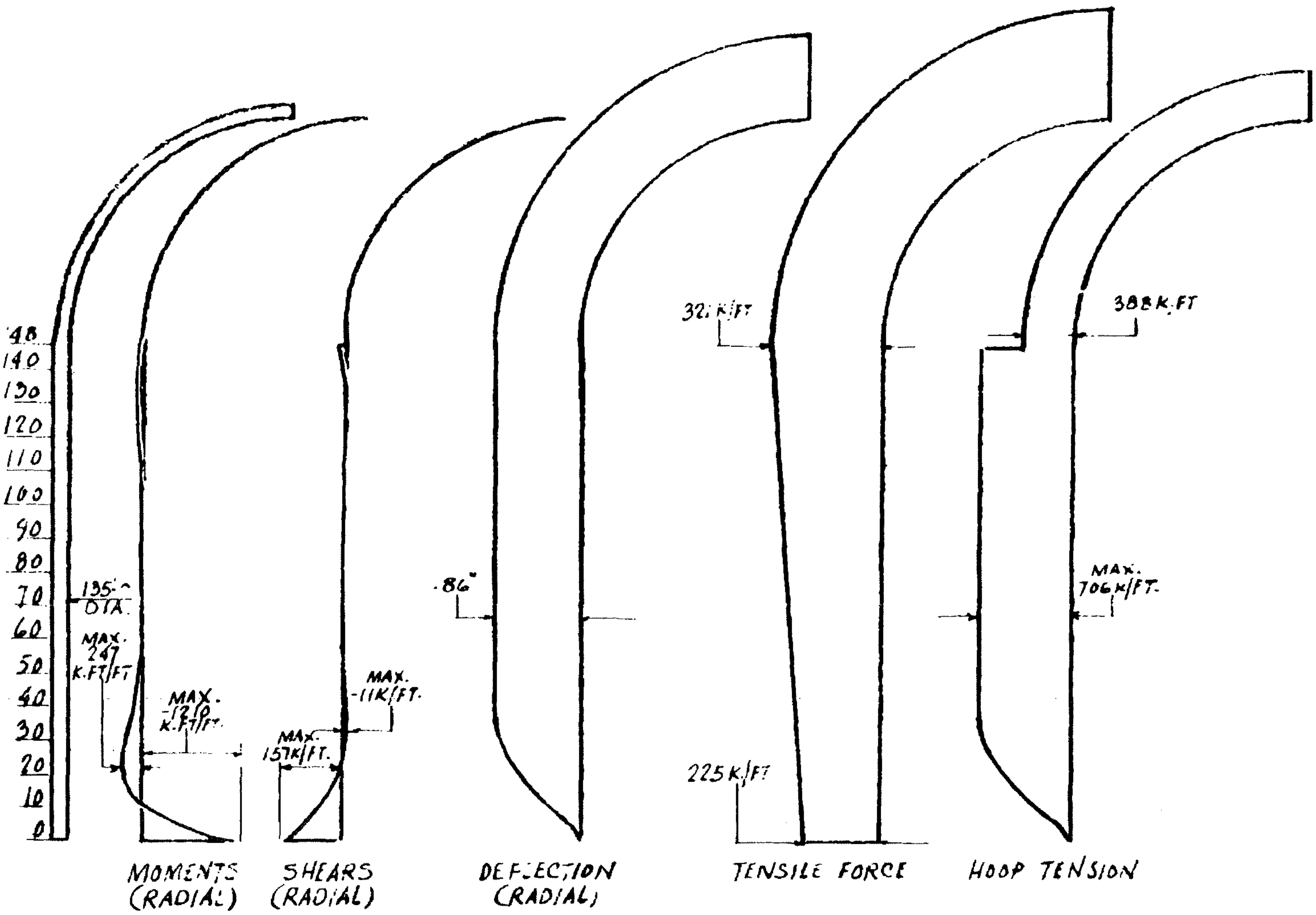
MIC. No. 1999MC3438 | REV. No. 17D

VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE



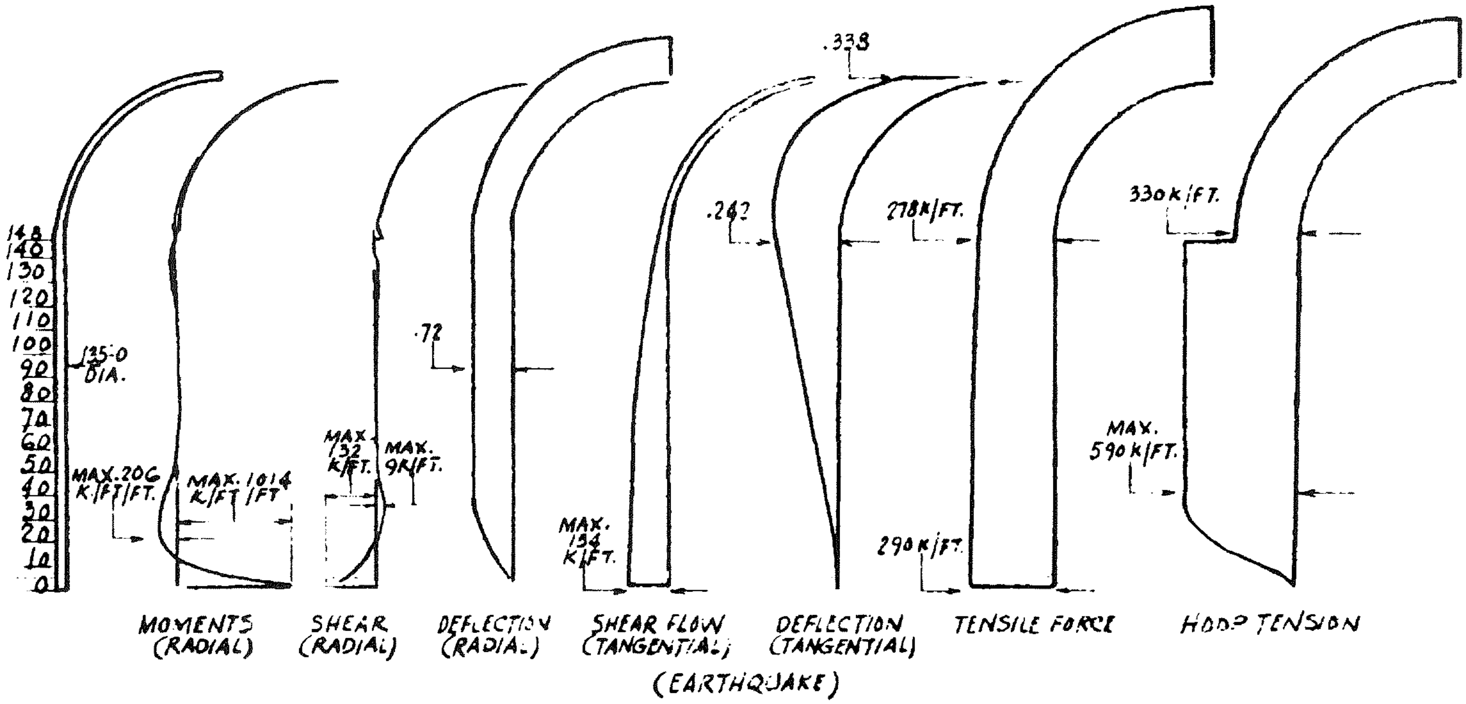
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-1 CONTAINMENT STRUCTURE	
MIC. No. 1999MC3746	REV. No. 17A

$C = 1.0D \pm 0.05D + 1.5P + 1.0(T+TL)$



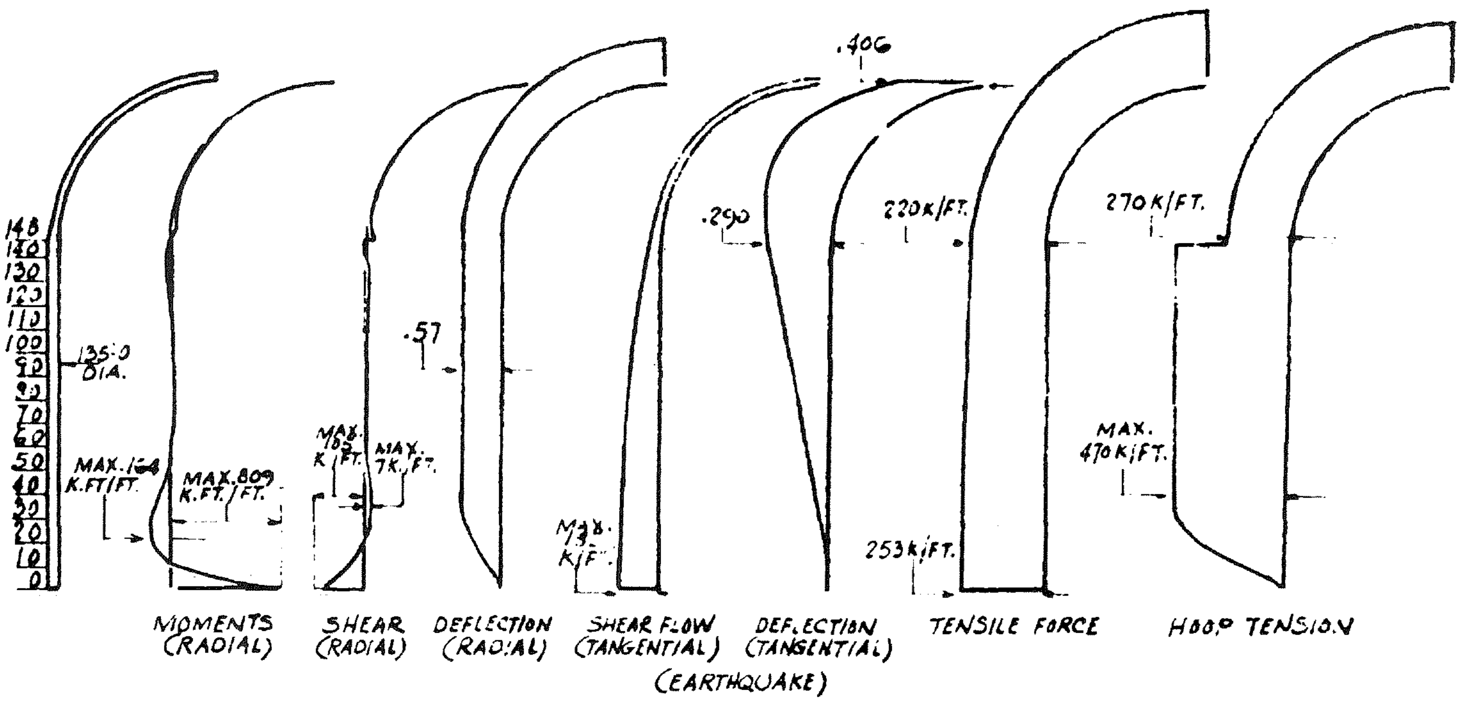
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-11	
CYLINDER AND DOME-LOAD CONDITION (A) - 1.5P	
MIC. No. 1999MC3750	REV. No. 17A

$C = 1.0D \pm 0.05D + 1.25P + 1.0(T' + T_L') + 1.25E$



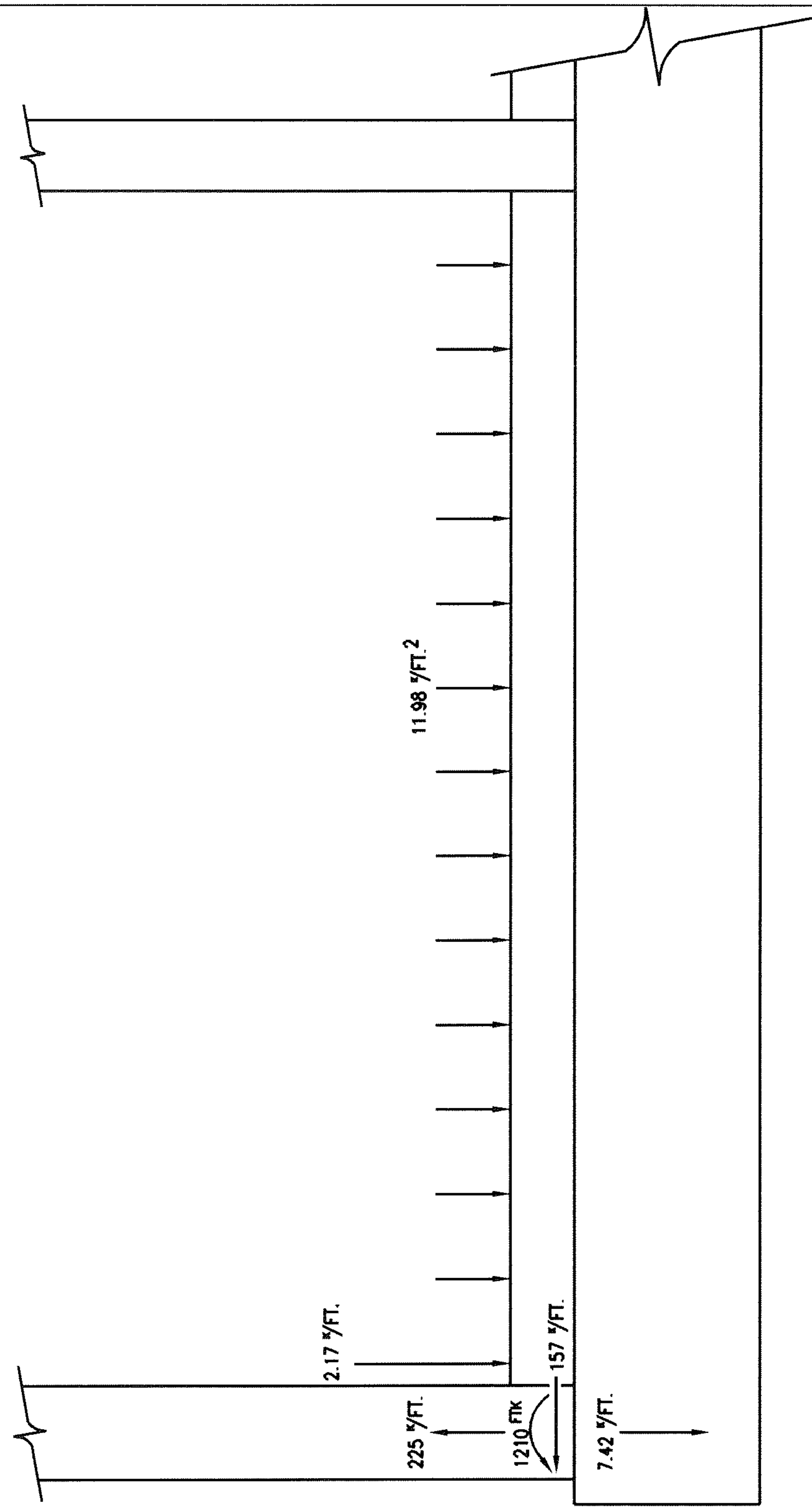
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-12 CYLINDER AND DOME-LOAD CONDITION (B) - 1.25P	
MIC. No. 1999MC3751	REV. No. 17A

$C = 1.0D \pm 0.05D + 1.0P + 1.0(T'' + TL'') + 1.0E'$



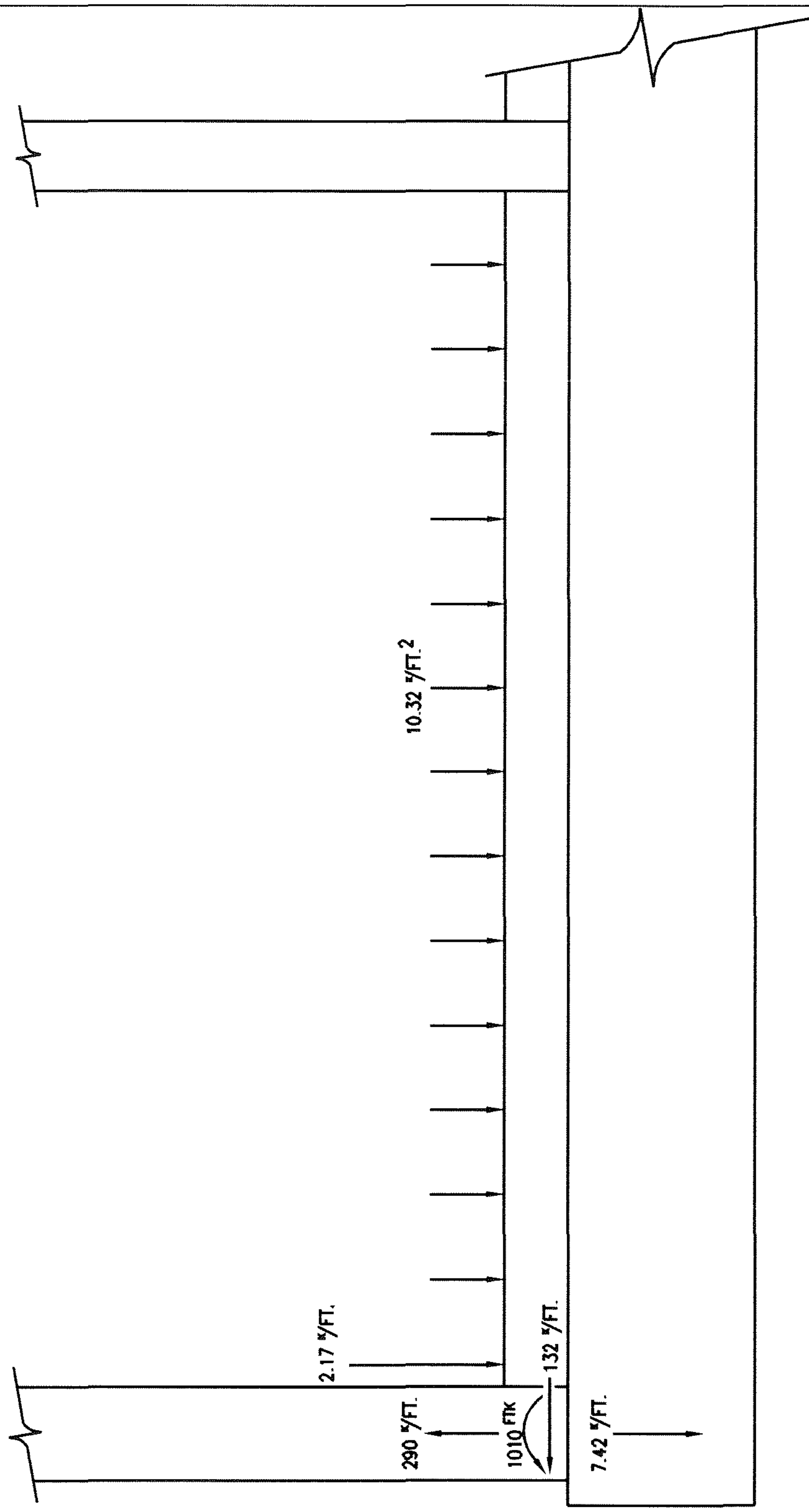
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-13 CYLINDER AND DOME-LOAD CONDITION (C) – 1.0P	
MIC. No. 1999MC3752	REV. No. 17A

$C = 1.0D \pm 0.05D + 1.5P + 1.0\ (T+TL)$



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-14 LOADING DIAGRAM IN MAT-LOAD CONDITION (A) - 1.5P	
MIC. No. 1999MC3753	REV. No. 17A

$$C = 1.0D \pm 0.05D + 1.25P + 1.0\,(T' + TL') + 1.25E'$$

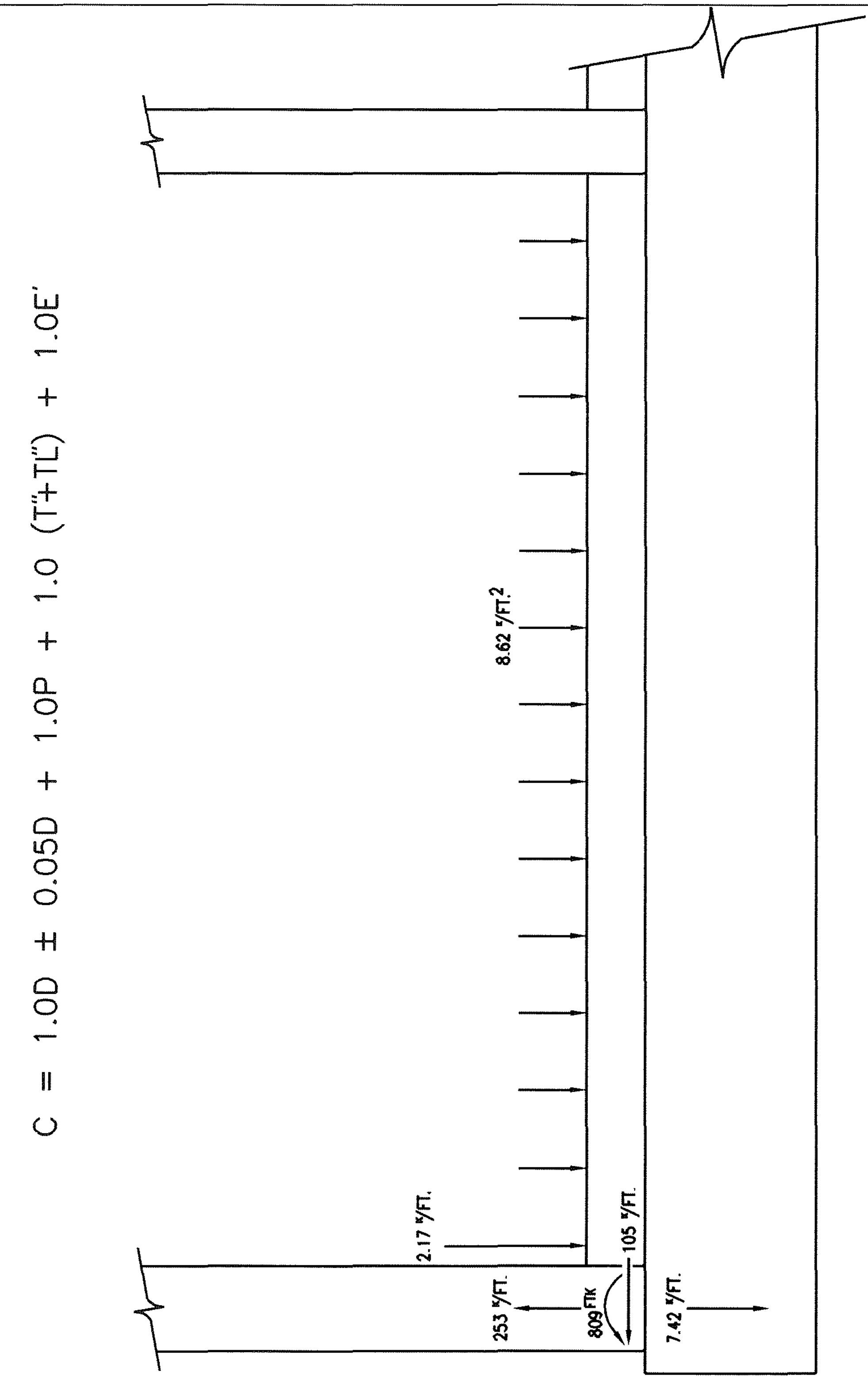


INDIAN POINT UNIT No. 2

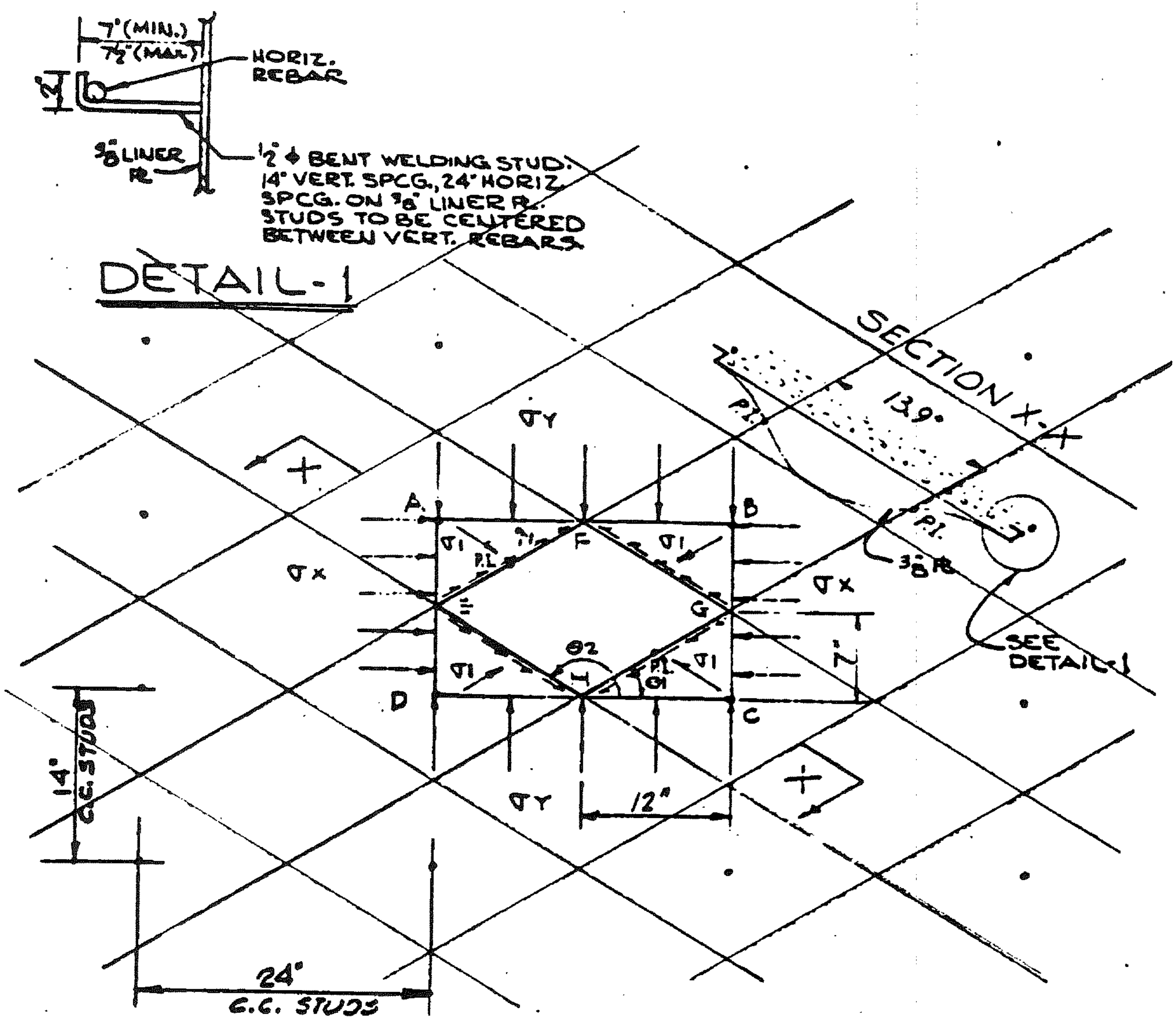
UFSAR FIGURE 5.1-15

LOADING DIAGRAM IN MAT-LOAD CONDITION (B) - 1.25P

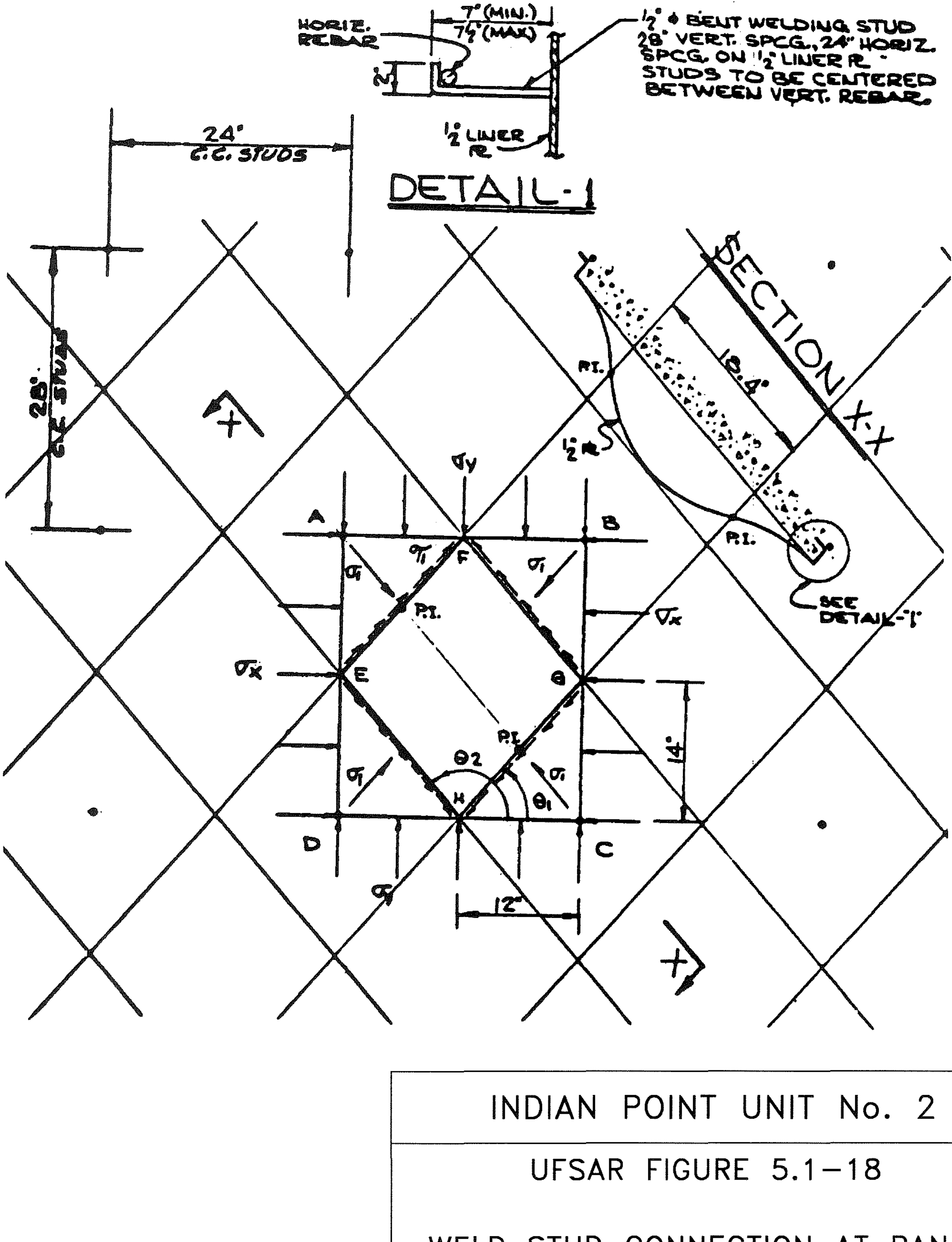
MIC. No. 1999MC3754	REV. No. 17A



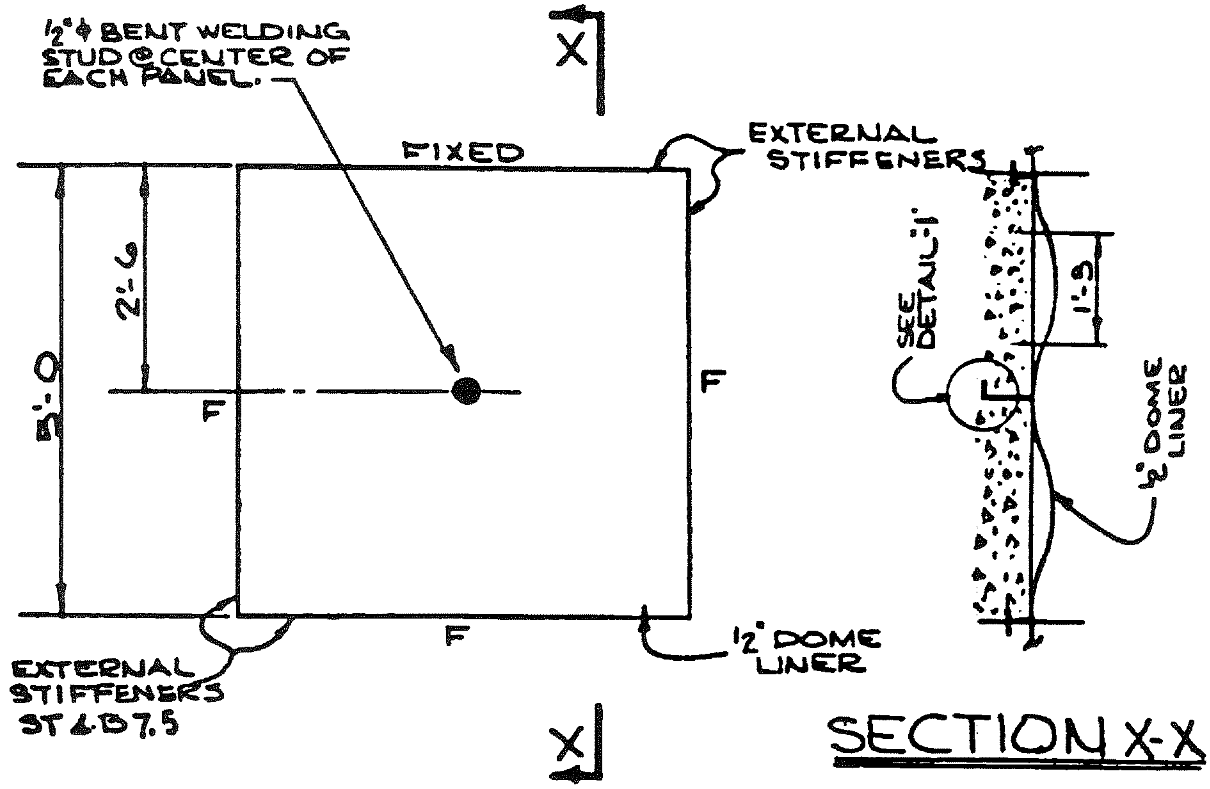
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-16 LOADING DIAGRAM IN MAT-LOAD CONDITION (C) - 1.0P	
MIC. No. 1999MC3755	REV. No. 17A

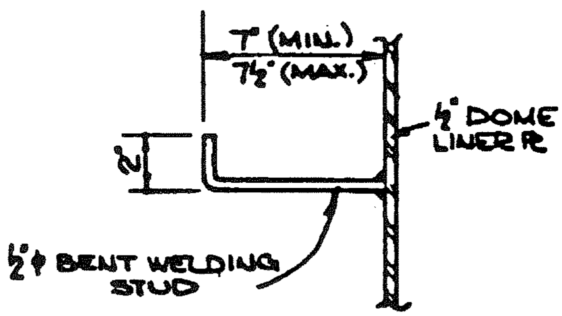


INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-17	
WELD STUD CONNECTION AT PANEL LOW POINT	
MIC. No. 1999MC3756	REV. No. 17A



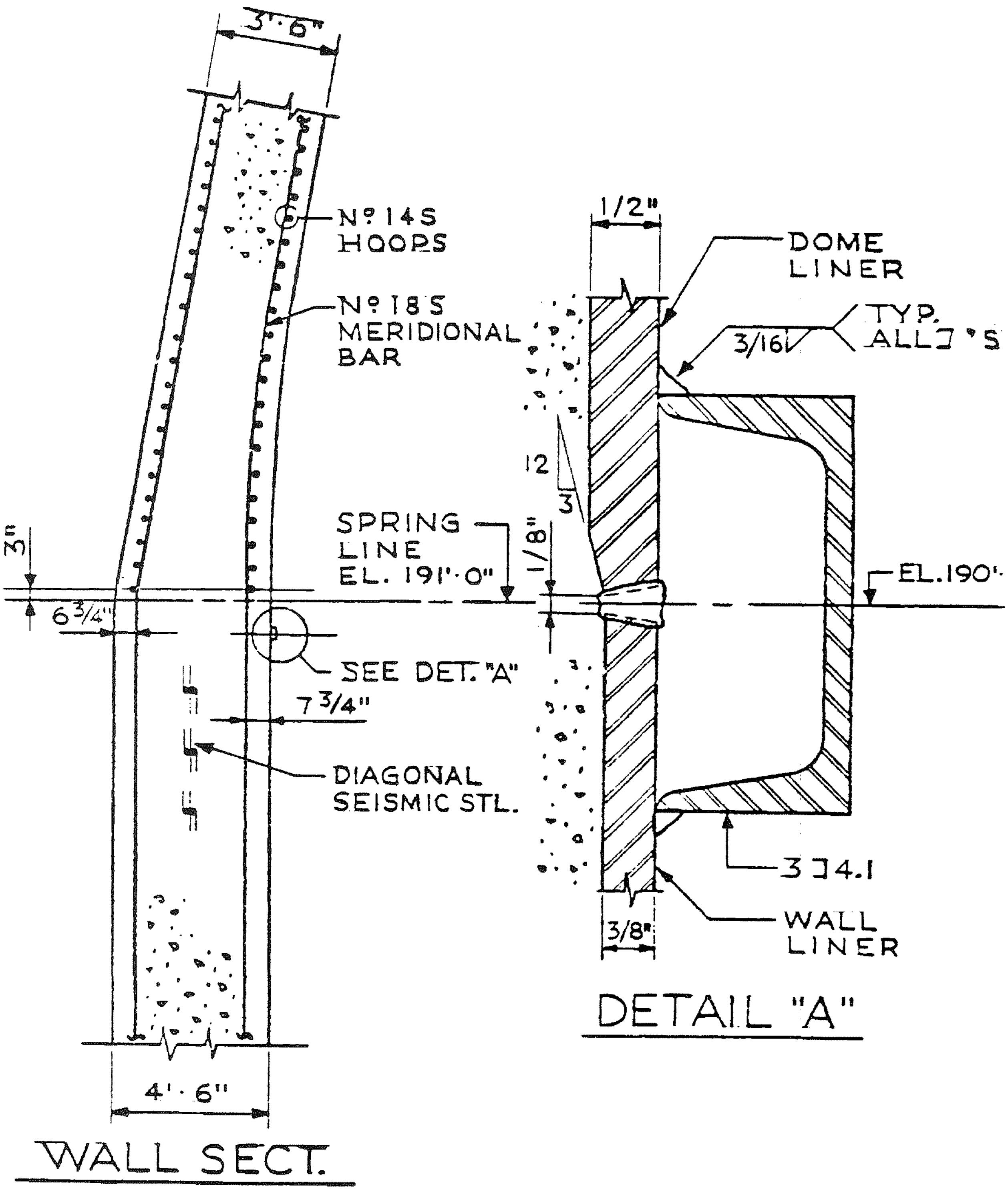
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-18	
WELD STUD CONNECTION AT PANEL LOW POINT	
MIC. No. 1999MC3757	REV. No. 17A



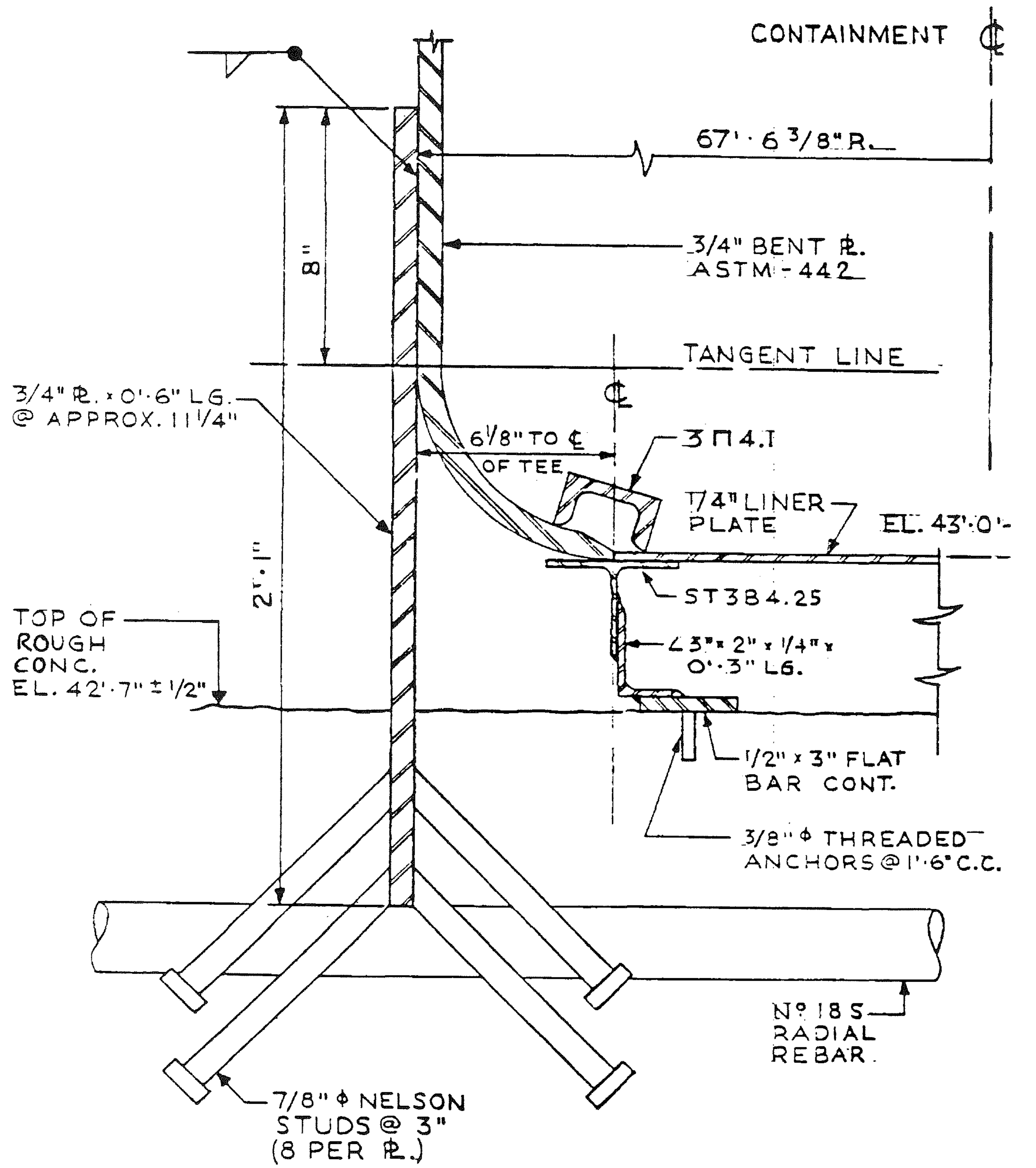


DETAIL 1

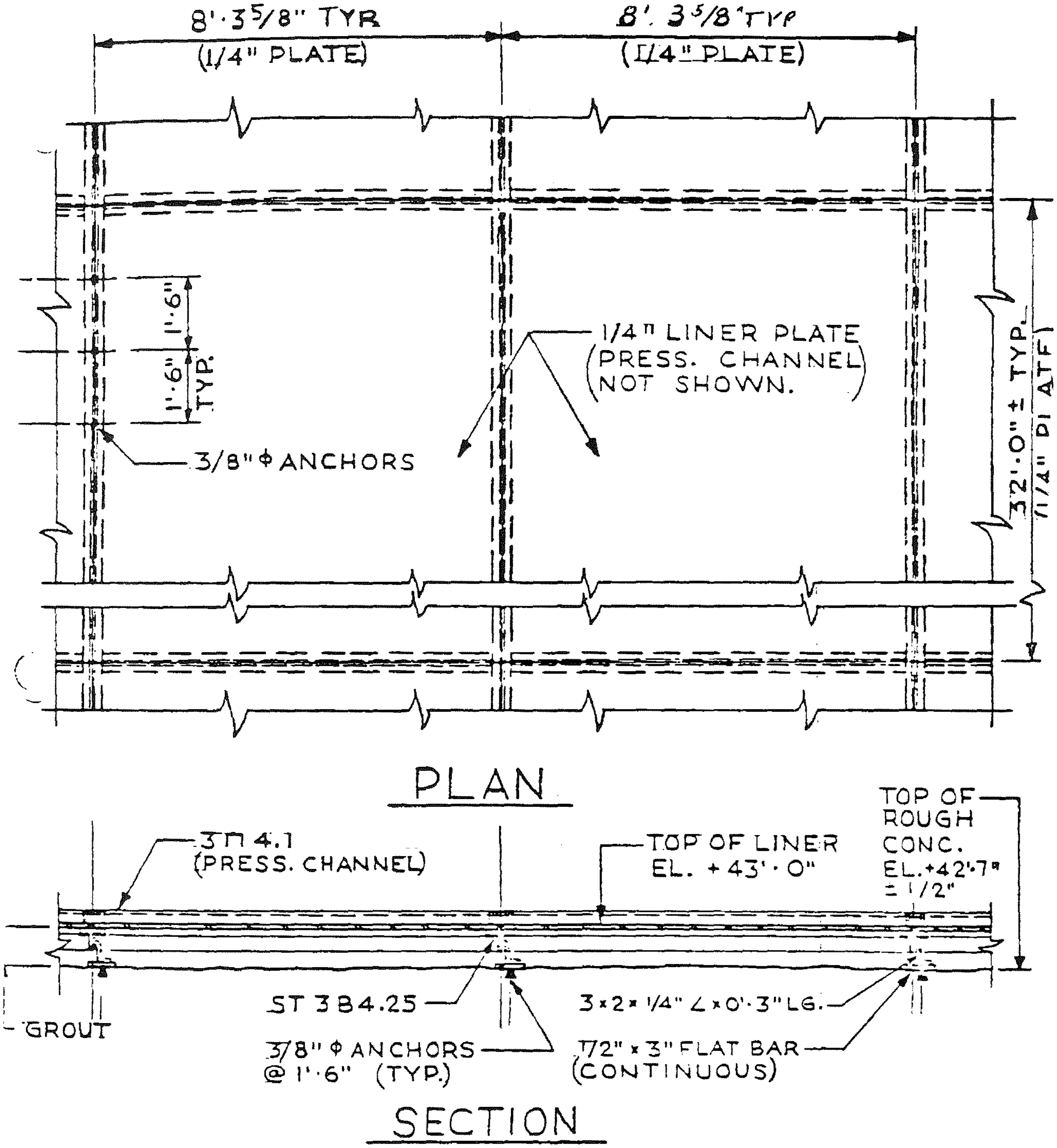
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-19 WELD STUD CONNECTION AT PANEL CENTER	
MIC. No. 1999MC3758	REV. No. 17A



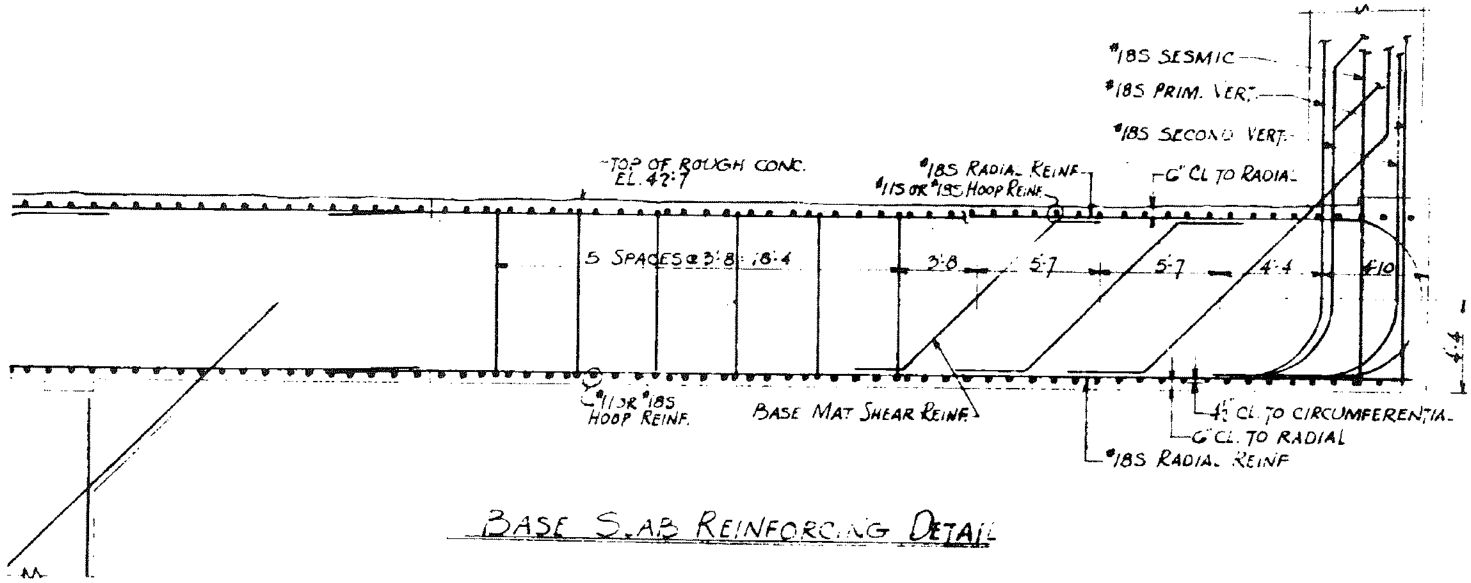
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1–20 WALL SECTION	
MIC. No. 1999MC3759	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE NO. 5.1-21	
CYLINDER BASE SLAB LINER JUNCTURE	
MIC. No. 1999MC3760	REV. No. 17A

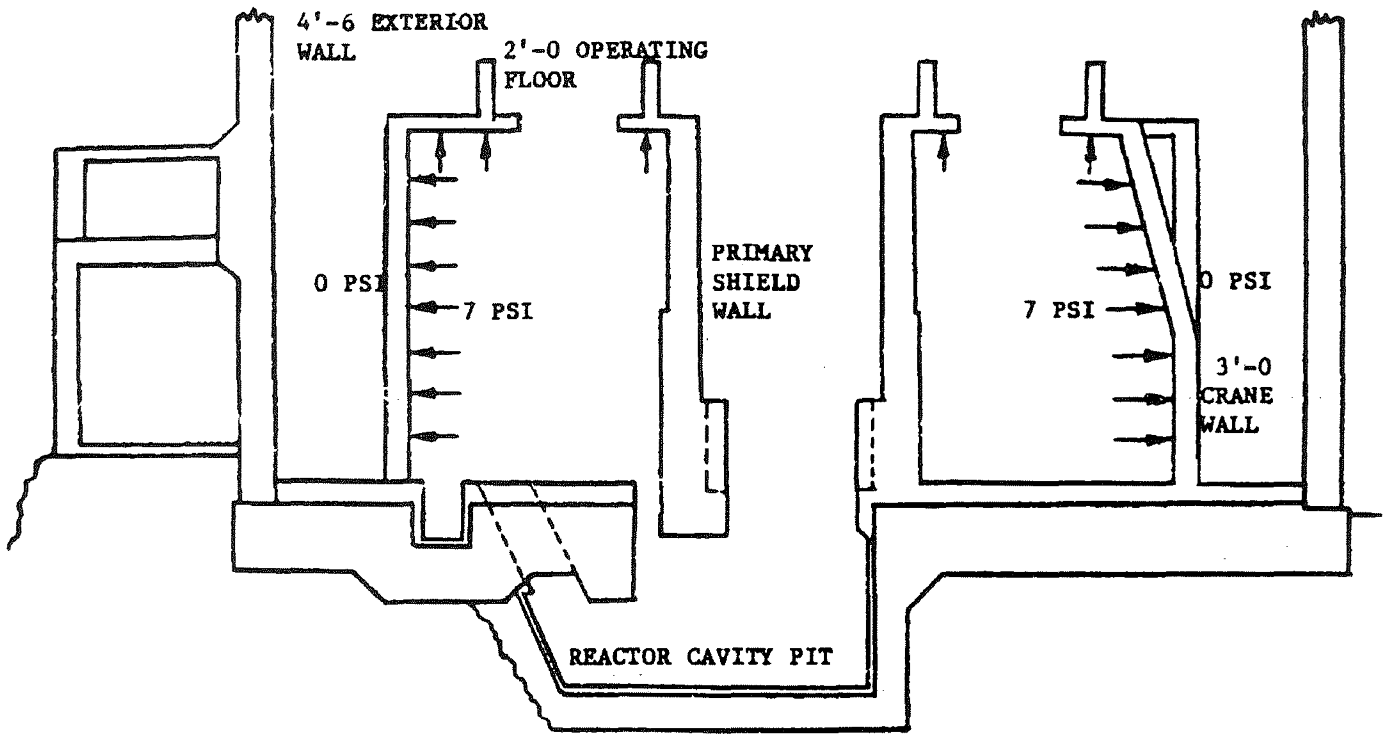


INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-22 TYPICAL BASE MAT LINER DETAIL	
MIC. No. 1999MC3761	REV. No. 17A

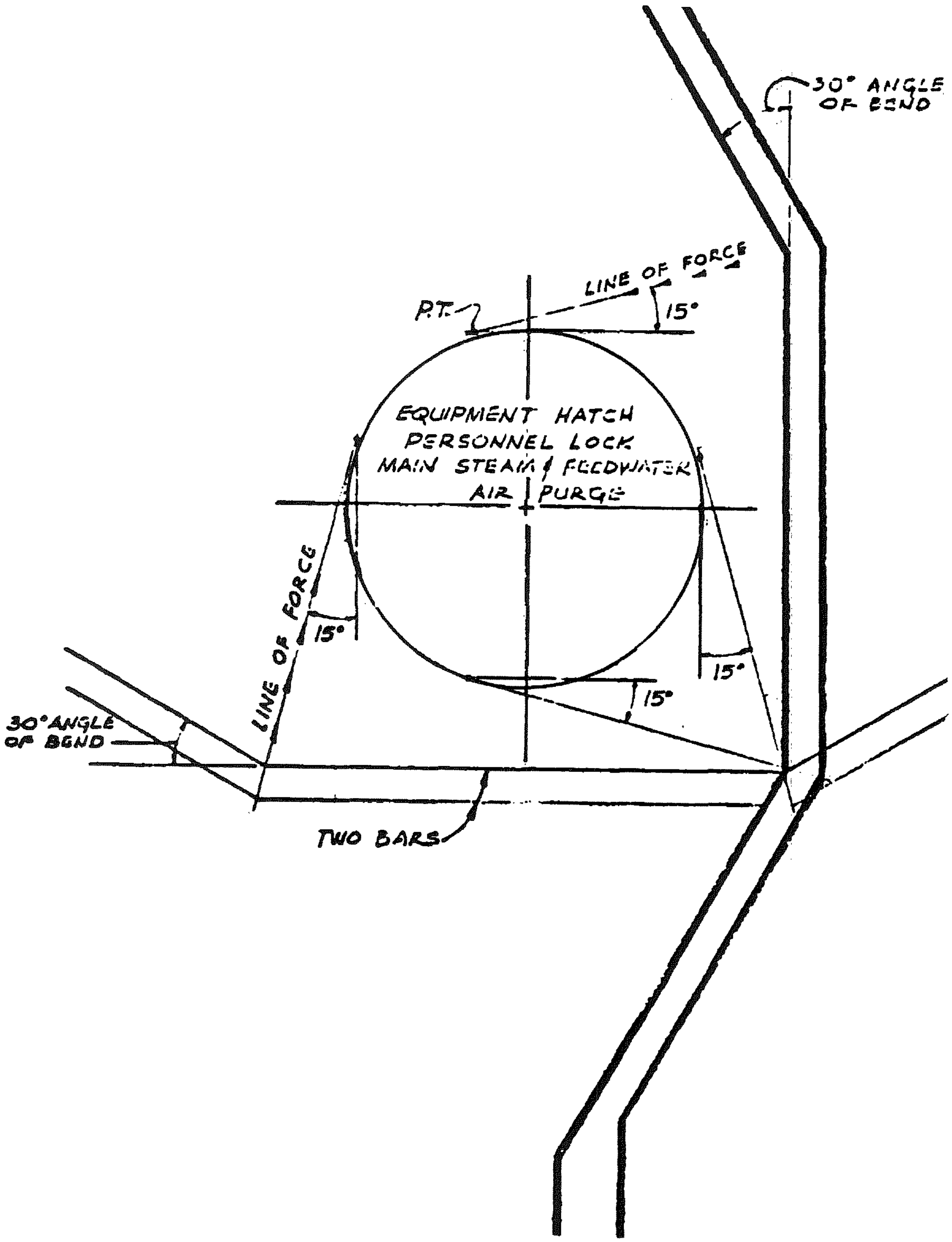


BASE SLAB REINFORCING DETAIL

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-23 BASE SLAB REINFORCING DETAIL	
MIC. No. 1999MC3762	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-24 REACTOR CAVITY PIT	
MIC. No. 1999MC3763	REV. No. 17A

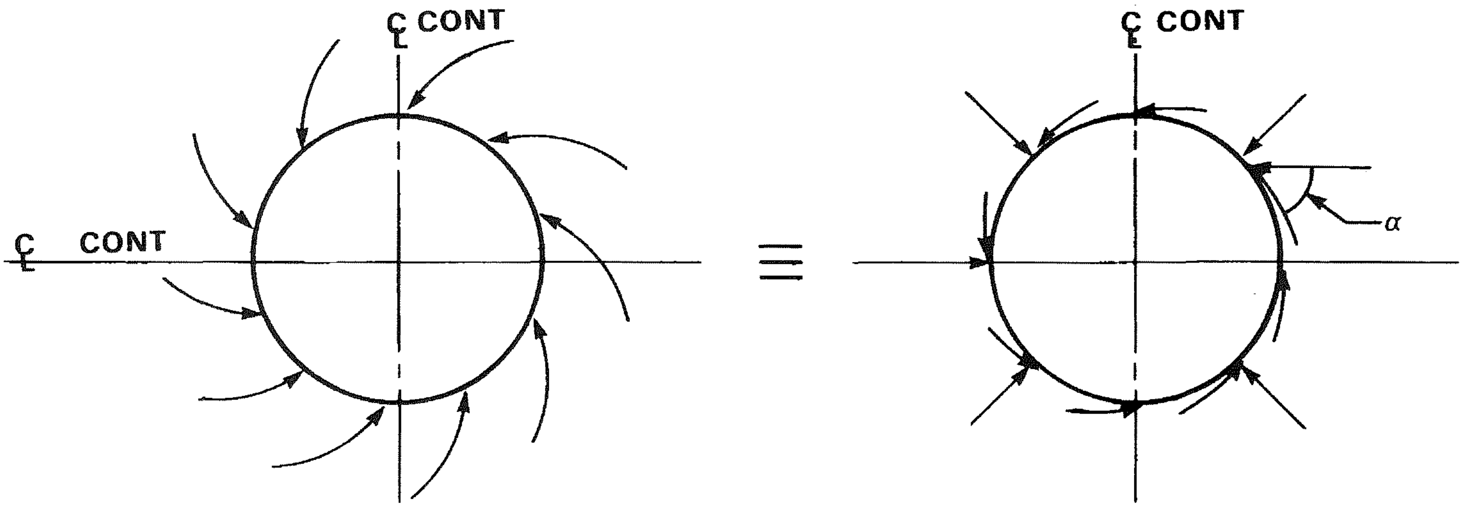


INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.1-25

EQUIPMENT HATCH PERSONNEL LOCK, MAIN STEAM AND FEEDWATER, AIR PURGE - REBAR

MIC. No. 1999MC3764	REV. No. 17A



INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.1–26

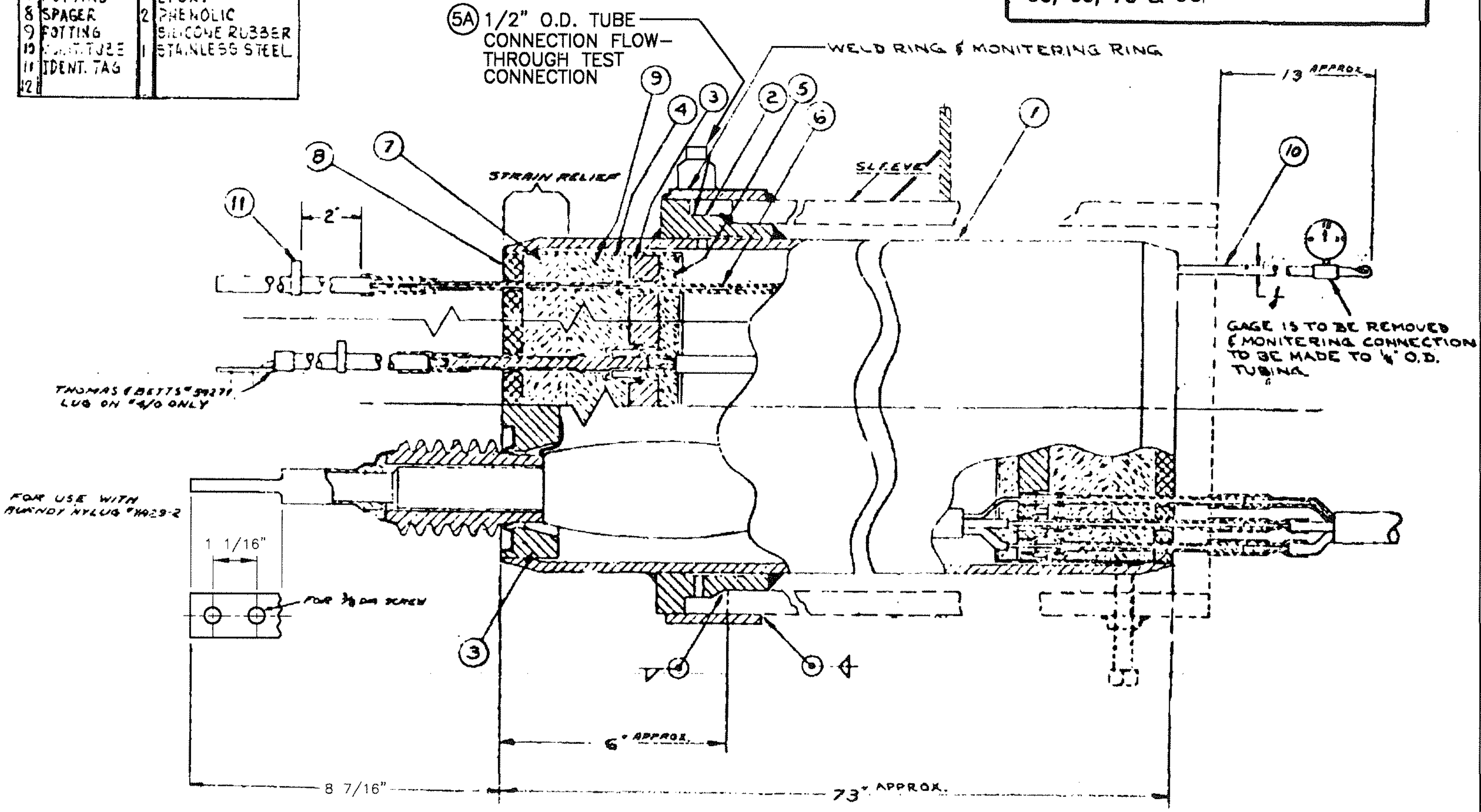
TORSIONAL EFFECTS

MIC. No. 1999MC3765	REV. No. 17A

NO.	PART NAME	QTY	DESCRIPTION
1	CONTAINER	1	STAINLESS STEEL
2	FLANGE	1	CARBON STEEL
3	FEADER	2	STAINLESS STEEL
4	TUBE	2	STAINLESS STEEL
5	POTTING		SILICONE RUBBER
6	INSULATION		SILICONE SLEEVING
7	POTTING		EPOXY
8	SPACER	2	PHENOLIC
9	POTTING		SILICONE RUBBER
10	[illegible] TUBE	1	STAINLESS STEEL
11	IDENT. TAG		
12			

THIS DWG. PERTAINS TO PENETRATIONS:

H11, 12, 13, 14, 15, 16, 17, 18, 19, 21, 22, 24, 26, 29, 31, 33, 34, 36, 37, 38, 39, 41, 42, 43, 44, 45, 47, 48, 49, 50, 51, 2, 53, 54, 55, 56, 57, 58, 59, 60, 61, 63, 64, 66, 67, 68, 69, 70 & 65.



INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.1-27

TYPICAL ELECTRICAL PENETRATION

MIC. No. 1999MC3766

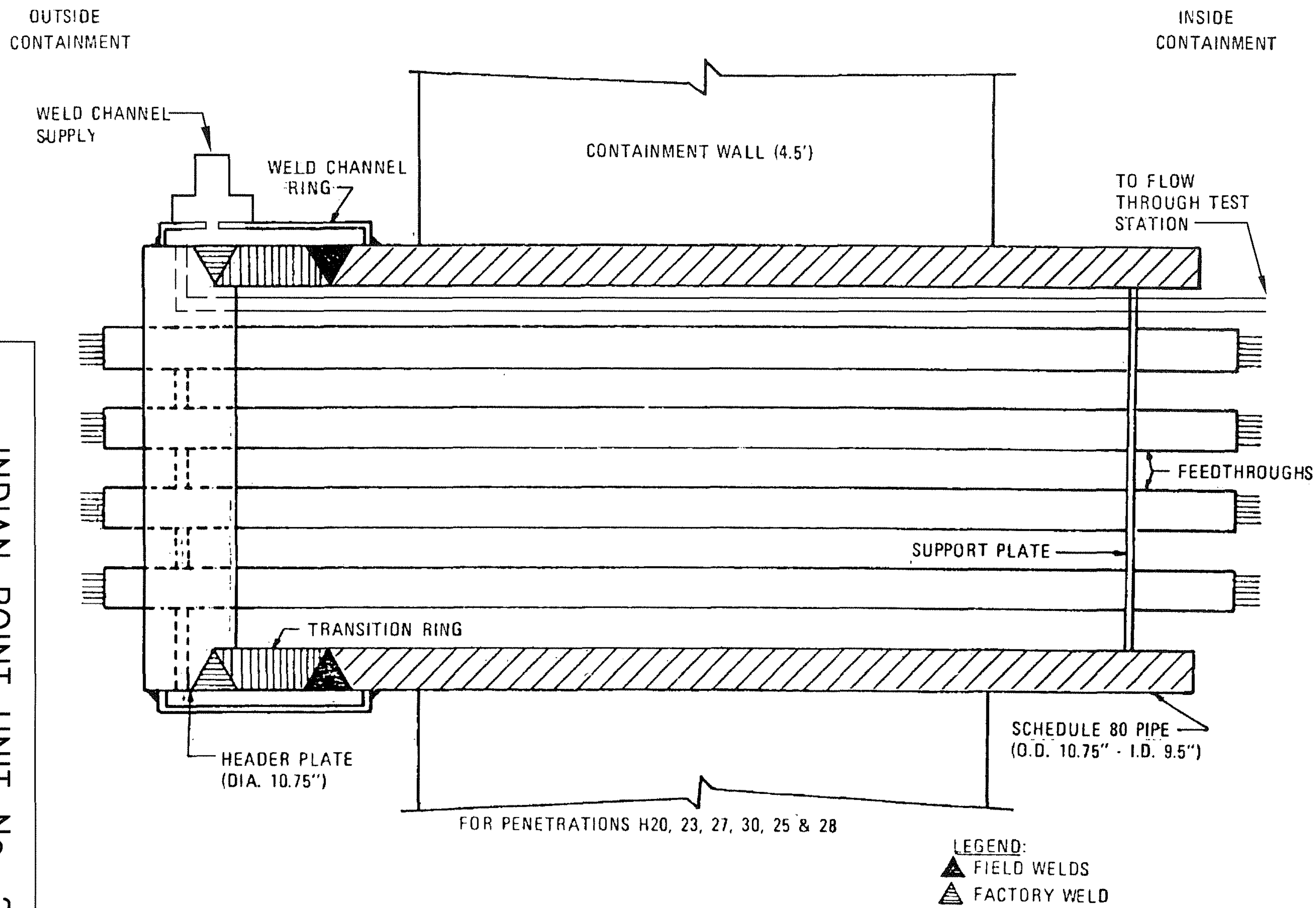
REV. No. 17A

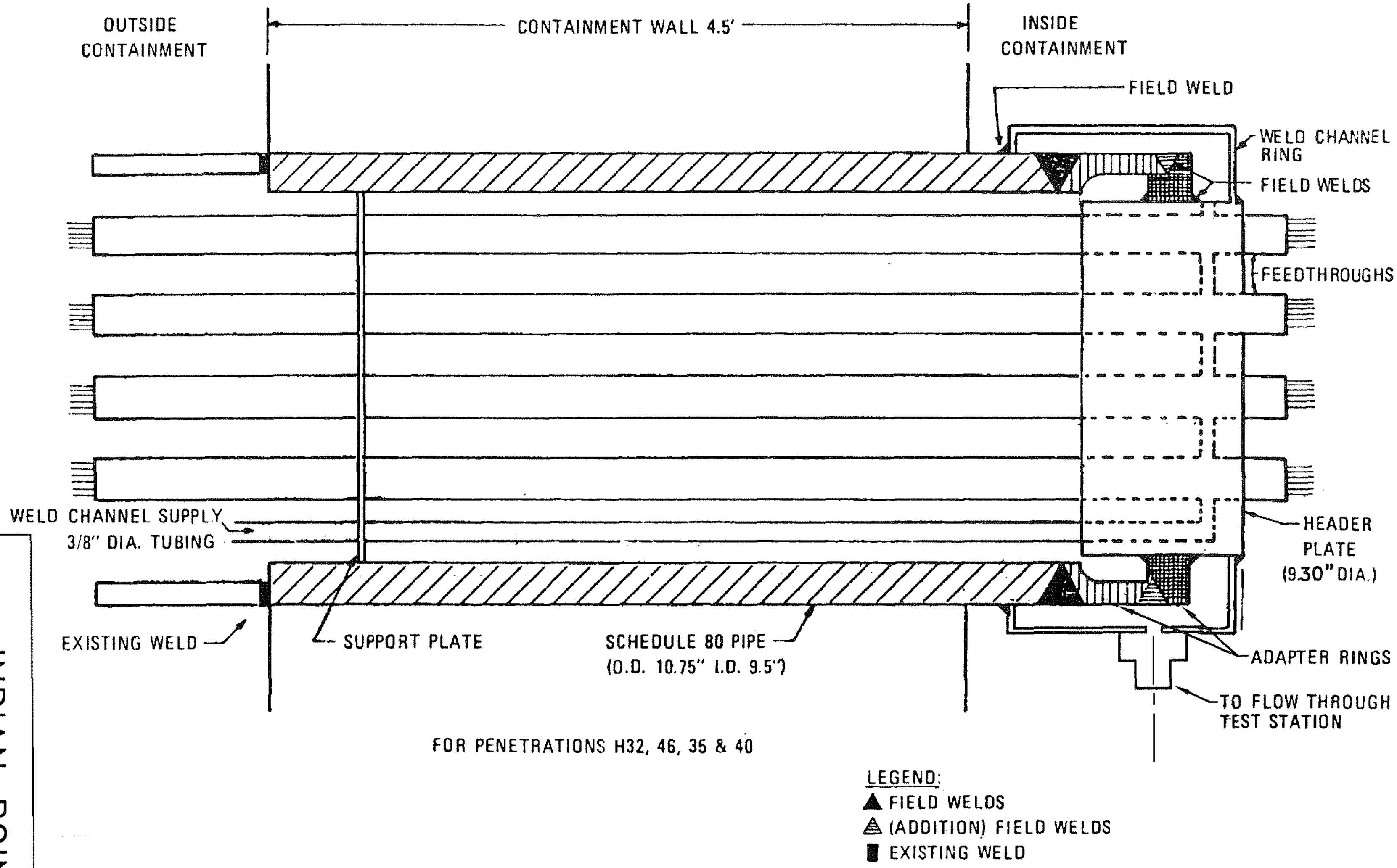
INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.1–28

CONAX PENETRATIONS – OUTSIDE CONTAINMENT WELD

MIC. No. 1999MC3767 | REV. No. 17A



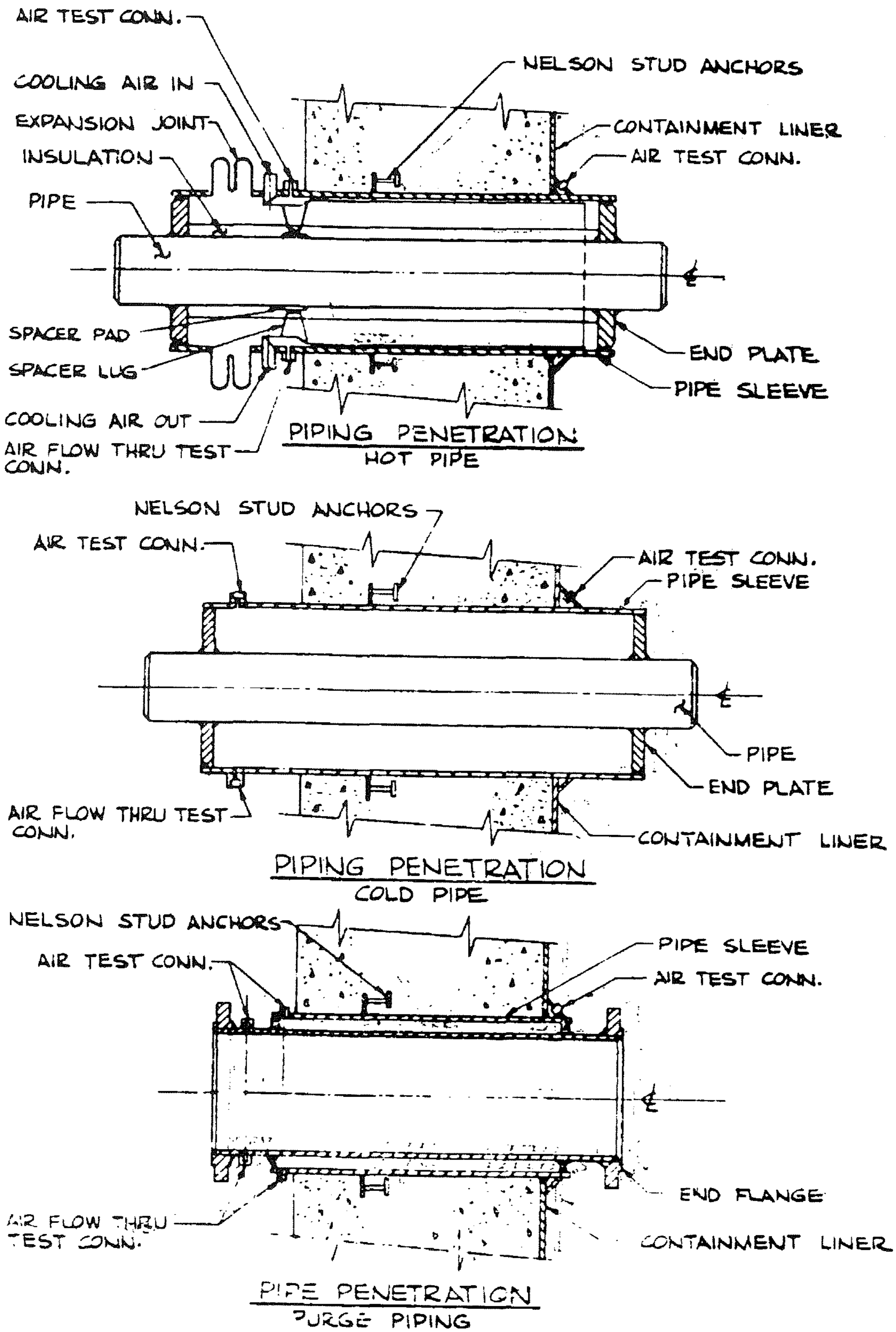


INDIAN POINT UNIT No. 2

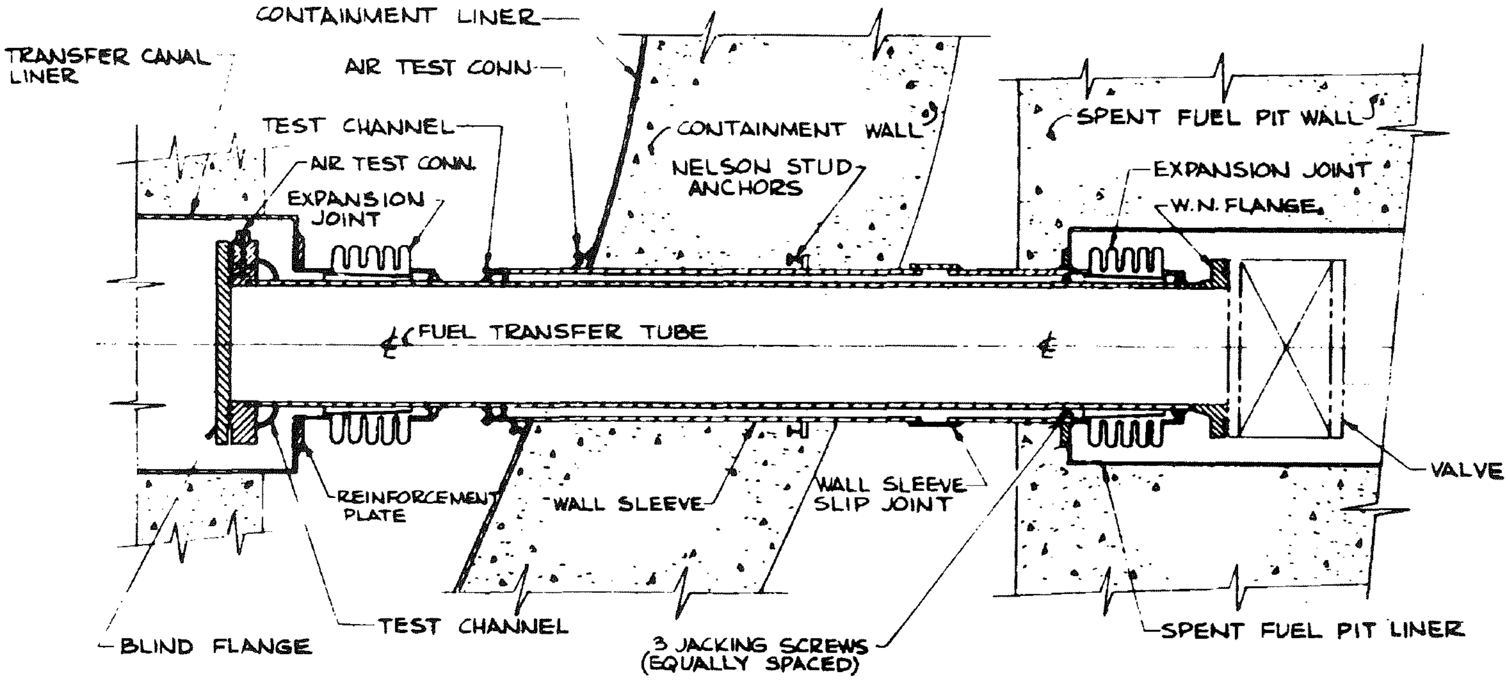
UFSAR FIGURE 5.1–29

CONAX PENETRATIONS – INSIDE CONTAINMENT WELD

MIC. No. 1999MC3768	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1–30 TYPICAL PIPING PENETRATION	
MIC. No. 1999MC3769	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1–31 FUEL TRANSFER TUBE PENETRATION (CONCEPTUAL DRAWING)	
MIC. No. 1999MC3770	REV. No. 17A

(Sample calculation)

GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC.

NAME OF COMPANY: Con Edison - Indian Point N° 2 | J. O. NO. 9321-06 | SHEET NO. 6 OF

SUBJECT: Containment - Stresses on Penetrations & Liner | DATE | COMP. BY A.B.S. | C'K'D BY F.X.M.

Electrical Penetrations

EL. 57'-0 D.A = 10.75
Sleeve t = .594

A) Compression V and Tension H

$$S' = \frac{(70.5)(837)(12)}{(1000)\,24.96}(.43) \text{ — restraint}$$

$= 12.2\ ^{k}/_{in^2}$ tension

Liner As/FT = (.75)(12) = 9.00
Seismic + 8(1/10)(1.414) = 2.26
Hoops + 16(.../12) = 13.70
= 24.96 $^{\square}/_{FT}$

$S \approx 20.2\ ^{k}/_{in^2}$ compression

$$\Delta_{UN} = \frac{2}{3}\cdot\frac{5.375}{29\times10^3}\left[5(20.2) - 12.2\right] = 11.00\times10^{-3} = .011 \times .5 = 0.0055$$

88.8

(radial deformation)

$$\lambda = \sqrt[4]{\frac{3(1-.25)}{(5.375^2)(.594)^2}} = .724$$

$$S_{SLEEVE} = \frac{(.0055)(29\times10^3)(.750)(.724)}{\left[(1-.25) + \frac{(.750)(5.375)(.724)}{2(.594)}\right]2(.594)}$$

3.21

$0.225\times10^3 = \underline{\underline{22.5}}^{ksi}$

$$S_{LINER} = \frac{(.0055)(29\times10^3)}{5.375\left[(1-.25) + \frac{(.750)(5.375)(.724)}{2(.594)}\right]} = .00931\times10^3 = \underline{\underline{9.3}}^{ksi}$$

3.21

INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.1-32

CONTAINMENT-STRESSES ON PENETRATIONS AND LINER - SHEET 6

MIC. No. 1999MC3771	REV. No. 17A

(Sample calculation)

FORM 50: GENERAL COMPUTATION SHEET

UNITED ENGINEERS & CONSTRUCTORS INC.

NAME OF COMPANY: CON EDISON - INDIAN POINT #2 | J. O NO. 9321-06 | SHEET NO. 7 OF ____

SUBJECT: CONTAINMENT - STRESSES ON PENETRATIONS & LINER | DATE ____ | COMP. BY B.B.S. | C'K'D BY F.C.

ELECTRICAL PENETRATIONS

8) TENSION V & TENSION H

$S = 12.2\ ^{K}/_{in^2}$

$S' \ \frac{297}{27.26} = 10.9\ ^{K}/_{in^2}$

As/FT (V)

LINER	$.75(12) =$	9
VERTICAL BARS	$4 \times 4 =$	16
SEISMIC	$8(\frac{12}{30})(1.414) =$	2.26
		27.26

$\Delta_{UN} = \frac{2}{3} \cdot \frac{5.375}{29 \times 10^3} \left[5\,\overset{71.9}{(12.2) + 10.9} \right]$

$\Delta_{UN} = \frac{71.9}{88.8} \times .011 = .0089$

$S_{SLEEVE} = \frac{71.9}{88.8} \times 22.5 = \underline{\underline{18.2}}\ ^{KSI}$

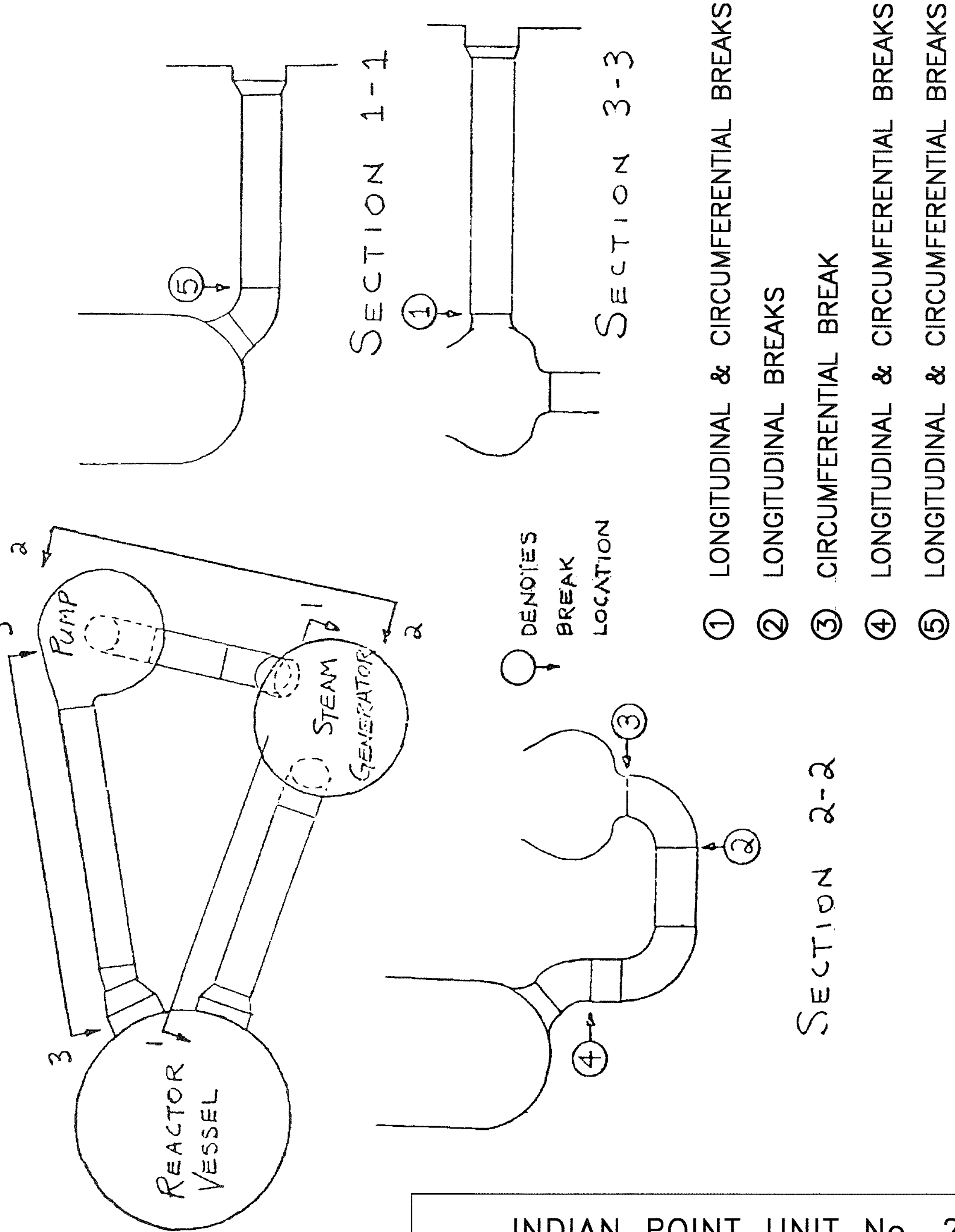
$S_{LINER} = \frac{71.9}{88.8} \times 9.3 = \underline{\underline{7.52}}\ ^{KSI}$

INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.1-33

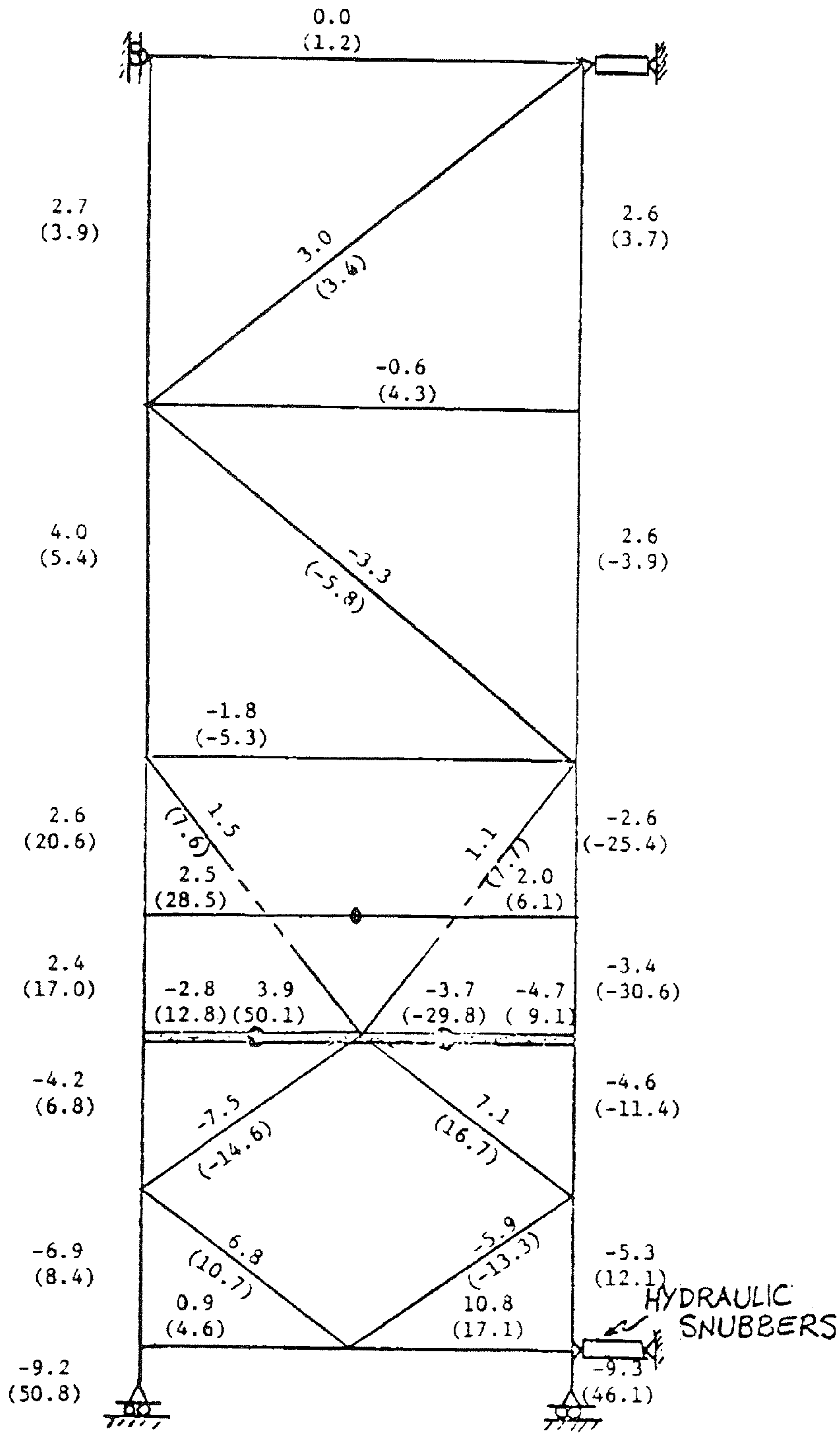
CONTAINMENT-STRESSES ON PENETRATIONS AND LINER - SHEET 7

MIC. No. 1999MC3772 | REV. No. 17A



(THIS FIGURE DEPICTS ORIGINAL HISTORICAL DESIGN DATA AND DOES NOT REFLECT THE CURRENT CONFIGURATION OR ANALYSIS.)

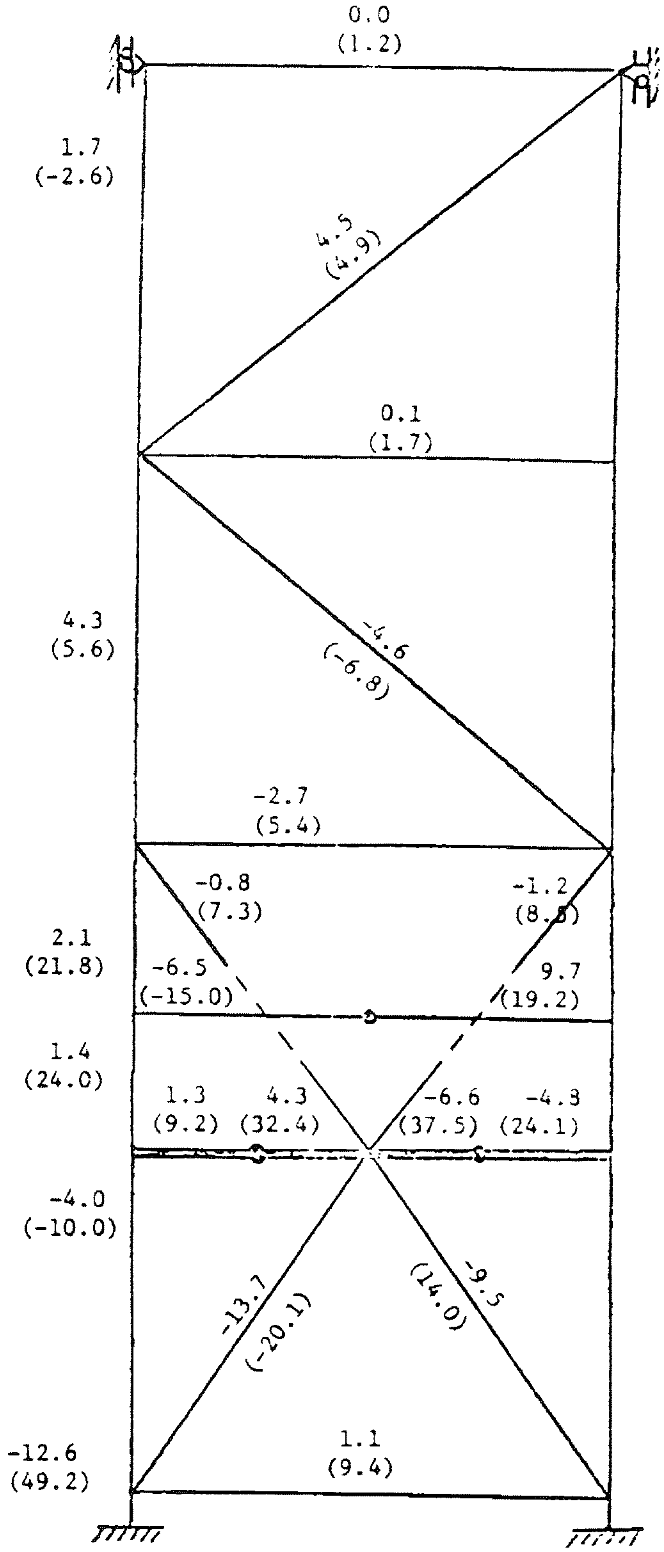
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-34 ASSUMED PIPE RUPTURE ACCIDENT BREAK LOCATIONS	
MIC. No. 1999MC3773	REV. No. 17A



For Section location,
see Figure 5.1-43

(THIS FIGURE DEPICTS ORIGINAL HISTORICAL DESIGN DATA AND DOES NOT REFLECT THE CURRENT CONFIGURATION OR ANALYSIS.)

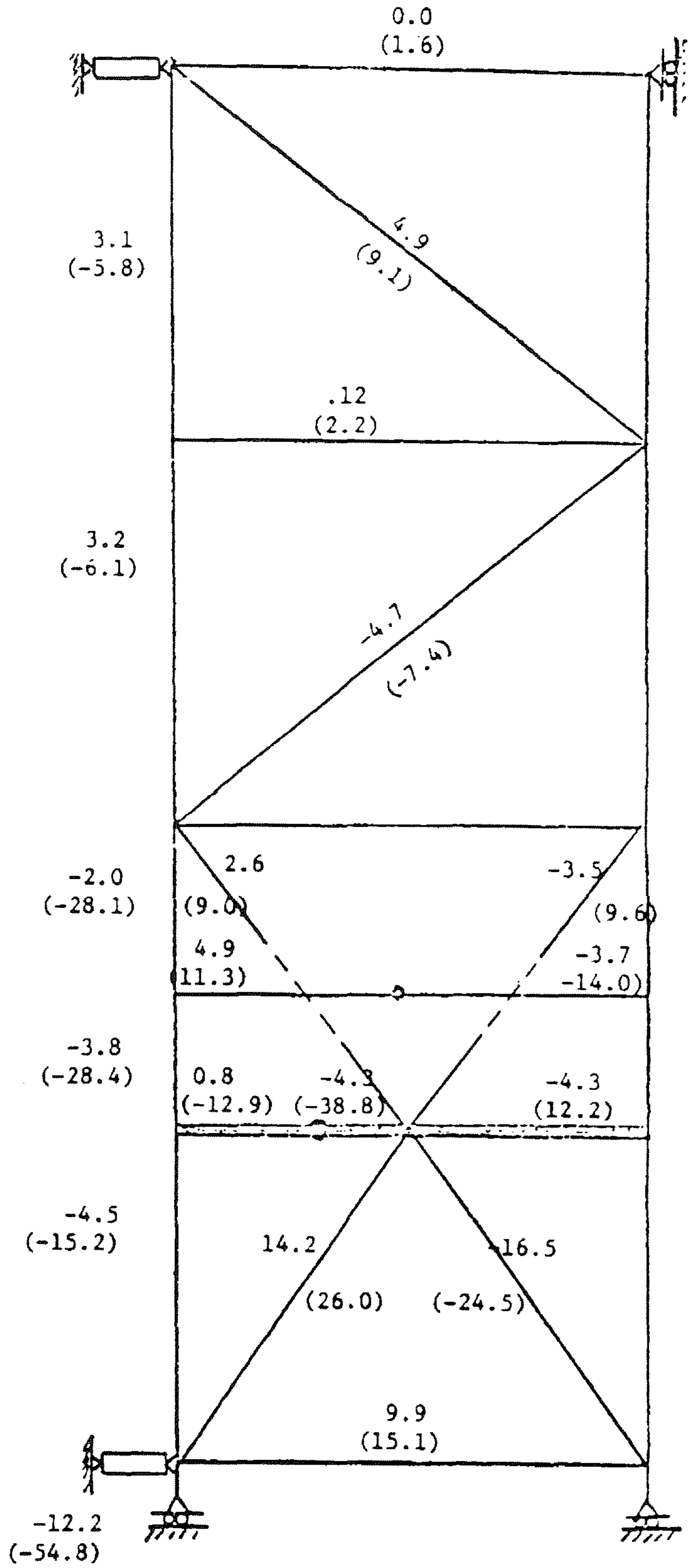
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-35 STEAM GENERATOR SUPPORT-SECTION 1-1	
MIC. No. 1999MC3774	REV. No. 17A



For Section location, see Figure 5.1–43

(THIS FIGURE DEPICTS ORIGINAL HISTORICAL DESIGN DATA AND DOES NOT REFLECT THE CURRENT CONFIGURATION OR ANALYSIS.)

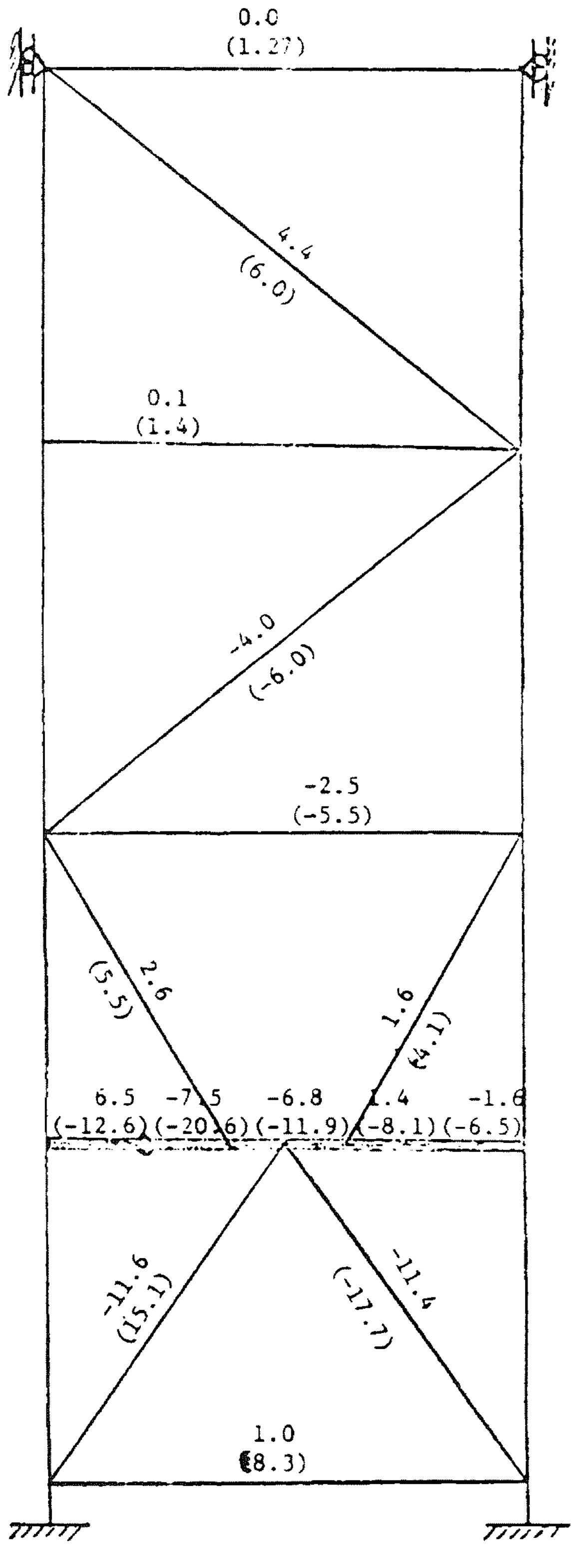
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1–36 STEAM GENERATOR SUPPORT–SECTION 2–2	
MIC. No. 1999MC3775	REV. No. 17A



For Section location, see Figure 5.1-43

(THIS FIGURE DEPICTS ORIGINAL HISTORICAL DESIGN DATA AND DOES NOT REFLECT THE CURRENT CONFIGURATION OR ANALYSIS.)

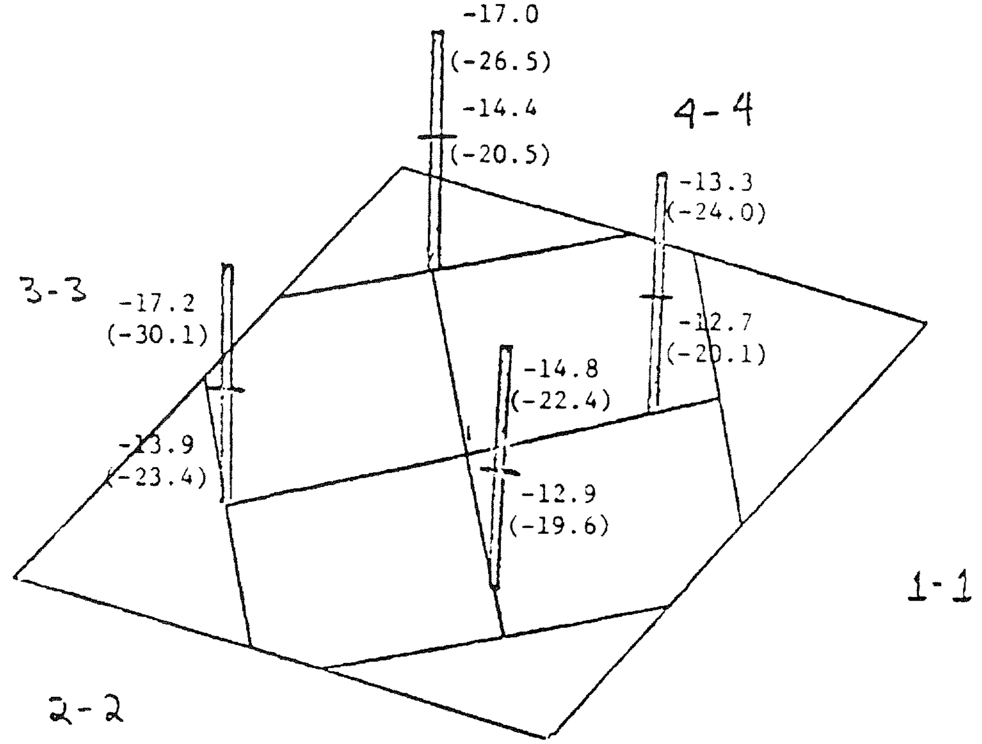
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-37 STEAM GENERATOR SUPPORT-SECTION 3-3	
MIC. No. 1999MC3776	REV. No. 17A



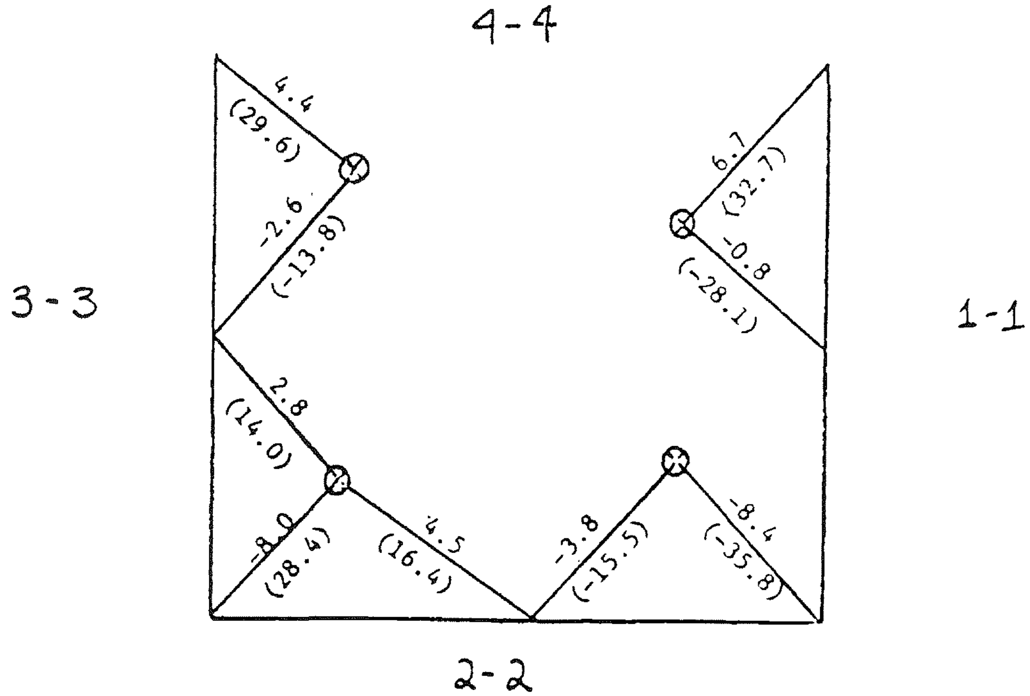
For Section location, see Figure 5.1–43

(THIS FIGURE DEPICTS ORIGINAL HISTORICAL DESIGN DATA AND DOES NOT REFLECT THE CURRENT CONFIGURATION OR ANALYSIS.)

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1–38 STEAM GENERATOR SUPPORT–SECTION 4–4	
MIC. No. 1999MC3777	REV. No. 17A



PLAN AT EL. 60'-0"

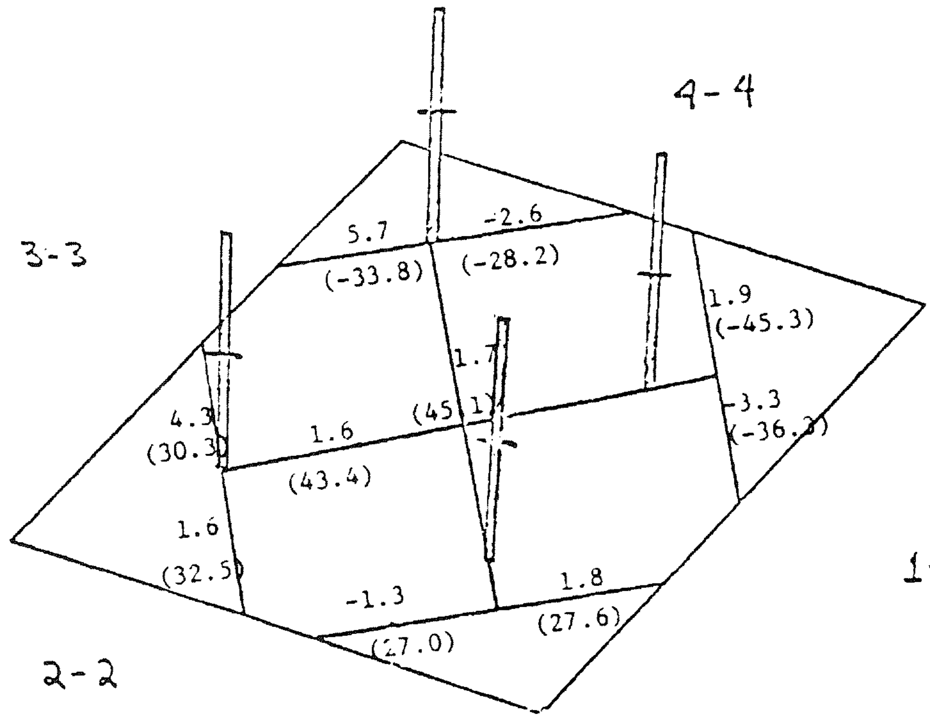


PLAN AT EL. 63'-0"

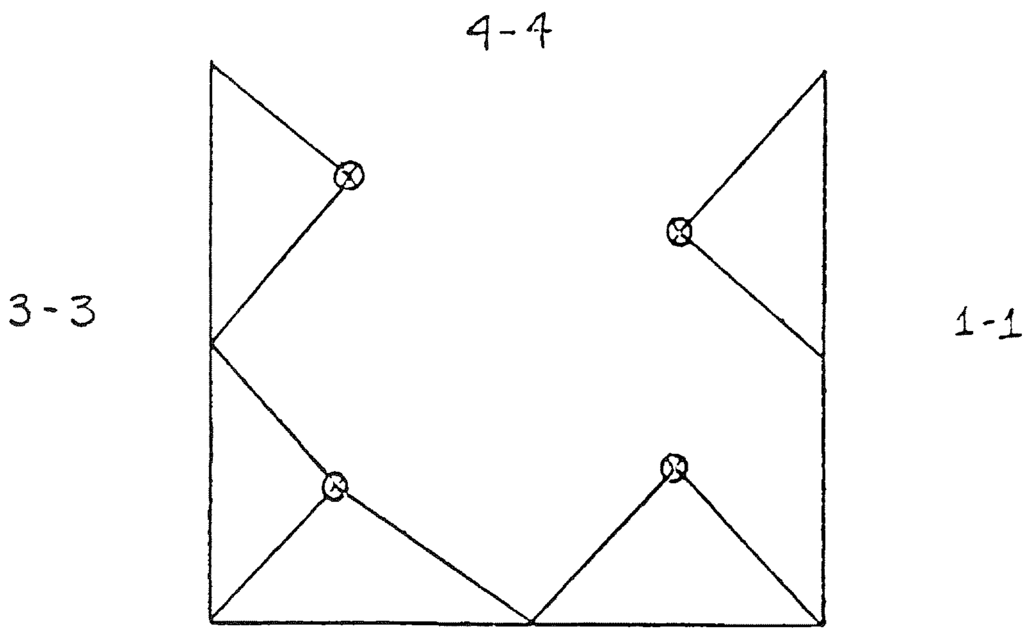
For Section location, see Figure 5.1-44

(THIS FIGURE DEPICTS ORIGINAL HISTORICAL DESIGN DATA AND DOES NOT REFLECT THE CURRENT CONFIGURATION OR ANALYSIS.)

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-39 STEAM GENERATOR SUPPORT-PLAN LOCATION ELEVATION 60 AND 63	
MIC. No. 1999MC3778	REV. No. 17A



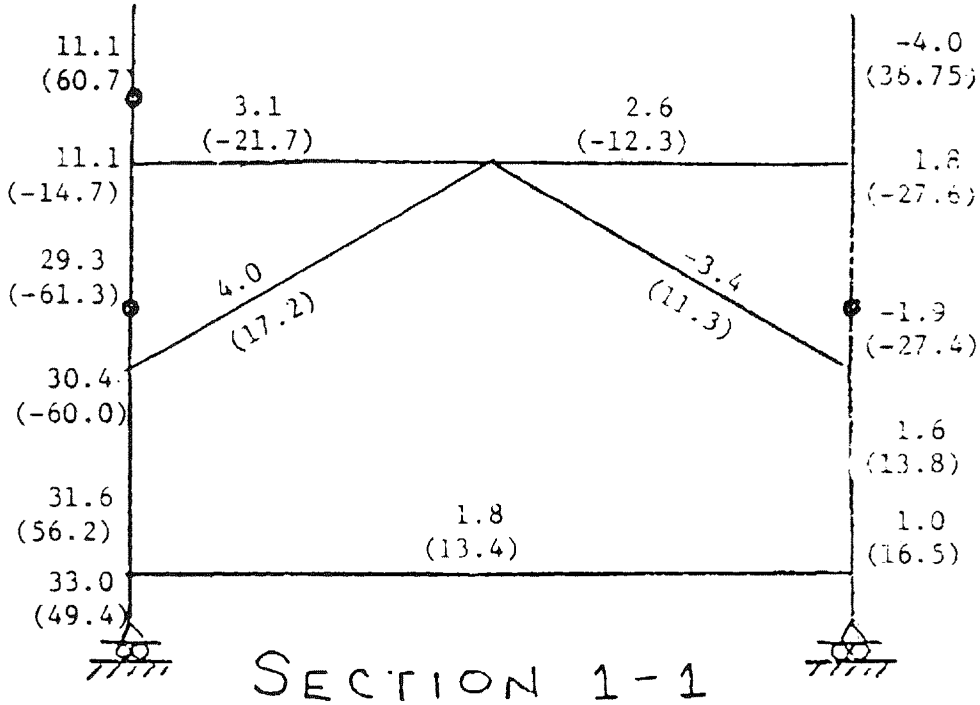
PLAN AT EL. 60'-0"



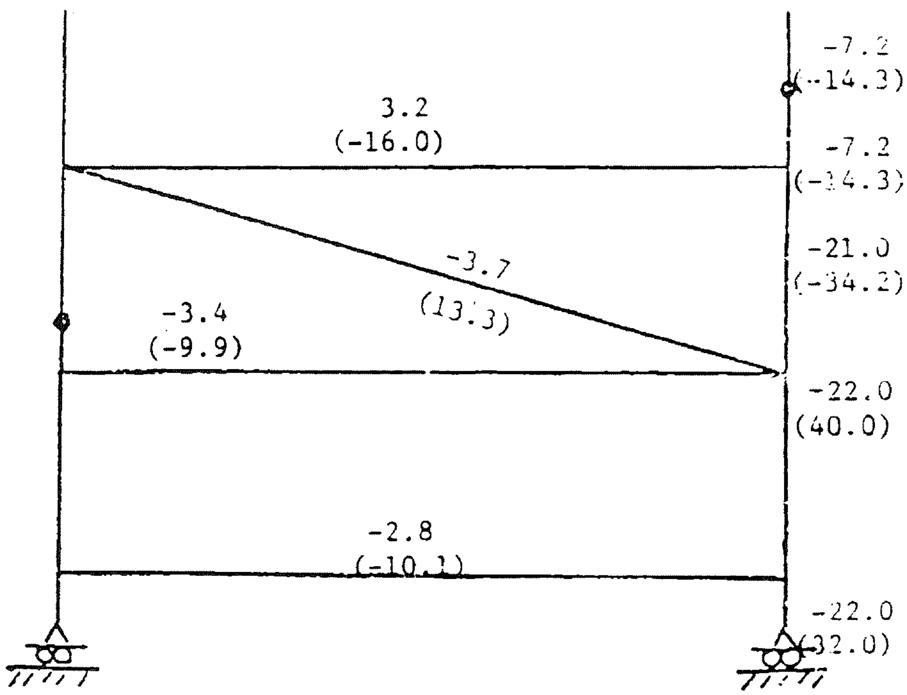
PLAN AT EL. 63'-0"

(THIS FIGURE DEPICTS ORIGINAL HISTORICAL DESIGN DATA AND DOES NOT REFLECT THE CURRENT CONFIGURATION OR ANALYSIS.)

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-40 STEAM GENERATOR SUPPORT-PLAN LOCATION ELEVATION 60 AND 63	
MIC. No. 1999MC3779	REV. No. 17A



SECTION 1-1

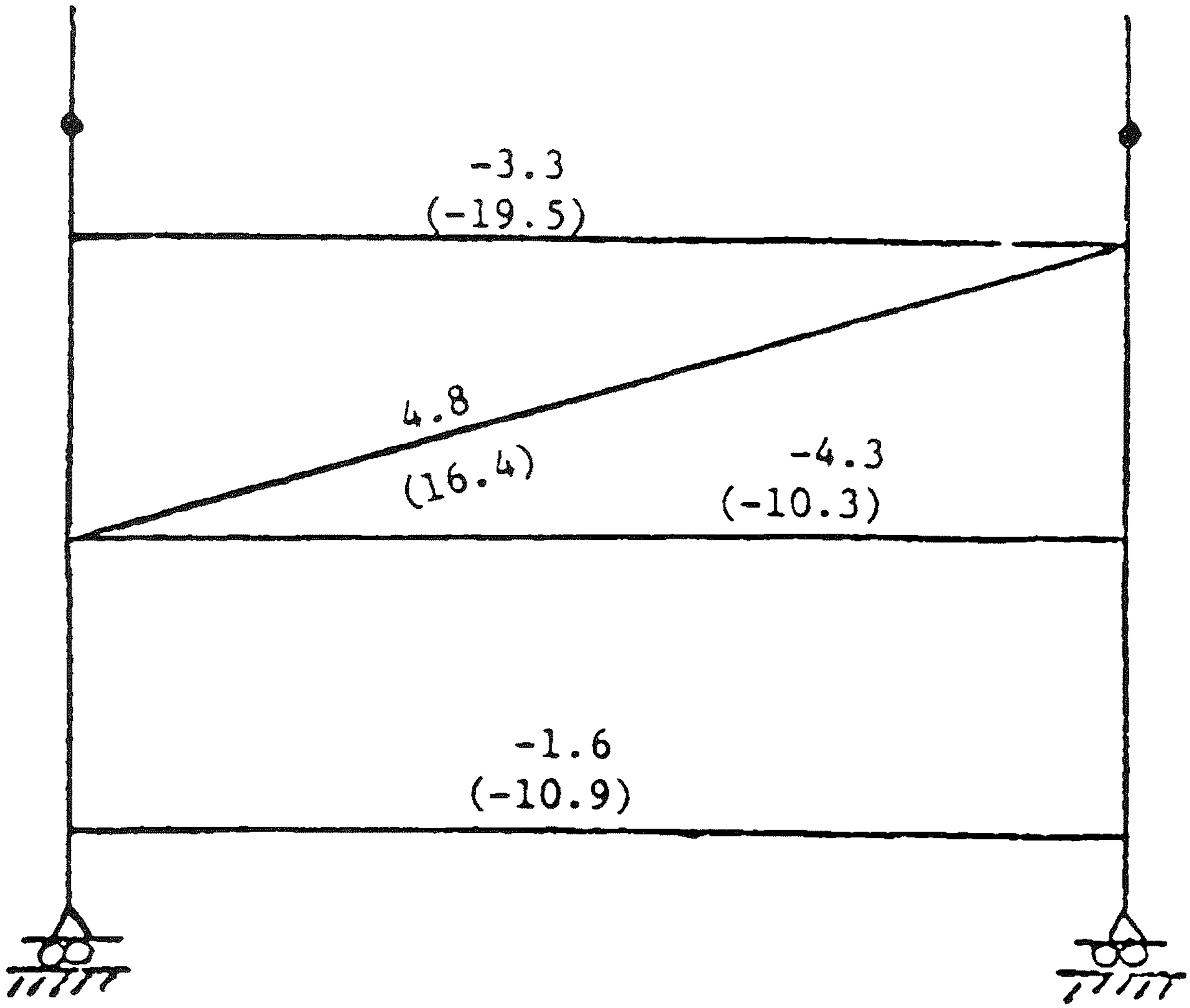


SECTION 2-2

For Section location, see Figure 5.1-44

(THIS FIGURE DEPICTS ORIGINAL HISTORICAL DESIGN DATA AND DOES NOT REFLECT THE CURRENT CONFIGURATION OR ANALYSIS.)

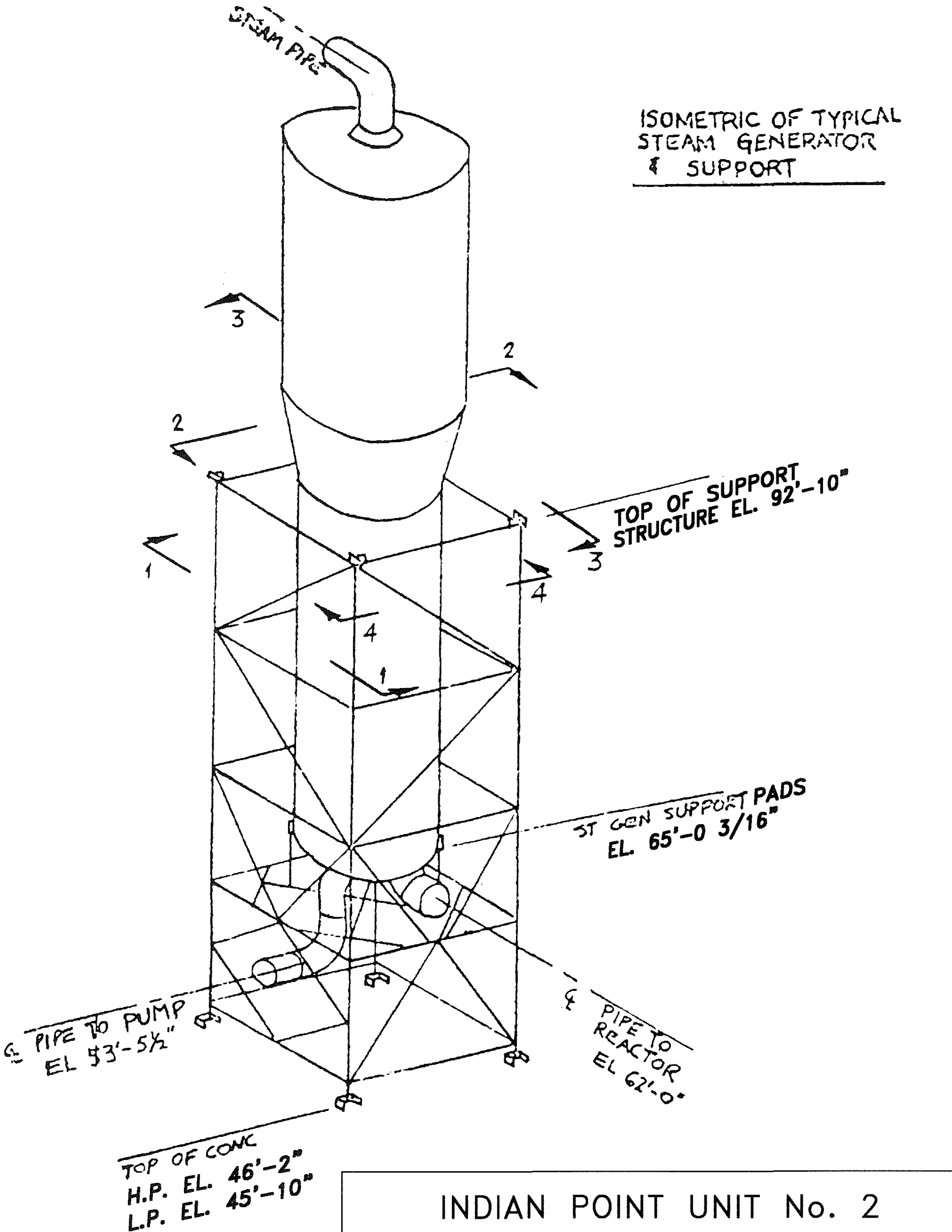
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-41 PUMP SUPPORT-SECTION 2-2 AND 3-3	
MIC. No. 1999MC3780	REV. No. 17A



For Section location, see Figure 5.1-44

(THIS FIGURE DEPICTS ORIGINAL HISTORICAL DESIGN DATA AND DOES NOT REFLECT THE CURRENT CONFIGURATION OR ANALYSIS.)

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-42 PUMP SUPPORT-SECTION 3-3	
MIC. No. 1999MC3781	REV. No. 17A

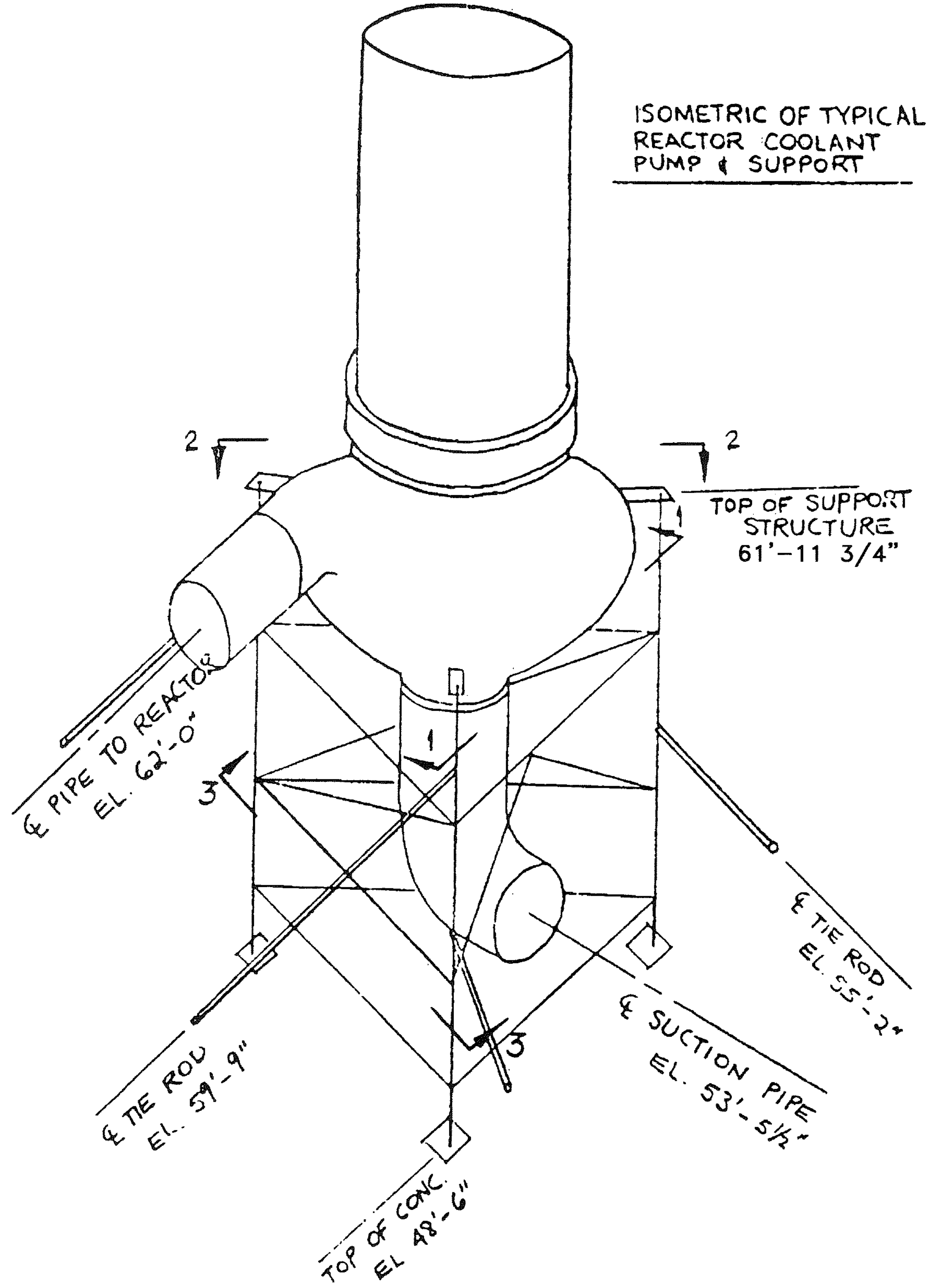


INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.1-43

ISOMETRIC VIEW-STEAM GENERATOR SUPPORT

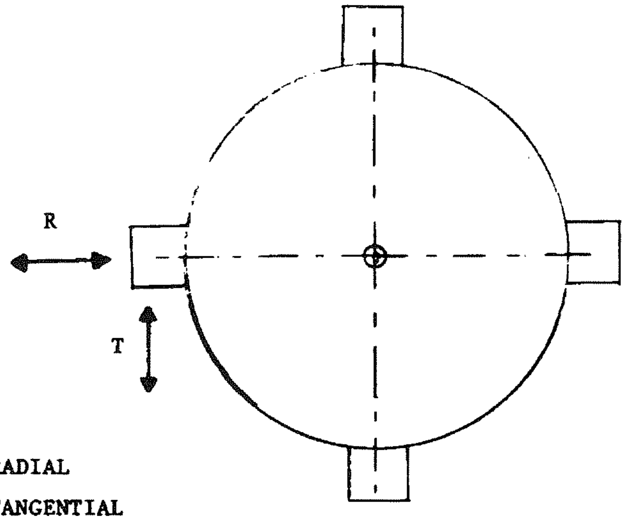
MIC. No. 1999MC3782	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-44 ISOMETRIC VIEW-REACTOR COOLANT PUMP SUPPORT	
MIC. No. 1999MC3783	REV. No. 17A

MAXIMUM FORCES
ACTING ON A REACTOR
VESSEL SUPPORT

	A	B	C	D	Σ		
	REACTOR VESSEL WEIGHT & PIPING REACTION	PIPE BREAK CASE II	PIPE BREAK CASE III	EARTH-QUAKE Z & VERTICAL DIRECTION	A+B	A+C	A+D
P (lb)	934,000	-	525,000	395,000	934,000	1,459,000	1,329,000
R (lb)	322,000	-	-	-	322,000	322,000	322,000
T (lb)	140,000	1,187,000	710,000	969,000	1,327,000	850,000	1,109,000



PIPE BREAK CASE II IS BETWEEN PUMP AND ELL ON REACTOR NOZZLE

PIPE BREAK CASE III IS AT STEAM GENERATOR INLET

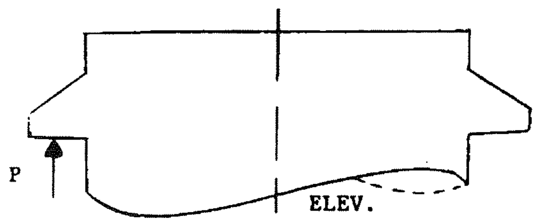
EARTHQUAKE IN Z DIRECTION ACTS NORMAL TO REACTOR COOLANT OUTLET LINE

R-RADIAL

T-TANGENTIAL

P-VERTICAL

PLAN



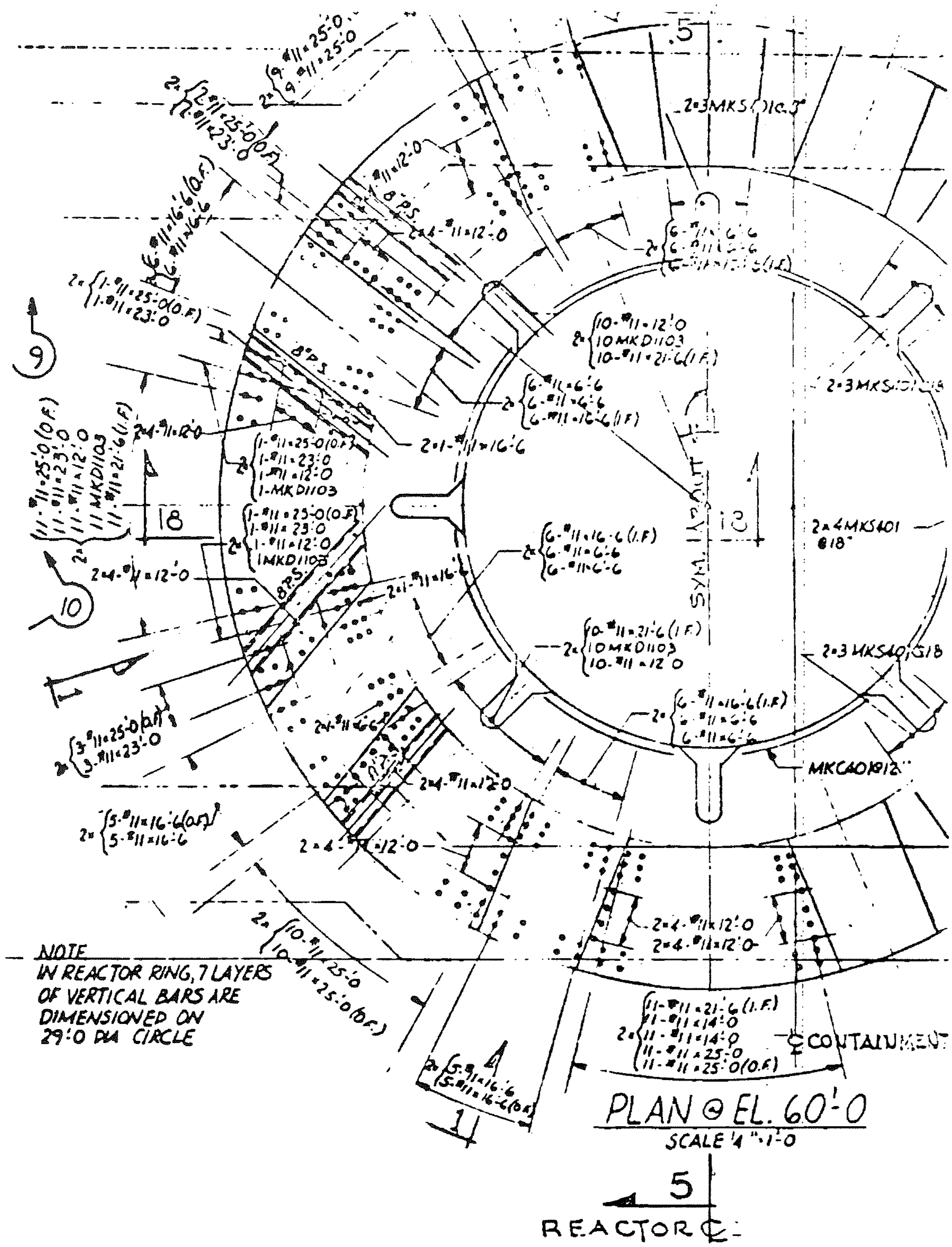
(THIS FIGURE DEPICTS ORIGINAL HISTORICAL DESIGN DATA AND DOES NOT REFLECT THE CURRENT CONFIGURATION OR ANALYSIS.)

INDIAN POINT UNIT No. 2

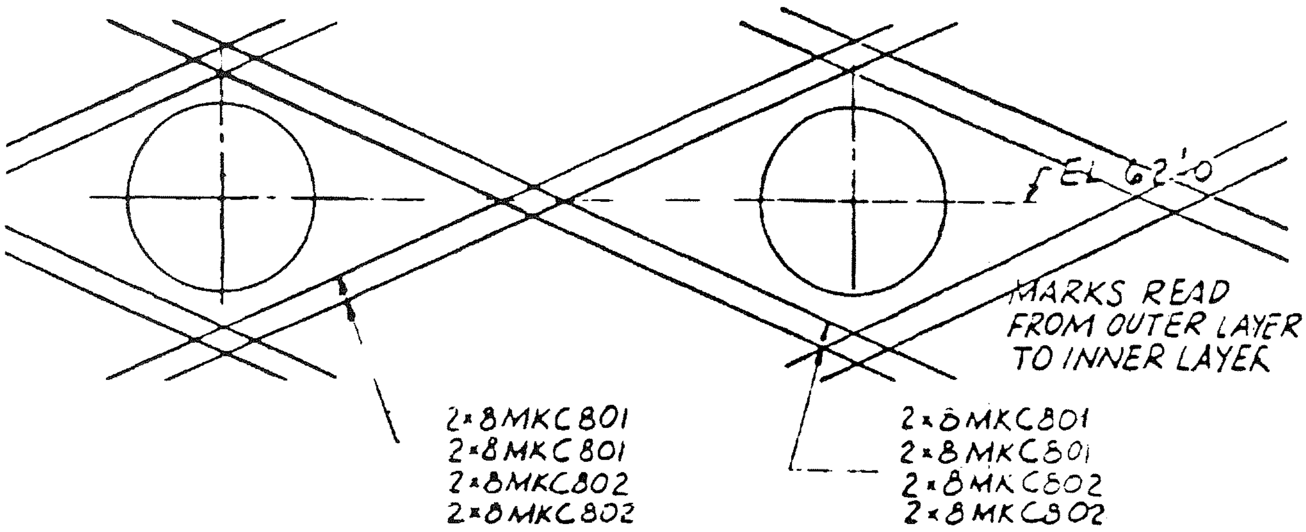
UFSAR FIGURE 5.1-45

MAXIMUM FORCES ACTING ON A REACTOR VESSEL SUPPORT

MIC. No. 1999MC3784 | REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1–46	
PLAN VIEW 60 FT – 0 IN.	
MIC. No. 1999MC3785	REV. No. 17A



TYP. LAYER @ REACTOR RING

SCALE 1/4" = 1'-0

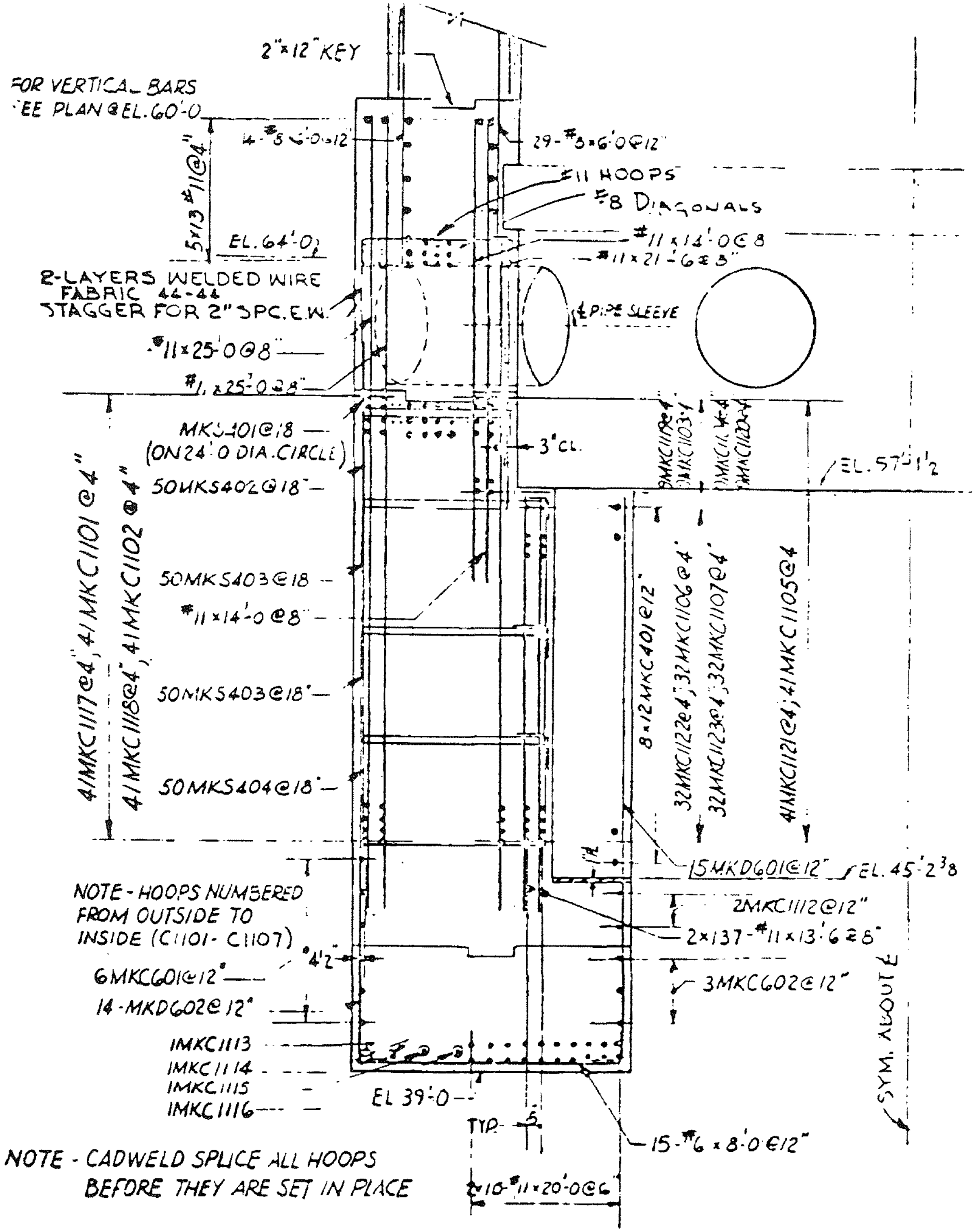
SECTION 1-1

INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.1-47

TYPICAL
LAYER-REACTOR RING

MIC. No. 1999MC3786 | REV. No. 17A

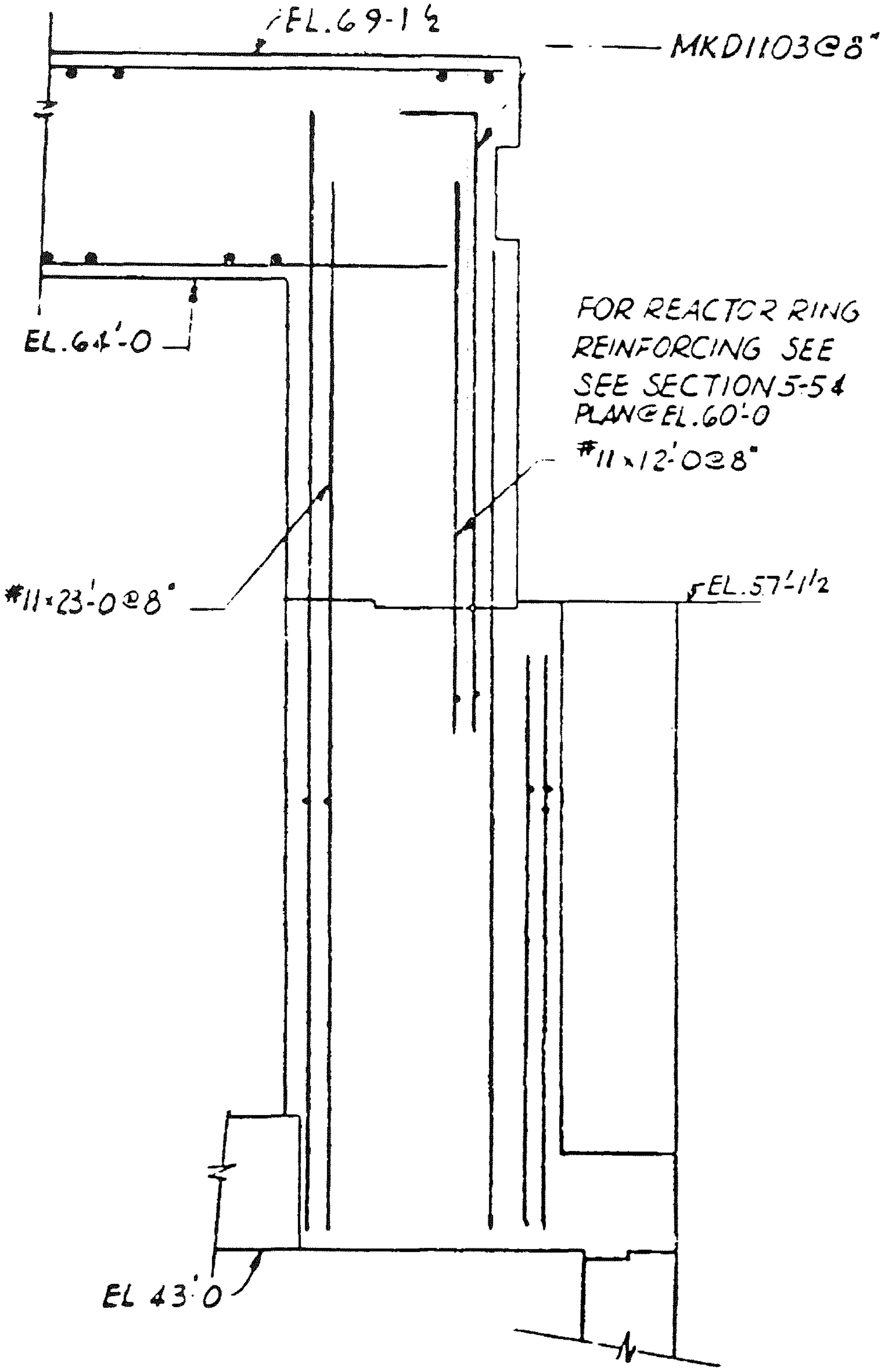


INDIAN POINT UNIT No. 2

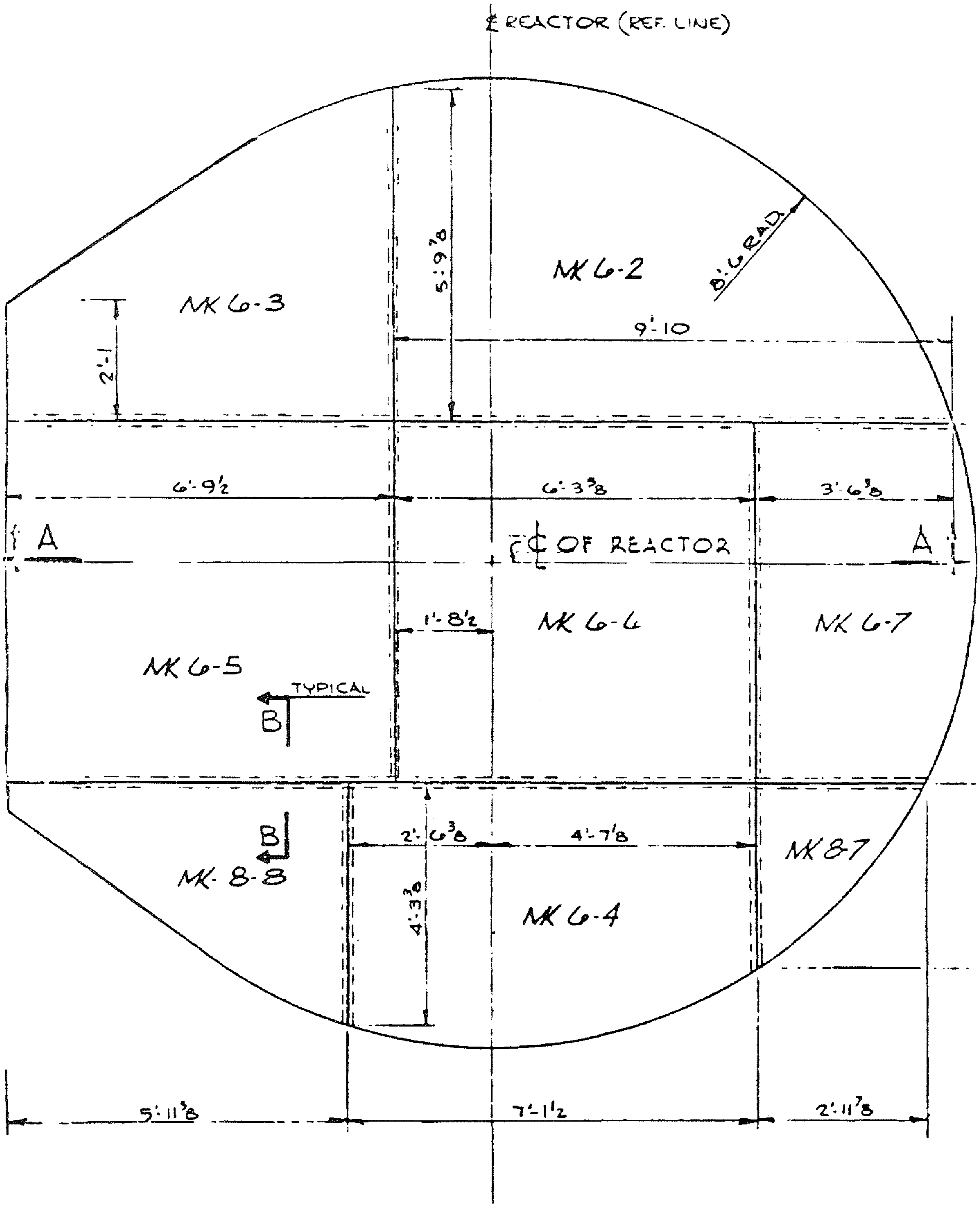
UFSAR FIGURE 5.1-48

SECTION 5-5

MIC. No. 1999MC3787	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-49	
SECTION 18-18	
MIC. No. 1999MC3788	REV. No. 17A

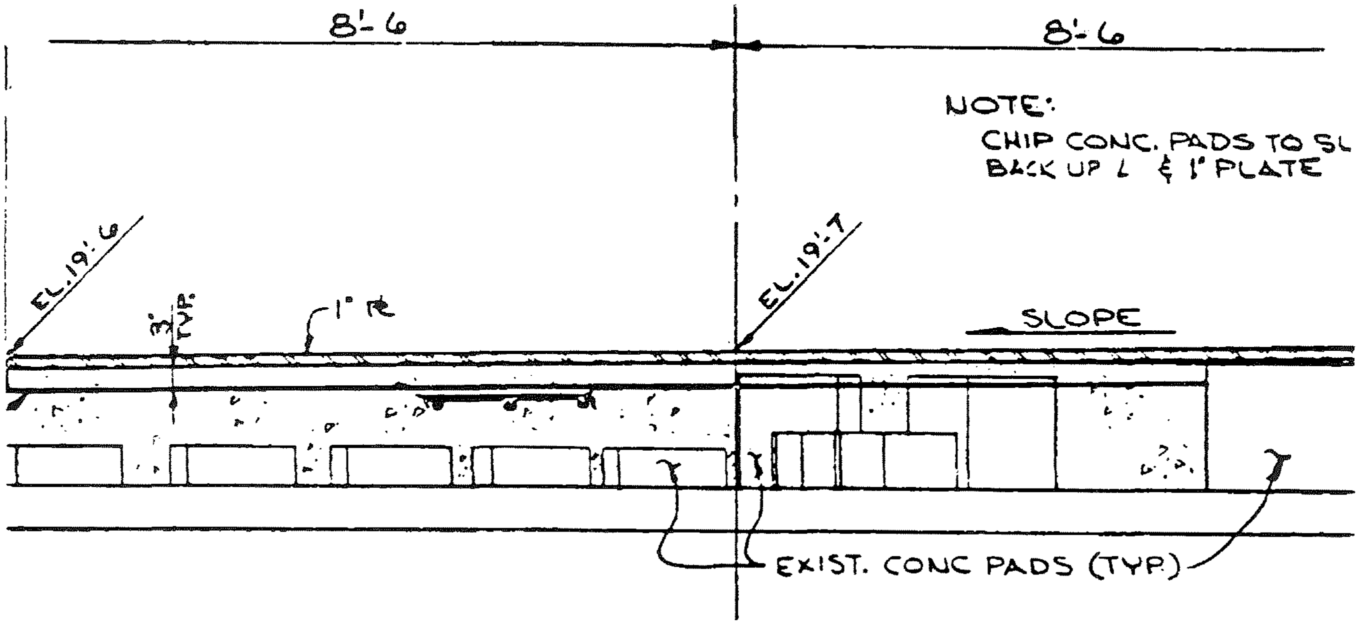


INDIAN POINT UNIT No. 2

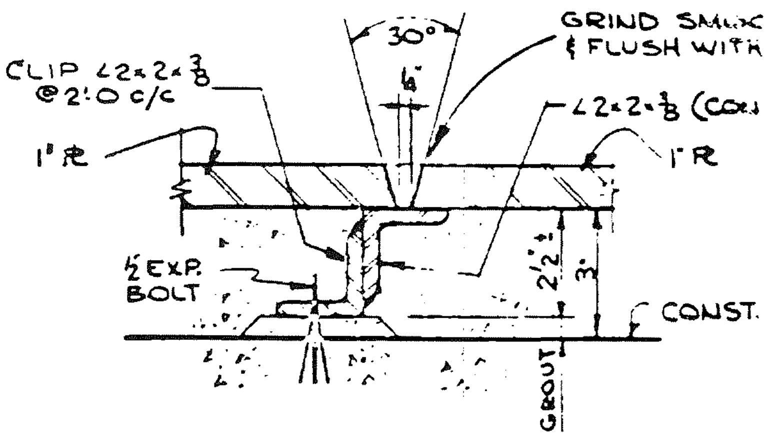
UFSAR FIGURE 5.1-50

PLAN VIEW AT ELEVATION 19 FT.-7 IN.

MIC. No. 1999MC3789	REV. No. 17A

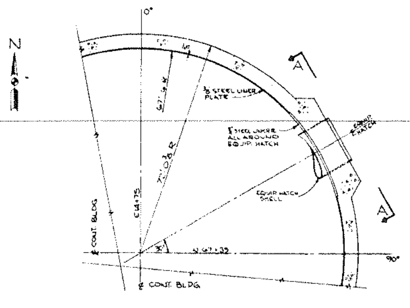


SECTION A-A



SECTION B-B

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-51 SECTION A-A AND SECTION B-B	
MIC. No. 1999MC3790	REV. No. 17A



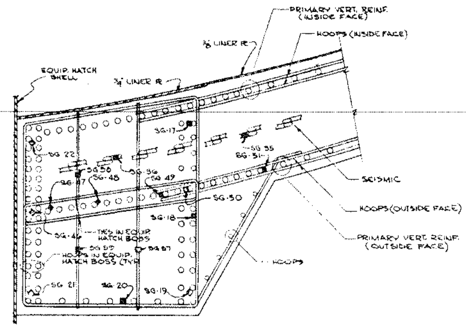
KEY PLAN

SCALE: 1/16" = 1'-0"

— CONTINUED FROM BELOW —

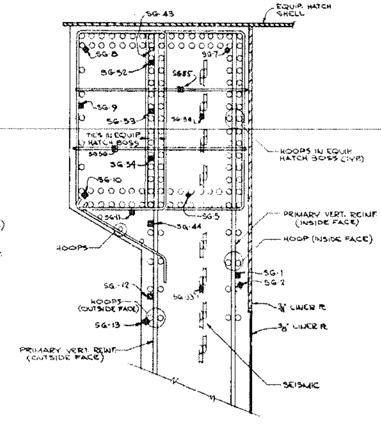
STRAIN GAUGE SCHEDULE

MARK Nº	* HORIZONTAL LOCATION LEFT	* HORIZONTAL LOCATION RIGHT	VERTICAL LOCATION (ELEVATION)	RESPECTIVE STRUCTURAL TYPE	REMARKS
SG-43	0'-0"	0'-0"	EL. 92'-6"	HOOP IN EQUIP HATCH BOSS	
SG-44	0'-0"	0'-0"	EL. 85'-4"	HOOP IN EQUIP HATCH BOSS	
SG-46	—	8'-7"	EL. 101'-6"	HOOP IN EQUIP HATCH BOSS	
SG-47	—	9'-6"	EL. 101'-8"	PRIMARY VERT (CENTER OF BOSS)	
SG-48	—	11'-7"	EL. 101'-1"	PRIMARY VERT (CENTER BOSS)	
SG-49	—	14'-1"	EL. 101'-6"	PRIMARY VERT (CENTER OF BOSS)	
SG-50	—	15'-2"	EL. 101'-5"	PRIMARY VERT (CENTER OF BOSS)	
SG-51	—	18'-5"	EL. 101'-11"	PRIMARY VERT (OUTSIDE FACE)	
SG-52	0'-0"	0'-3"	EL. 92'-0"	HOOP (CENTER OF EQUIP HATCH BOSS)	
SG-53	0'-0"	0'-7"	EL. 90'-5"	HOOP (CENTER OF EQUIP HATCH BOSS)	
SG-54	0'-0"	0'-8"	EL. 88'-4"	HOOP (CENTER OF EQUIP HATCH BOSS)	
SG-55	0'-0"	1'-1"	EL. 91'-6"	RADIAL SHEAR BAR	4'-5 FROM LINER PL
SG-56	0'-5"	0'-0"	EL. 88'-9"	"	5'-6 FROM LINER PL
SG-57	—	12'-2"	EL. 100'-10"	"	6'-9 FROM LINER PL
SG-58	—	7'-10"	EL. 101'-3"	"	3'-0 FROM LINER PL
SG-59	—	10'-1"	EL. 101'-4"	"	5'-3 FROM LINER PL



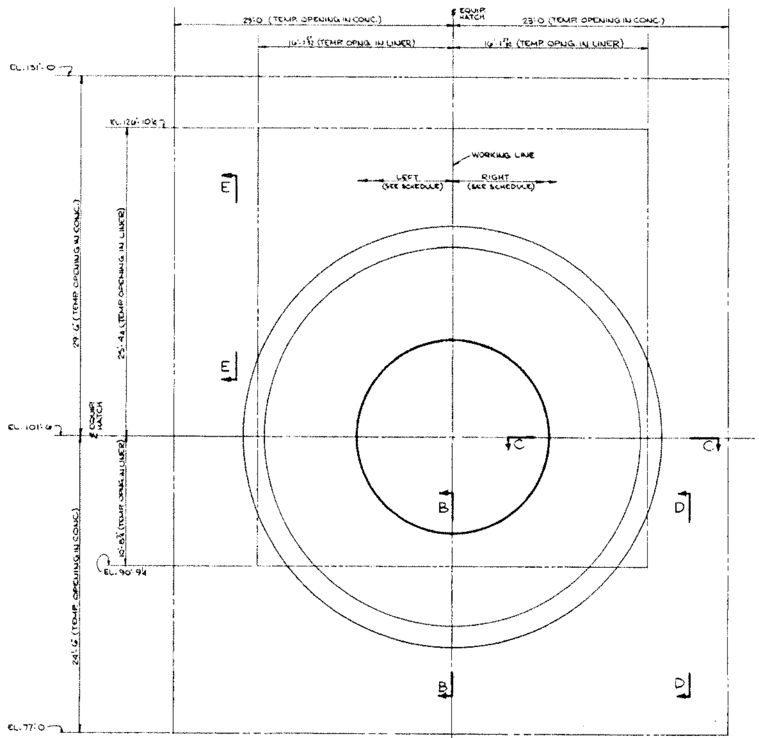
SECTION C-C

SCALE: 1/2" = 1'-0"



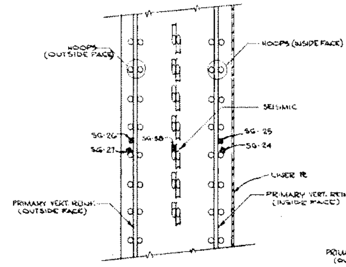
SECTION B-B

SCALE: 1/2" = 1'-0"



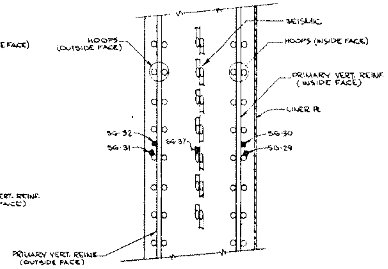
SECTION A-A (DEVELOPED ELEVATION)

SCALE: 1/4" = 1'-0"



SECTION D-D

SCALE: 1/2" = 1'-0"



SECTION E-E

SCALE: 1/2" = 1'-0"

STRAIN GAUGE SCHEDULE

MARK Nº	* HORIZONTAL LOCATION LEFT	* HORIZONTAL LOCATION RIGHT	VERTICAL LOCATION (ELEVATION)	RESPECTIVE STRUCTURAL TYPE	REMARKS
SG-1	0'-0"	0'-0"	EL. 83'-4"	PRIMARY VERT (INSIDE FACE)	
SG-2	0'-0"	0'-0"	EL. 82'-9"	HOOP (INSIDE FACE)	
SG-5	0'-0"	0'-0"	EL. 87'-3"	HOOP IN EQUIPMENT HATCH BOSS	2'-7 FROM LINER PL
SG-7	0'-0"	0'-0"	EL. 92'-5"	HOOP IN EQUIPMENT HATCH BOSS	
SG-8	0'-0"	0'-0"	EL. 92'-6"	HOOP IN EQUIPMENT HATCH BOSS	
SG-9	0'-0"	0'-0"	EL. 90'-2"	TIE IN EQUIPMENT HATCH BOSS	
SG-10	0'-0"	0'-0"	EL. 86'-5"	HOOP IN EQUIPMENT HATCH BOSS	
SG-11	0'-7"	0'-0"	EL. 85'-11"	TIE IN EQUIPMENT HATCH BOSS	
SG-12	0'-0"	0'-0"	EL. 82'-7"	PRIMARY VERT. (OUTSIDE FACE)	
SG-13	0'-0"	0'-0"	EL. 82'-1"	HOOP (OUTSIDE FACE)	
SG-17	—	15'-3"	EL. 101'-6"	HOOP IN EQUIPMENT HATCH BOSS	
SG-18	—	15'-4"	EL. 100'-10"	TIE IN EQUIPMENT HATCH BOSS	
SG-19	—	15'-2"	EL. 101'-0"	HOOP IN EQUIPMENT HATCH BOSS	
SG-20	—	12'-6"	EL. 100'-5"	TIE IN EQUIPMENT HATCH BOSS	
SG-21	—	8'-10"	EL. 100'-8"	HOOP IN EQUIPMENT HATCH BOSS	
SG-22	—	8'-7"	EL. 101'-6"	HOOP IN EQUIPMENT HATCH BOSS	
SG-24	—	21'-0"	EL. 82'-9"	HOOP (INSIDE FACE)	
SG-25	—	21'-0"	EL. 83'-4"	PRIMARY VERT. (INSIDE FACE)	
SG-26	—	22'-3"	EL. 82'-7"	PRIMARY VERT. (OUTSIDE FACE)	
SG-27	—	22'-0"	EL. 82'-4"	HOOP (OUTSIDE FACE)	
SG-29	21'-0"	—	EL. 120'-3"	HOOP (INSIDE FACE)	
SG-30	21'-0"	—	EL. 120'-10"	PRIMARY VERT. (INSIDE FACE)	
SG-31	21'-7"	—	EL. 120'-9"	HOOP (OUTSIDE FACE)	
SG-32	21'-5"	—	EL. 121'-2"	PRIMARY VERT. (OUTSIDE FACE)	
SG-33	0'-0"	0'-0"	EL. 82'-9"	SEISMIC	
SG-34	0'-0"	0'-0"	EL. 89'-9"	SEISMIC	
SG-35	—	13'-10"	EL. 101'-6"	SEISMIC	
SG-36	—	17'-0"	EL. 101'-6"	SEISMIC	
SG-37	21'-0"		EL. 120'-3"	SEISMIC	
SG-38	—	21'-0"	EL. 82'-9"	SEISMIC	

— FOR CONTINUATION OF STRAIN GAUGE SCHEDULE SEE ABOVE —

* FOR CORRECT ORIENTATION OF LEFT-RIGHT DIRECTIONS SEE WORKING LINE ON SECTION A-A

GENERAL NOTES

1. WHERE STRAIN GAUGES AND RESPECTIVE STRUCTURAL TYPE LOCATIONS DO NOT COINCIDE ACCORDING TO GIVEN DIMENSIONS AND/OR GIVEN ELEVATIONS, PLACE STRAIN GAUGE ON NEAREST STRUCTURAL ELEMENT OF THE SAME TYPE.

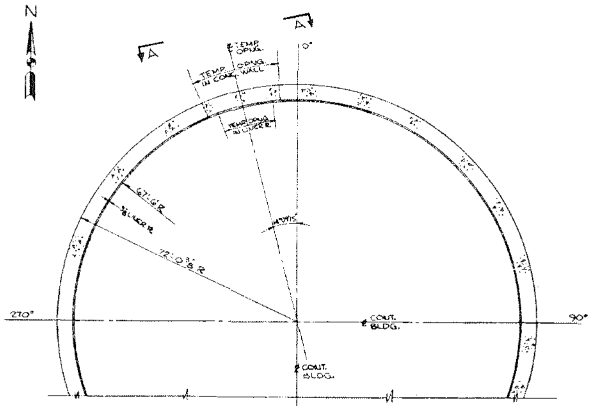
INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.1-53

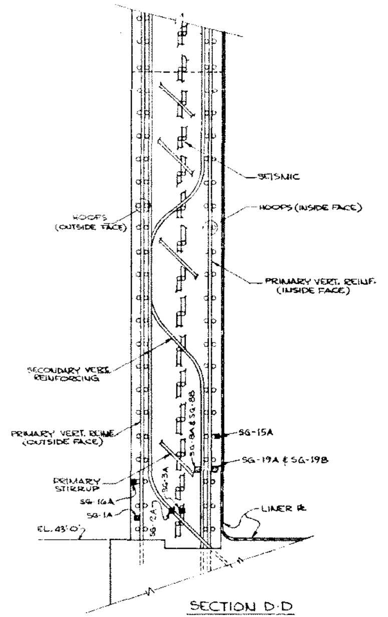
CONTAINMENT EQUIPMENT HATCH STRAIN GAUGE TEST LOCATIONS

MIC. No. 1999MC3792

REV. No. 17A

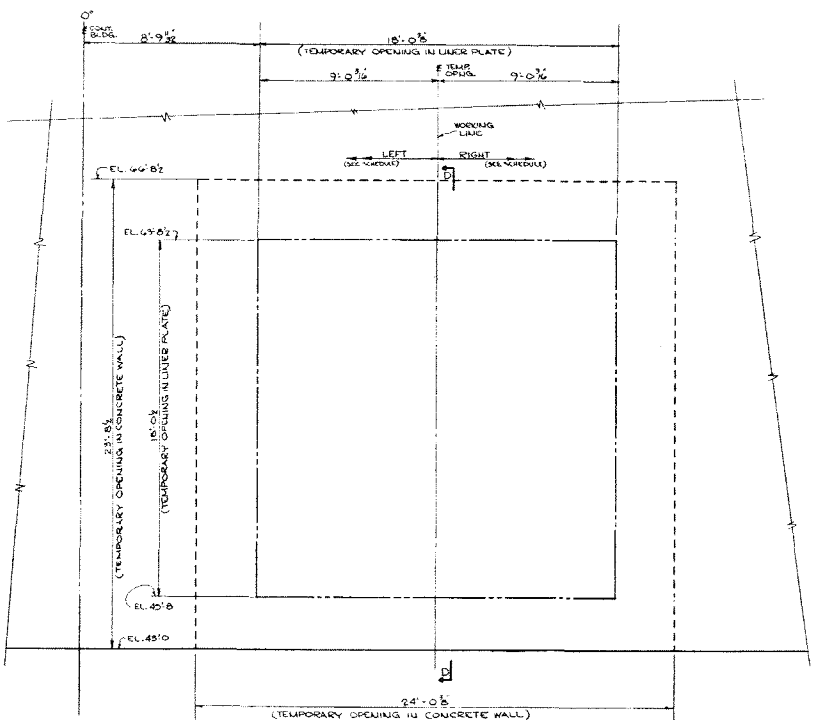


KEY PLAN
SCALE: 1/16"=1'-0"



SECTION D-D

1/1 REDUNDANCY REQUIRED ON GUAGES SG-8A & SG-19A



SECTION A-A (DEVELOPED ELEVATION)
SCALE: 3/8"=1'-0"

STRAIN GAUGE SCHEDULE					
MARK Nº	* HORIZONTAL LOCATION		VERTICAL LOCATION (ELEVATION)	RESPECTIVE STRUCTURAL TYPE	REMARKS
	LEFT	RIGHT			
SG-1A		0'-6"	EL. 44'-2	PRIMARY VERT. (OUTSIDE FACE)	
SG-2A	——	0'-6"	EL. 44'-2	SECONDARY VERTICAL	
SG-3A	——	0'-6"	EL. 44'-2	SEISMIC	
SG-8A	——	0'-6"	EL. 46'-6	SECONDARY VERTICAL	
SG-8B		0'-6	EL. 46'-6	SECONDARY VERTICAL	"REDUNDANT GUAGE"
SG-15A	——	0'-6	EL. 48'-0	HOOP (INSIDE FACE)	
SG-16A	——	0'-6	EL. 45'-11	HOOP (OUTSIDE FACE)	
SG-19A	——	0'-6	EL. 46'-6	PRIMARY VERT. (INSIDE FACE)	
SG-19B		0'-6	EL. 46'-6	PRIMARY VERT (INSIDE FACE)	"REDUNDANT GUAGE"

* FOR CORRECT ORIENTATION OF *LEFT & RIGHT* DIRECTIONS SEE *WORKING LINE* ON SECTION A-A

GENERAL NOTES

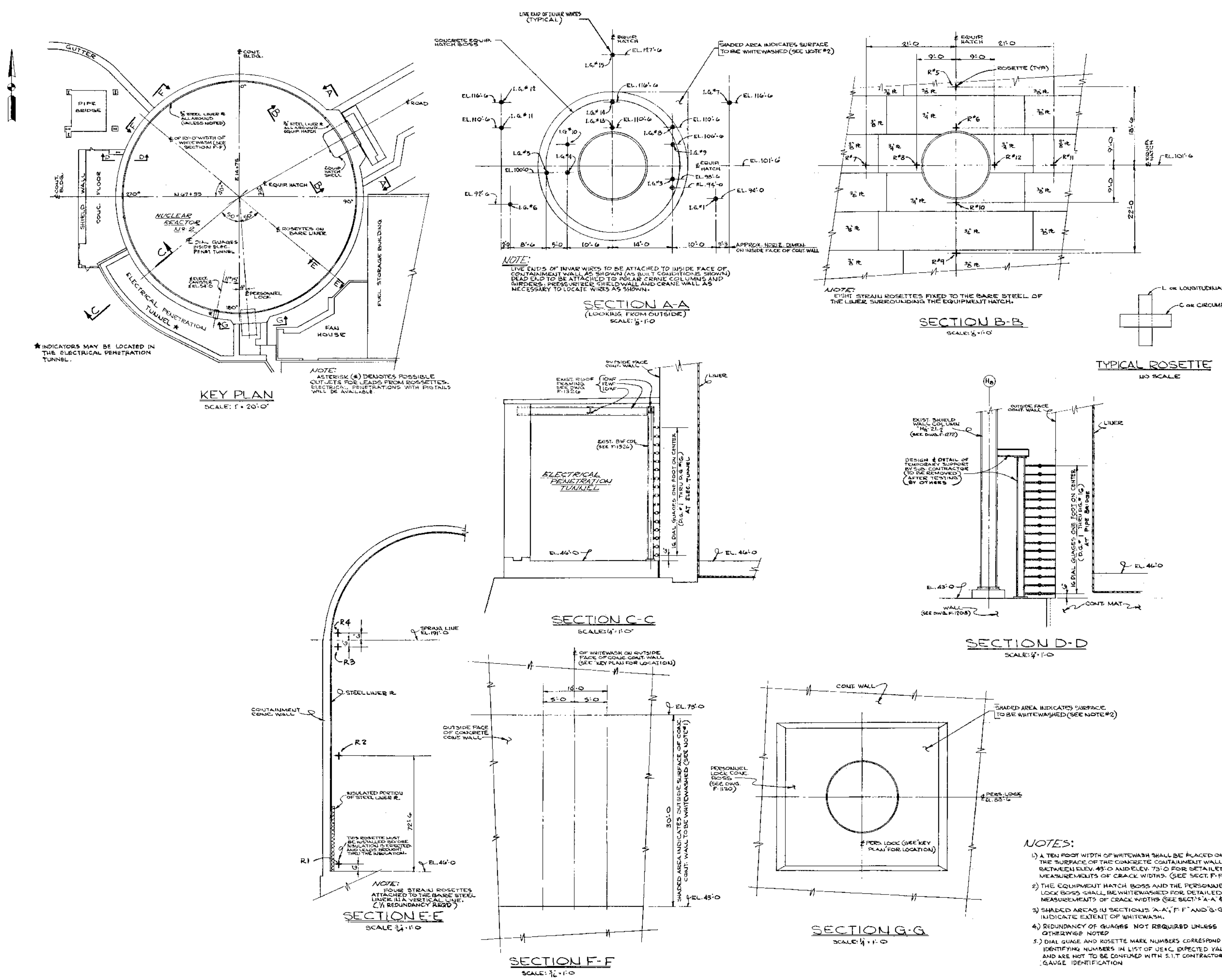
1- WHERE STRAIN GAUGES AND RESPECTIVE STRUCTURAL TYPE LOCATIONS DO NOT COINCIDE ACCORDING TO GIVEN DIMENSIONS AND/OR GIVEN ELEVATIONS, PLACE STRAIN GAUGE ON NEAREST STRUCTURAL ELEMENT OF THE SAME TYPE.

INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.1-54

CONTAINMENT TEMPORARY OPENING IN NW QUADRANT STRAIN GAUGE TEST LOCATIONS

MIC. No. 1999MC3793 | REV. No. 17A



INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.1–55

CONTAINMENT STRAIN GAUGE TEST LOCATIONS

MIC. No. 1999MC3794 | REV. No. 17A

NOTES:

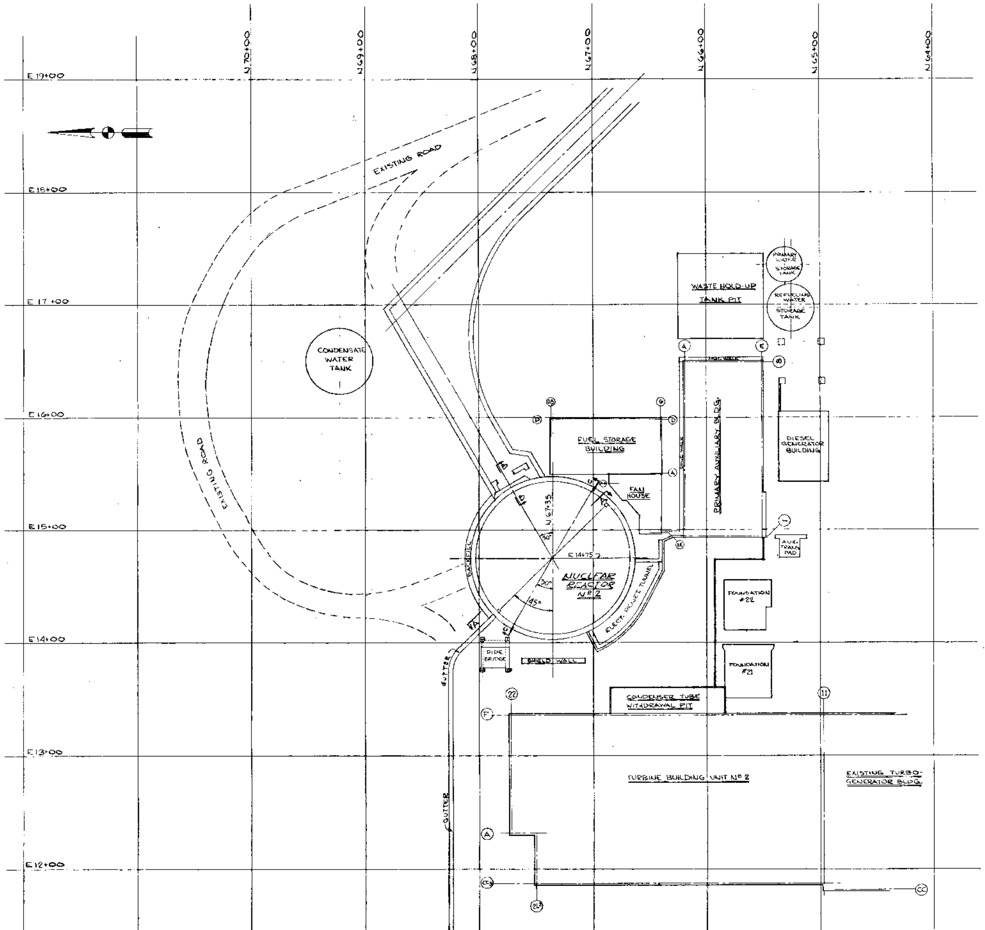
1.) A TEN FOOT WIDTH OF WHITEWASH SHALL BE PLACED ON THE SURFACE OF THE CONCRETE CONTAINMENT WALL BETWEEN ELEV. 43'-0 AND ELEV. 73'-0 FOR DETAILED MEASUREMENTS OF CRACK WIDTHS. (SEE SECT. F-F)

2.) THE EQUIPMENT HATCH BOSS AND THE PERSONNEL LOCK BOSS SHALL BE WHITEWASHED FOR DETAILED MEASUREMENTS OF CRACK WIDTHS (SEE SECT.'S "A-A" & "G-G")

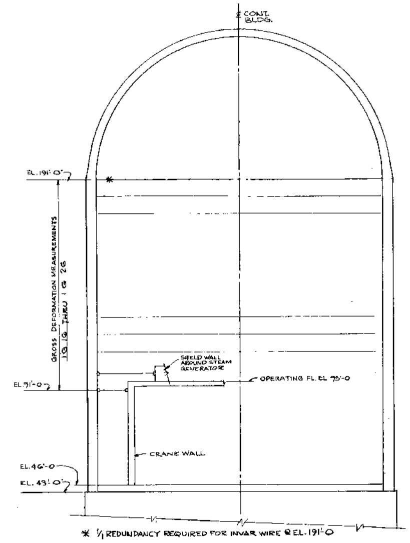
3.) SHADED AREAS IN SECTIONS "A-A", "F-F" AND "G-G" (ONLY) INDICATE EXTENT OF WHITEWASH.

4.) REDUNDANCY OF GUAGES NOT REQUIRED UNLESS OTHERWISE NOTED

5.) DIAL GUAGE AND ROSETTE MARK NUMBERS CORRESPOND TO IDENTIFYING NUMBERS IN LIST OF UE&C EXPECTED VALUES AND ARE NOT TO BE CONFUSED WITH S.I.T CONTRACTOR'S GAUGE IDENTIFICATION

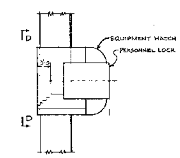


PLAN
SCALE: 1" = 40'-0"

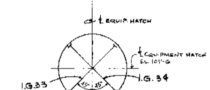


✱ 1/1 REDUNDANCY REQUIRED FOR INVAR WIRE @ EL. 191'-0

SECTION A-A
SCALE: 3/64" = 1'-0



SECTION B-B
SCALE: 3/32" = 1'-0

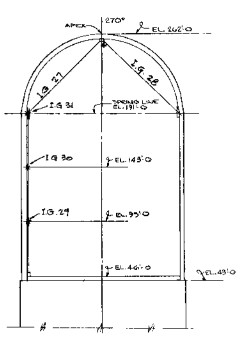


SECTION D-D
SCALE: 3/32" = 1'-0

NOTES

1. PROOF TEST GROSS DEFORMATION INSTRUMENTATION SHALL BE LOCATED BETWEEN ELEVATIONS SHOWN IN SECTION "A-A" AT EVERY TEN FEET ALONG VERTICAL PLANE OF CONTAINMENT BUILDING - AS SHOWN IN PLAN.
2. ALL GROSS DEFORMATION MEASUREMENTS SHALL BE OBTAINED BY AN INVAR WIRE EXTENSOMETER WITH A LINEAR POTENTIOMETER SENSING DEVICE (19 REQ'D).
3. ALL NECESSARY ATTACHMENTS TO THE LINER OR EQUIPMENT HATCH ARE BY OTHERS.
4. LOCATIONS GIVEN INDICATE THE DESIRED POSITION OF INVAR TAPE. IF INTERFERENCE IS ENCOUNTERED INSIDE THE CONTAINMENT BUILDING TAPE SHALL BE LOCATED AS CLOSE TO GIVEN POSITION AS POSSIBLE.
5. TOTAL TRAVEL OF POTENTIOMETER SHALL BE 2" MINIMUM EXCEPT FOR VERTICAL MEASUREMENTS AT EL 95'-0, EL 143'-0 AND TOP OF DOME, RADIUS CHANGE AT EL. 91'-0 AND CONCRETE OPENING DIAMETER CHANGE AT EQUIPMENT HATCH WHERE POTENTIOMETERS WITH 1.5" TRAVEL SHALL BE USED.
6. FIELD SHALL MEASURE AND RECORD RADIUS AT EACH POINT WHERE INVAR TAPES IN SECTION A-A ARE ATTACHED TO THE LINER.
7. FIELD SHALL INSTALL THERMOCOUPLES ON OUTSIDE OF CONTAINMENT SHELL. THEY SHALL BE EMBEDDED IN CONCRETE SO NOT EXPOSED TO DIRECT RADIATION FROM SUN. AS A MINIMUM ONE (1) THERMOCOUPLE SHALL BE INSTALLED IN CYLINDER WALL BELOW AND ABOVE INSULATION AND IN THE DOME.
8. STRUCTURAL INTEGRITY TEST CONTRACTOR SHALL SUPPLY RECORDING INSTRUMENTATION NECESSARY FOR THERMOCOUPLES.

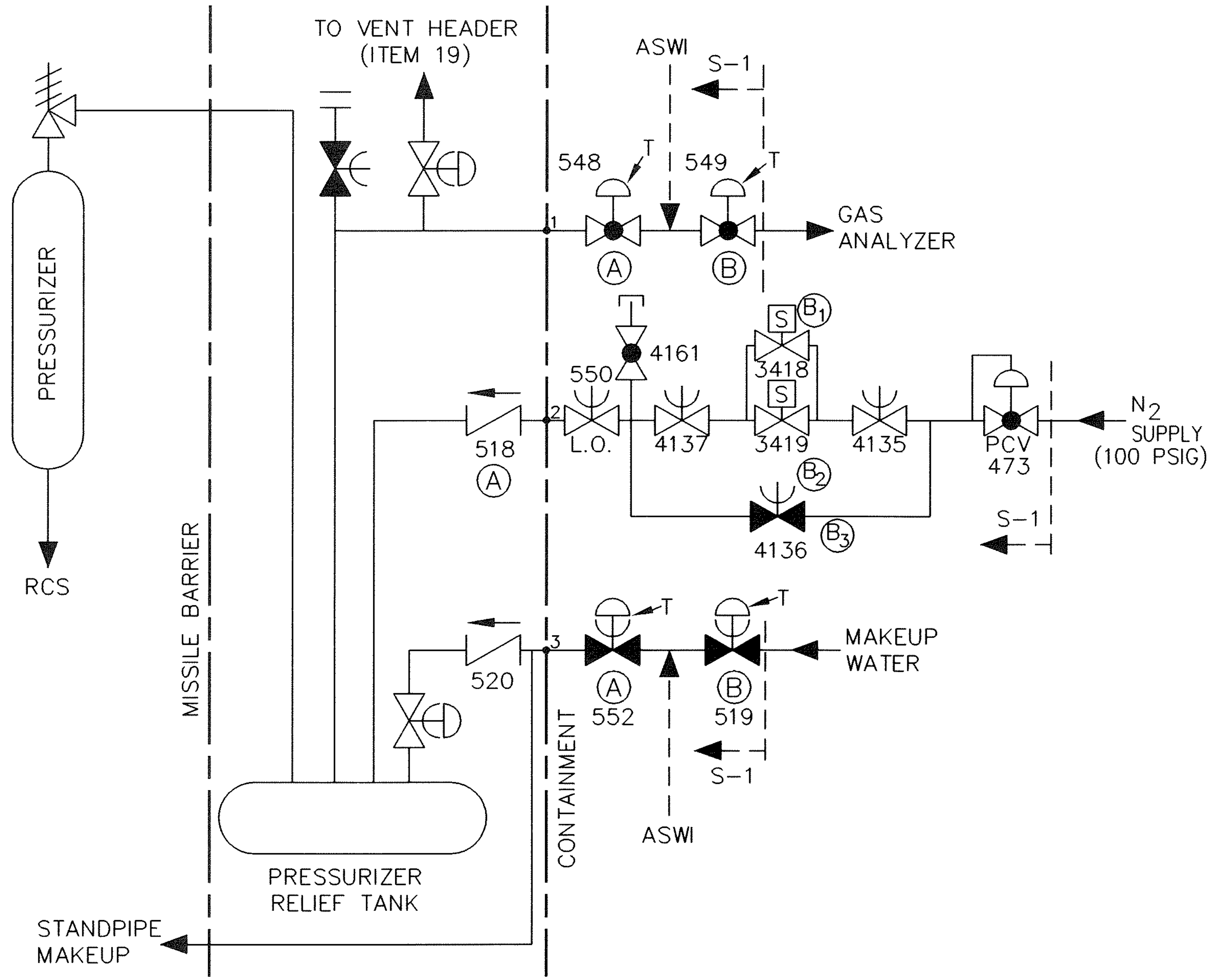
VERTICAL MEASUREMENT READINGS SHALL BE OBTAINED AT ELEVATIONS MARKED WITH AN ASTERISK (✱). THE TAPE SHALL BE ATTACHED TO THE LINER BY OTHERS AND ANCHORED TO THE MAT AT EL. 46'-0.



SECTION C-C
SCALE: 1" = 40'-0"

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.1-56	
CONTAINMENT PROOF TEST GROSS DEFORMATION MEASUREMENTS	
MIC. No. 1999MC3795	REV. No. 17A

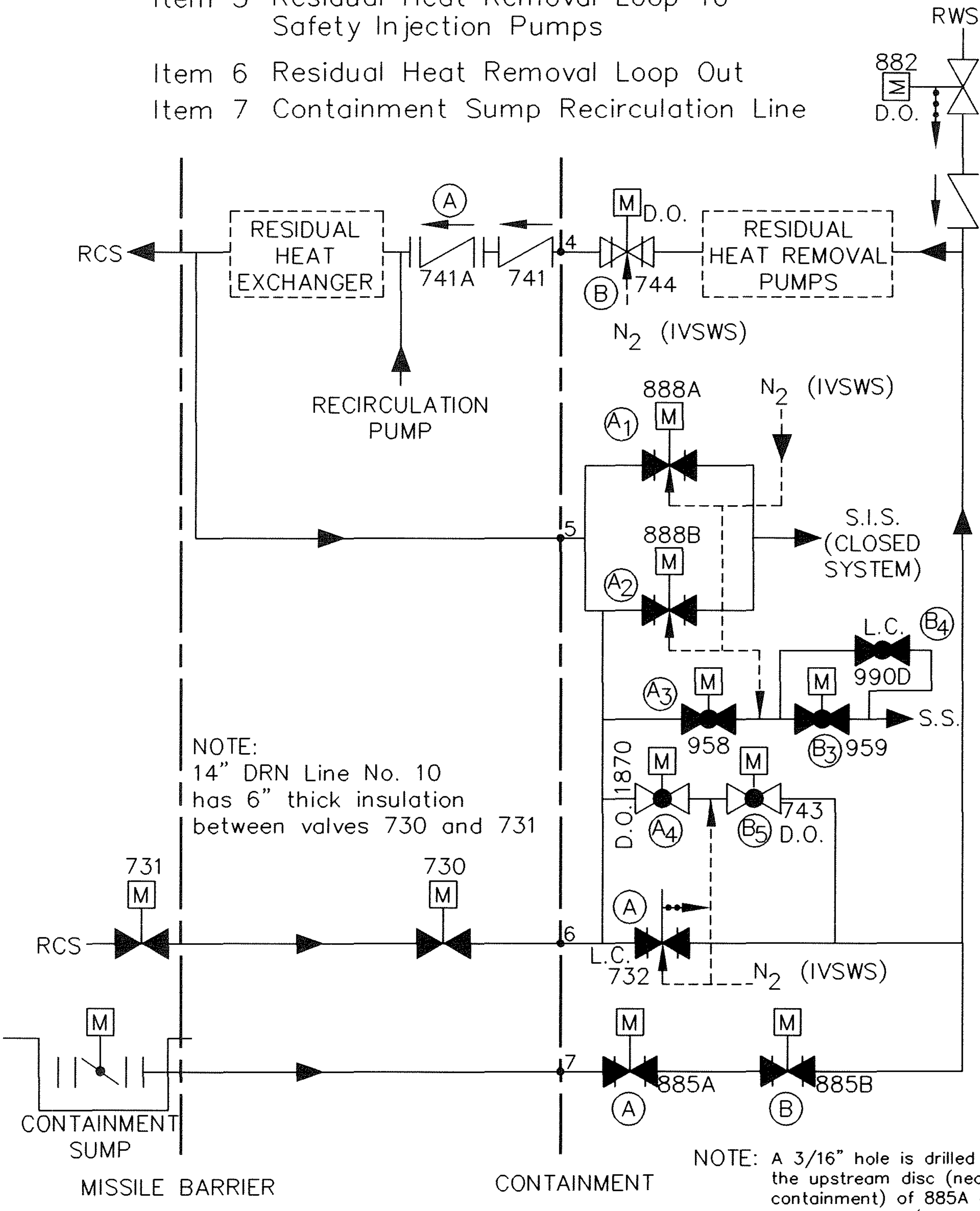
Item 1 Pressurizer Relief Tank to Gas Analyzer
Item 2 Pressurizer Relief Tank N_2 Supply
Item 3 Pressurizer Relief Tank Makeup



Although the Pressurizer Relief Tank is missile protected, these penetrating lines can become exposed to containment atmosphere if the pressurizer discharge header is breached during the accident.

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2-1 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3411	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

Item 4 Residual Heat Removal Return
Item 5 Residual Heat Removal Loop To Safety Injection Pumps
Item 6 Residual Heat Removal Loop Out
Item 7 Containment Sump Recirculation Line



NOTE: A 3/16" hole is drilled through the upstream disc (nearest containment) of 885A and the downstream disc (RHR Loop Side) of 885B to prevent pressure locking.

N_2 – Manual N_2 Pressurization
SS – Sampling System

ENTIRE SYSTEM SHOWN IS SEISMIC CLASS I DESIGN

INDIAN POINT UNIT No. 2

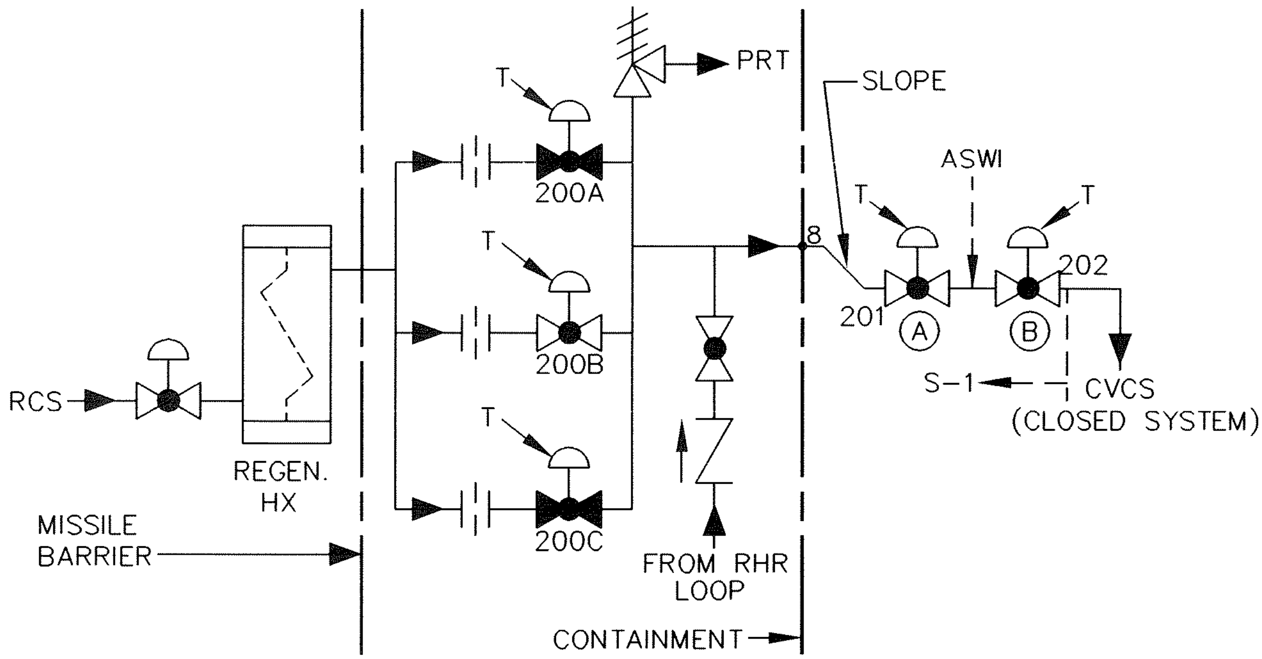
UFSAR FIGURE 5.2–2

CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS

MIC. No. 1999MC3382	REV. No. 17A

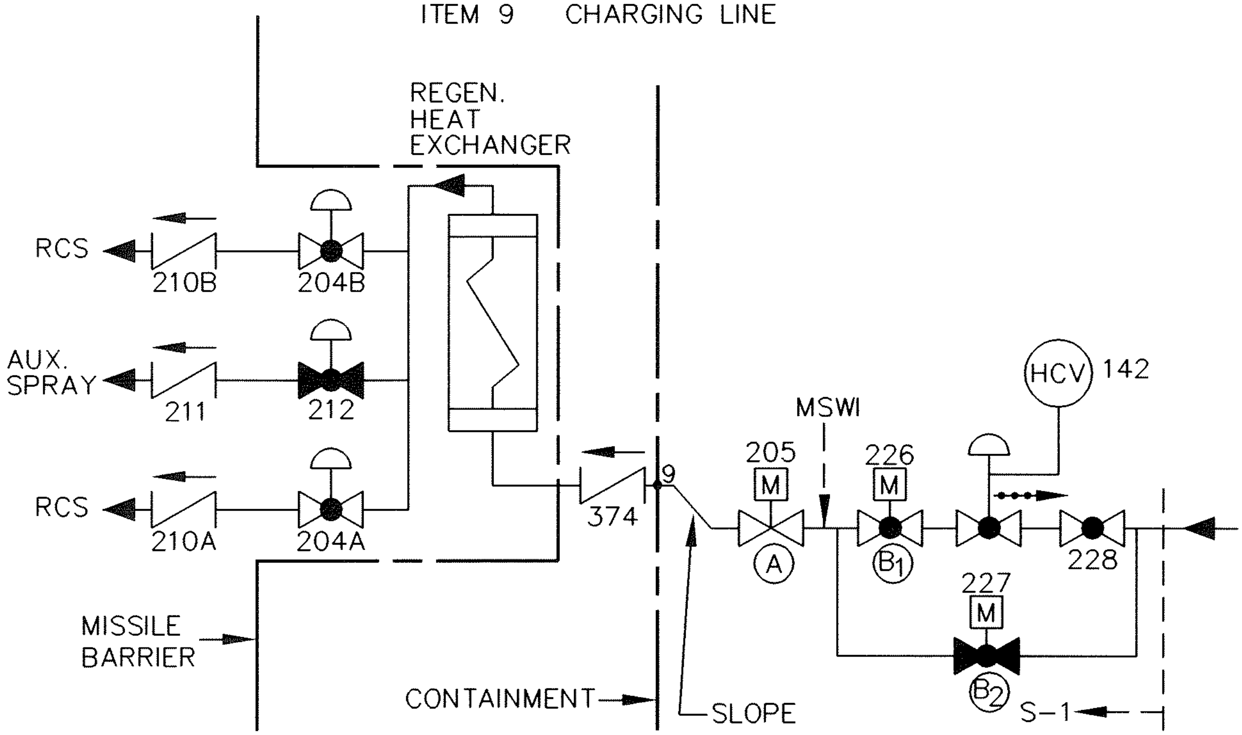
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE

ITEM 8 LETDOWN LINE



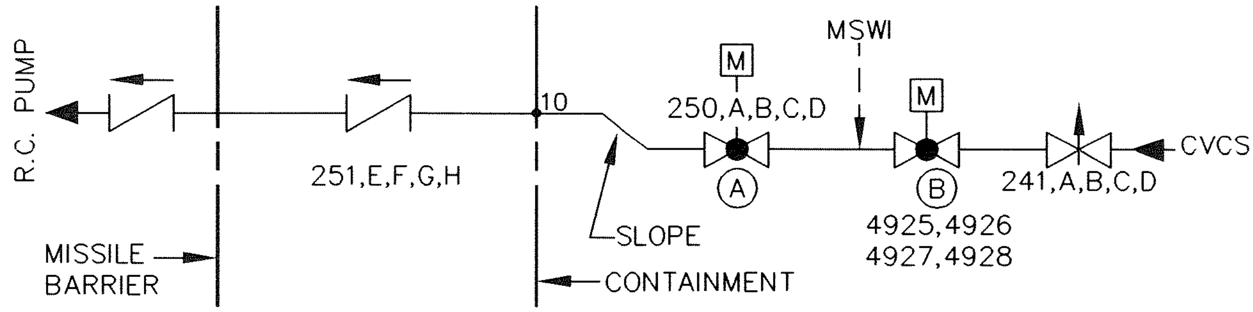
PRT – PRESSURIZER RELIEF TANK

ITEM 9 CHARGING LINE



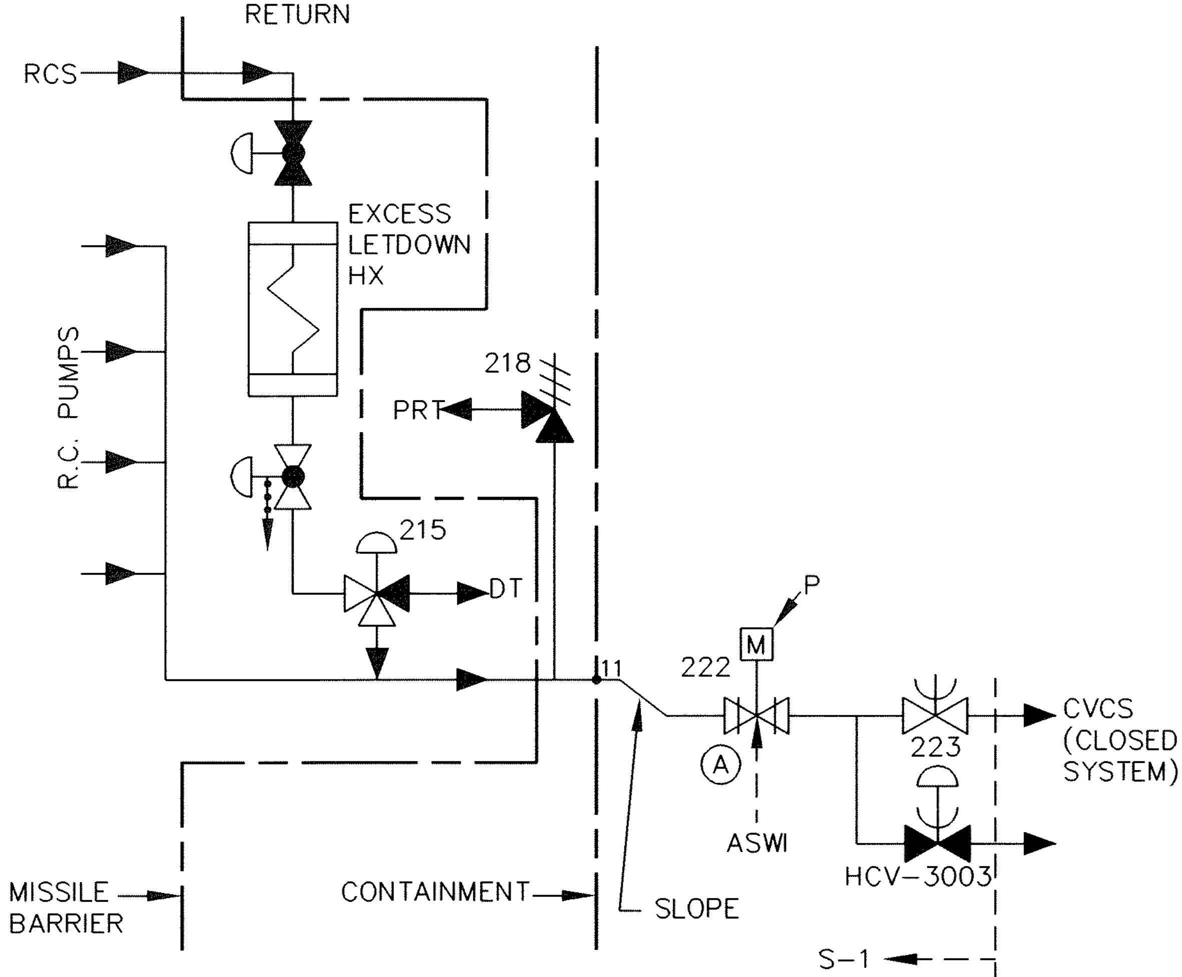
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2-3 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3383	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

ITEM 10 REACTOR COOLANT PUMP SEAL-WATER SUPPLY LINES



ENTIRE SYSTEM SHOWN IS SEISMIC CLASS 1 DESIGN

ITEM 11 REACTOR COOLANT PUMP SEAL-WATER RETURN



DT – REACTOR COOLANT DRAIN TANK

P – TRIPPED CLOSED BY CONTAINMENT ISOLATION SIGNAL PHASE B

PRT – PRESSURIZER RELIEF TANK

INDIAN POINT UNIT No. 2

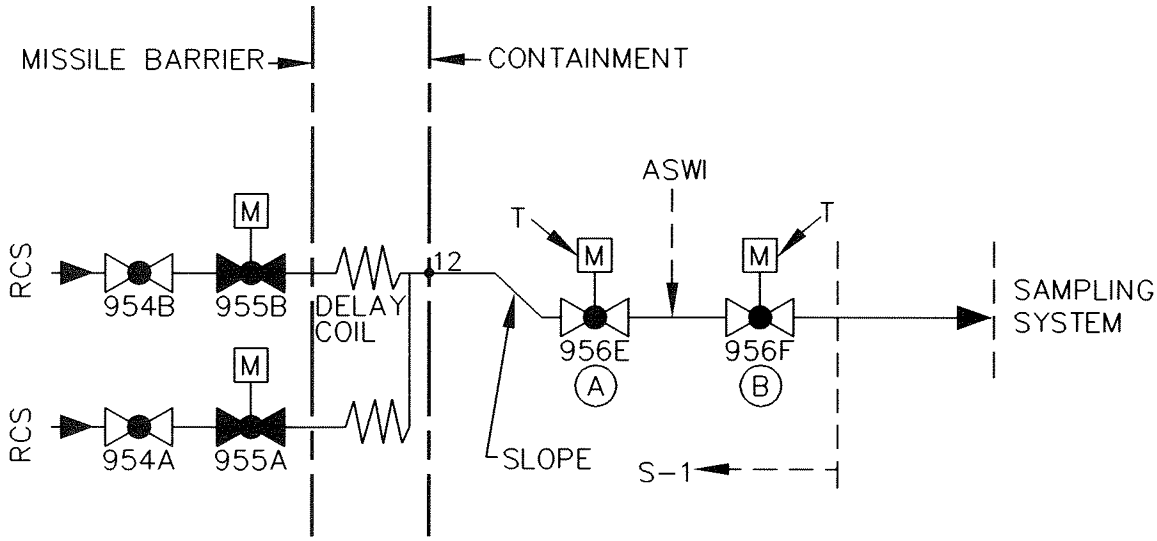
UFSAR FIGURE 5.2-4

CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS

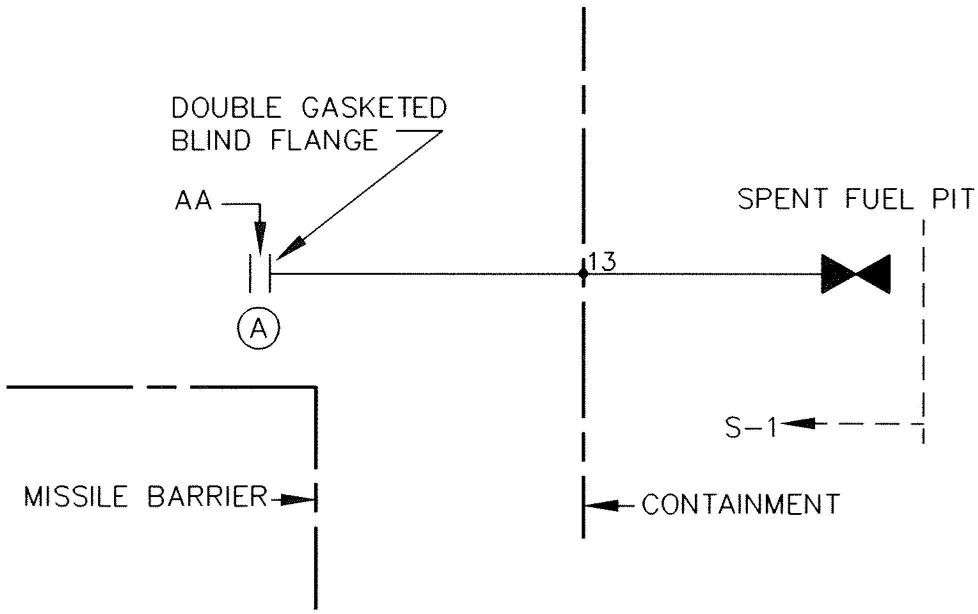
MIC. No. 1999MC3385	REV. No. 17A

VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE

ITEM 12 REACTOR COOLANT SYSTEM SAMPLE LINES



ITEM 13 FUEL TRANSFER TUBE



INDIAN POINT UNIT No. 2

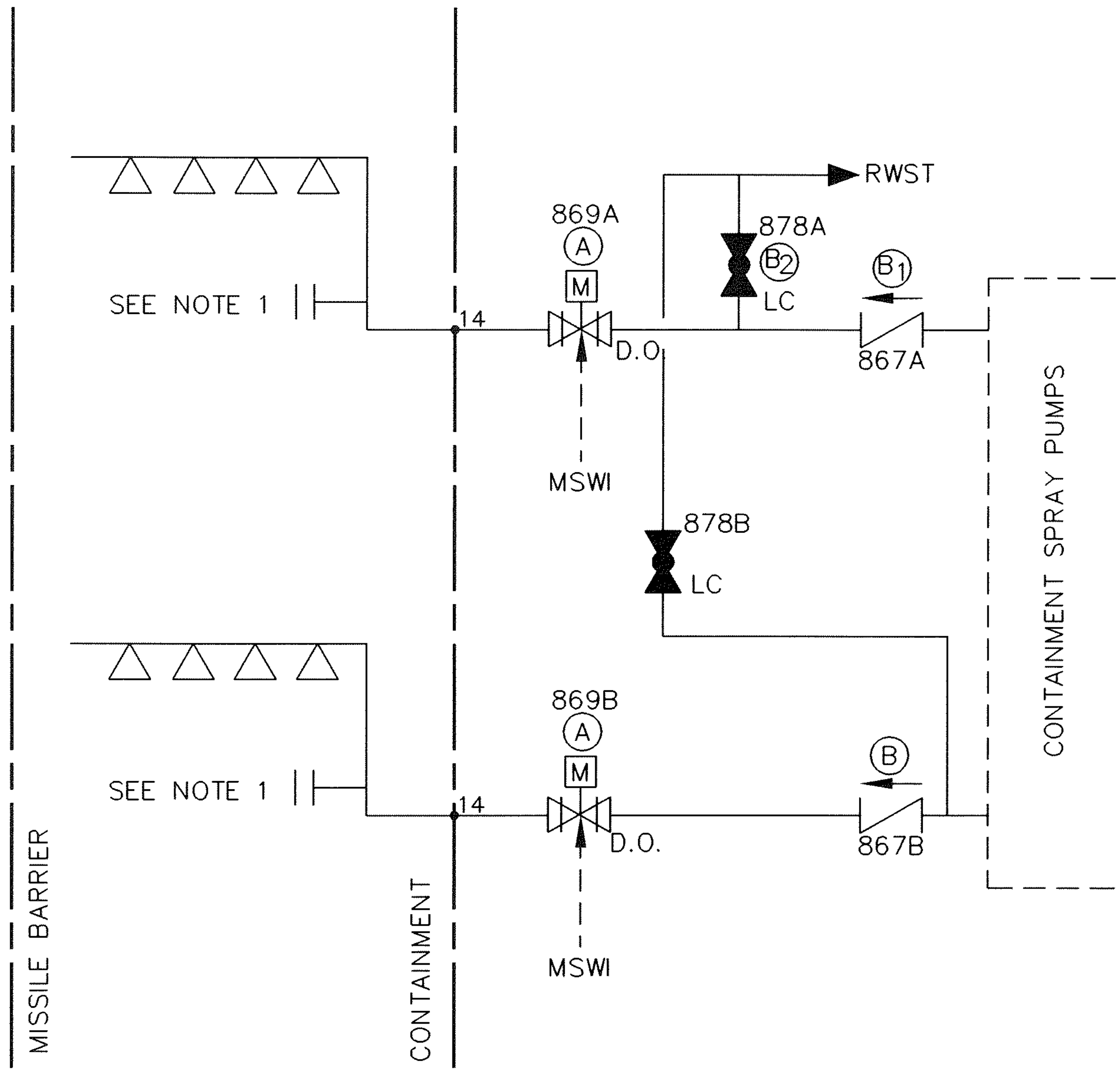
UFSAR FIGURE 5.2-5

CONTAINMENT ISOLATION SYSTEM
PENETRATION SCHEMATICS

MIC. No. 1999MC3386 | REV. No. 17A

VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE

ITEM 14 CONTAINMENT SPRAY HEADERS



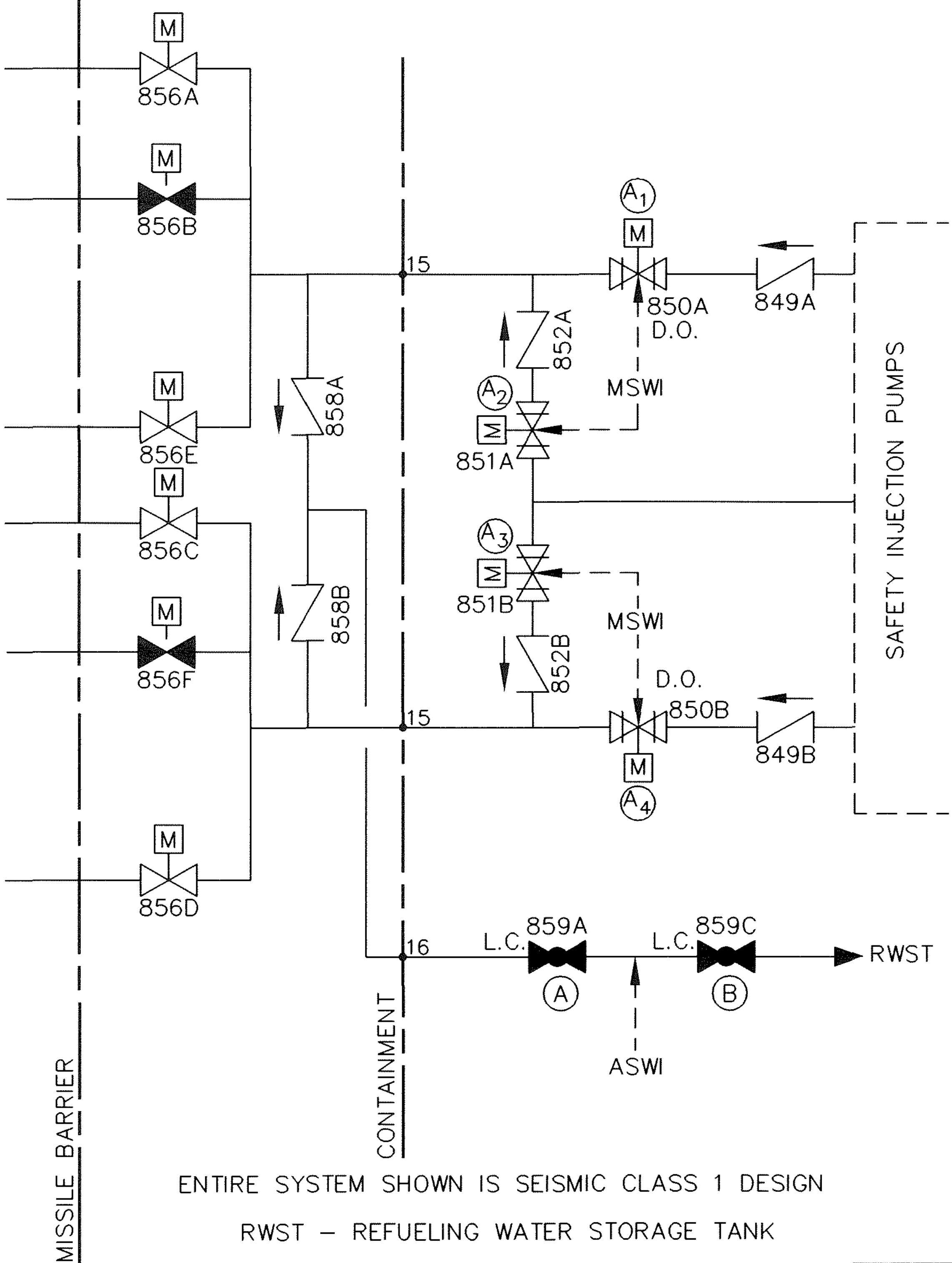
RWST – REFUELING WATER STORAGE TANK

ENTIRE SYSTEM SHOWN IS SEISMIC CLASS 1 DESIGN

NOTE 1:
FLANGED ELBOW USED DURING REFUELING OPERATIONS TO FILL REACTOR REFUELING CAVITY AND CANAL

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2–6 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3387	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

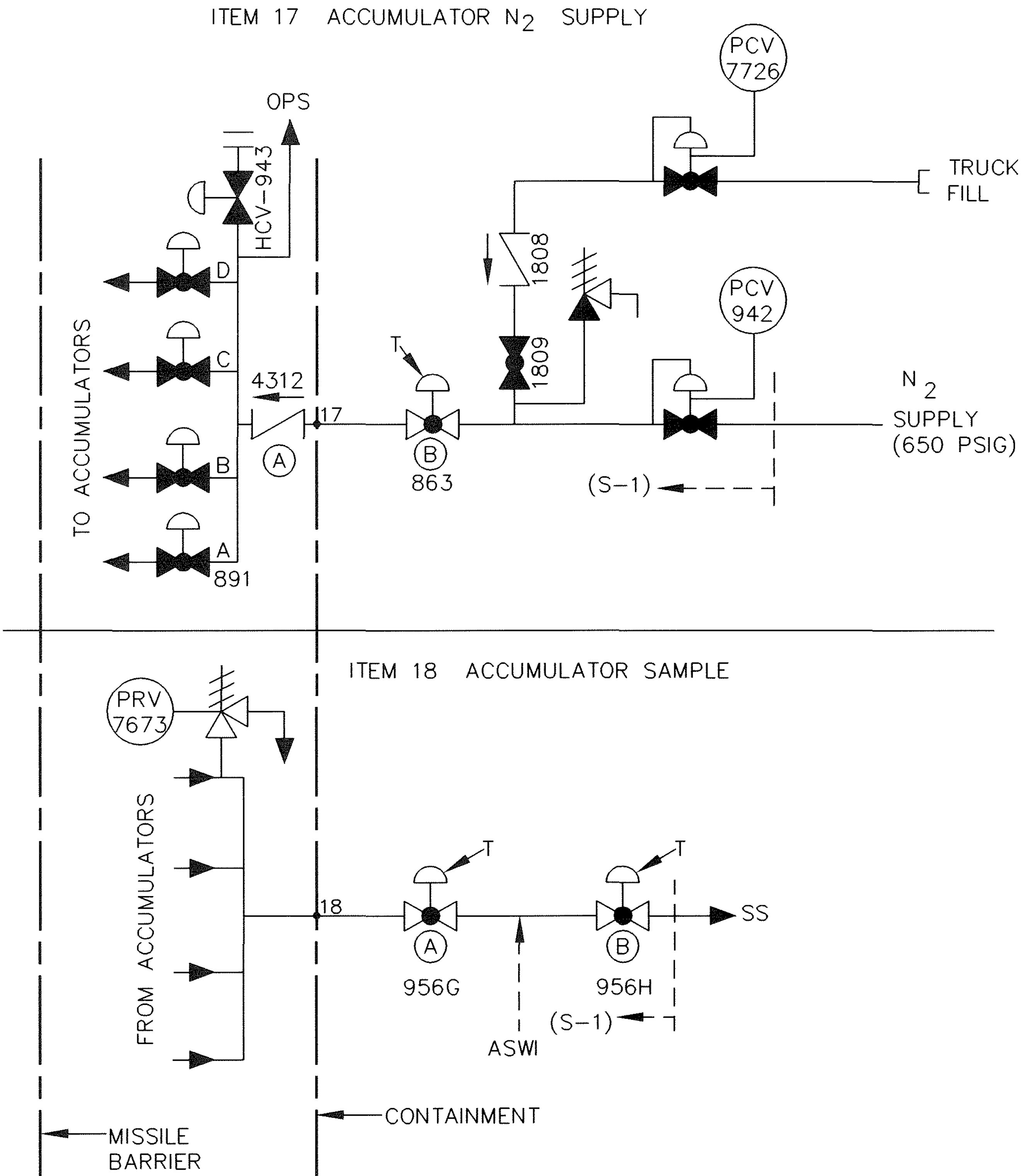
ITEM 15 SAFETY INJECTION HEADERS
ITEM 16 SAFETY INJECTION TEST LINE



ENTIRE SYSTEM SHOWN IS SEISMIC CLASS 1 DESIGN

RWST – REFUELING WATER STORAGE TANK

INDIAN POINT UNIT No. 2
UFSAR FIGURE 5.2–7 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS
MIC. No. 1999MC3388 — REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE



INDIAN POINT UNIT No. 2

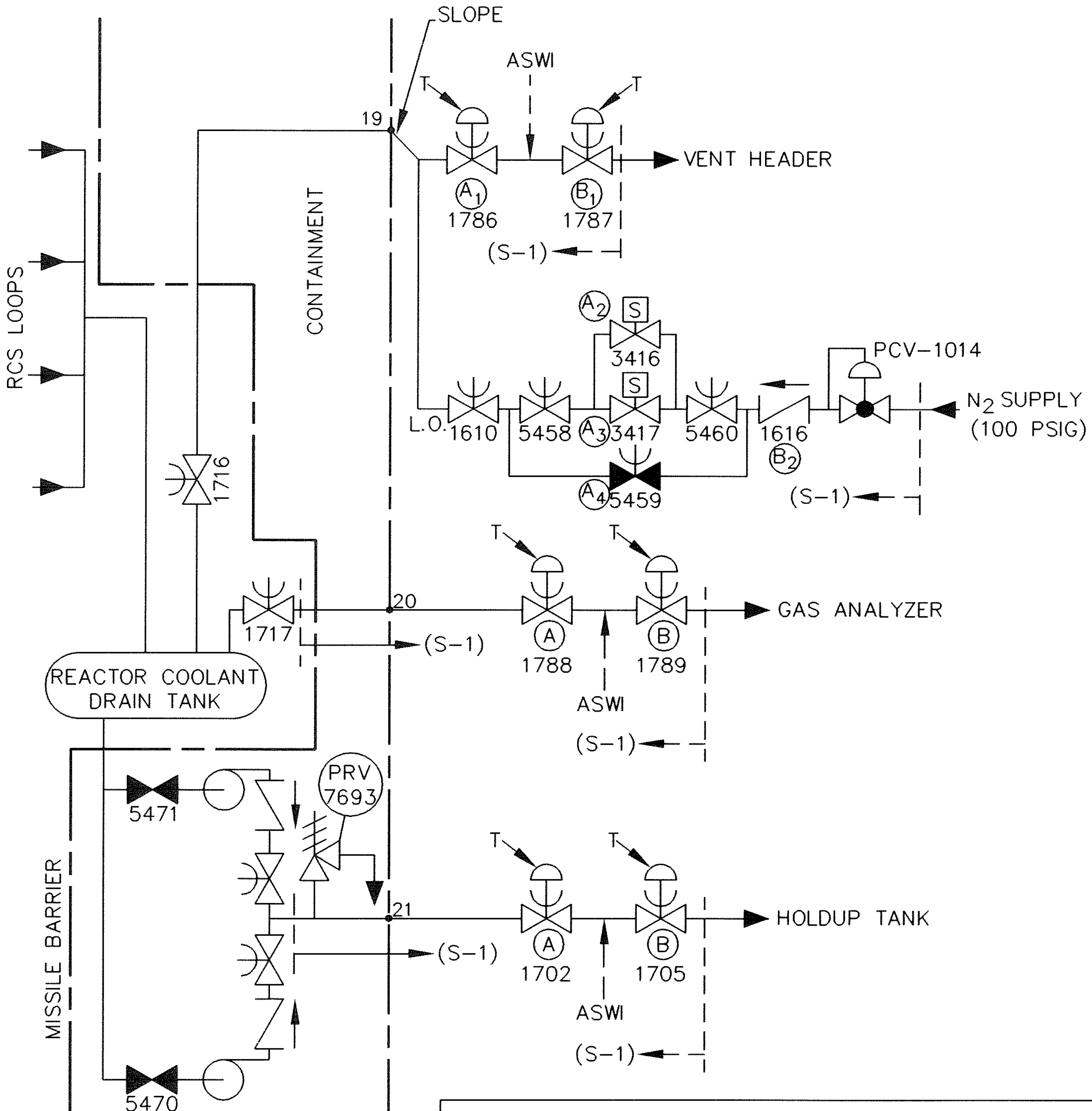
UFSAR FIGURE 5.2-8

CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS

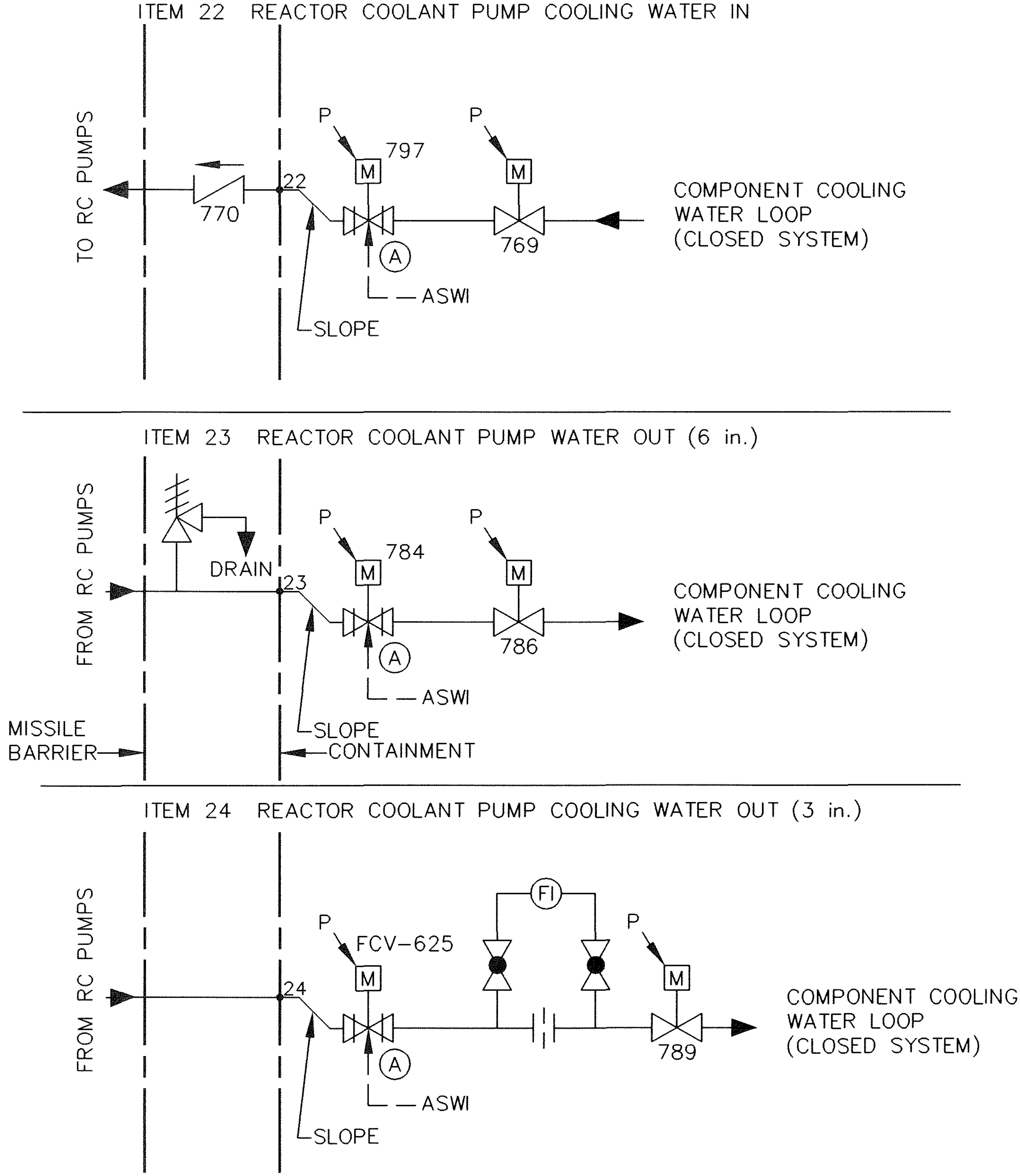
MIC. No. 1999MC3389	REV. No. 17A

VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE

ITEM 19 PRIMARY SYSTEM VENT HEADER AND N_2 SUPPLY LINE
ITEM 20 REACTOR COOLANT DRAIN TANK TO GAS ANALYZER
ITEM 21 RCDT PUMP DISCHARGE



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2-9	
CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3390	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

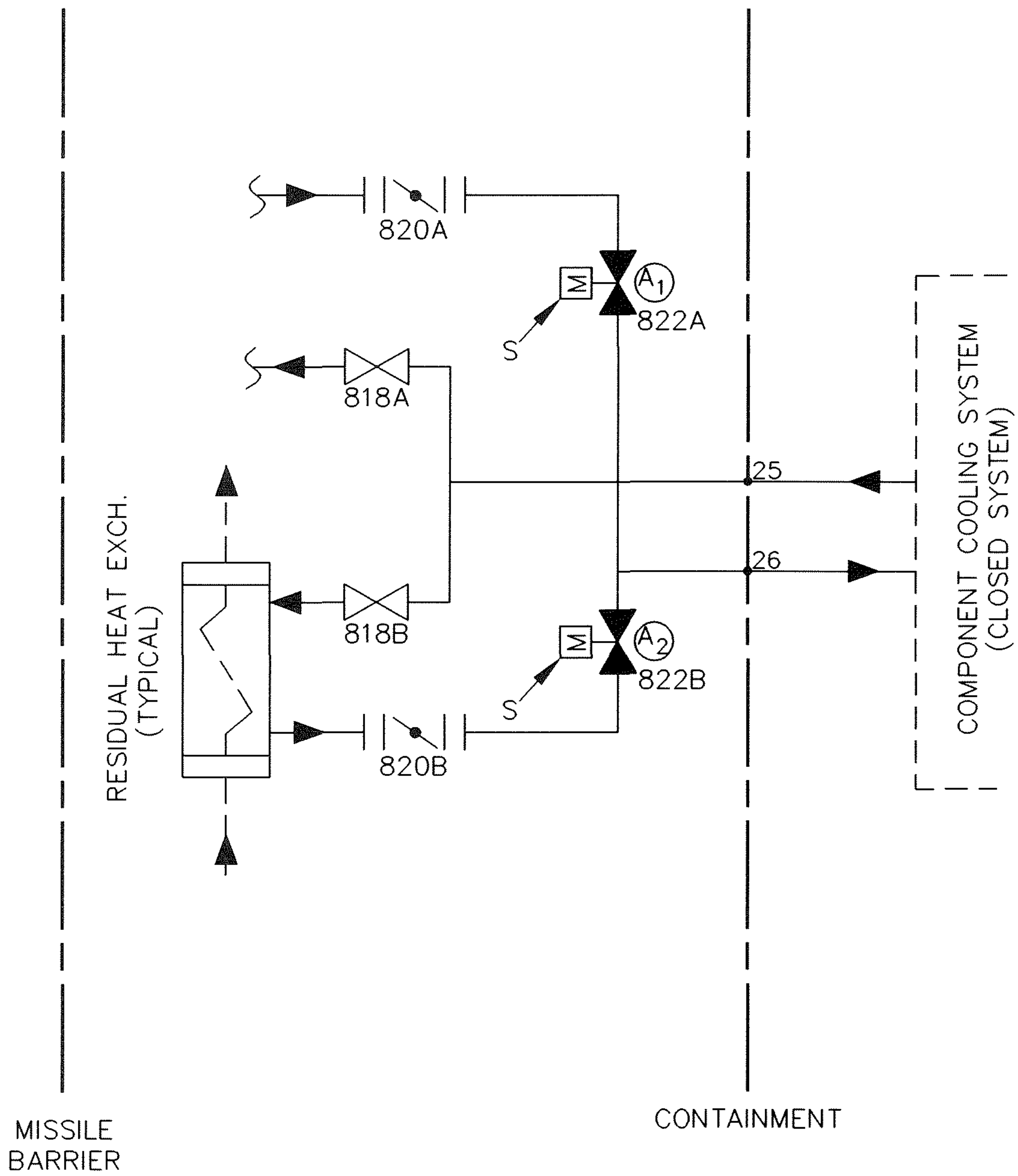


ENTIRE COMPONENT COOLING WATER SYSTEM IS SEISMIC CLASS 1 DESIGN

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2−10 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3391	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

ITEM 25 RESIDUAL HEAT EXCHANGER COOLING WATER IN

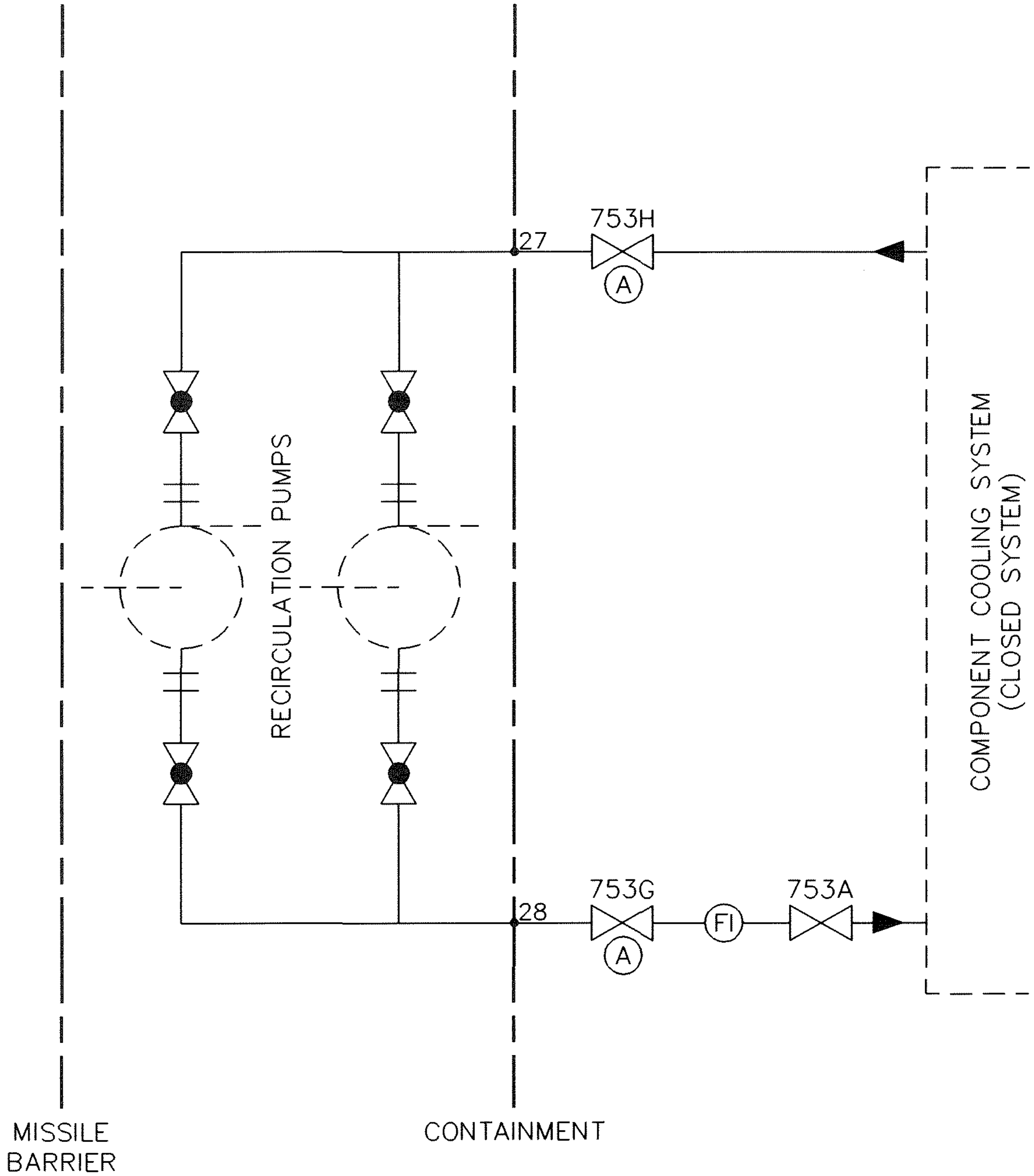
ITEM 26 RESIDUAL HEAT EXCHANGER COOLING WATER RETURN



ENTIRE SYSTEM SHOWN IS SEISMIC CLASS 1 DESIGN
S – OPEN S.I. SIGNAL, PHASE A

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2–11 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3392	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

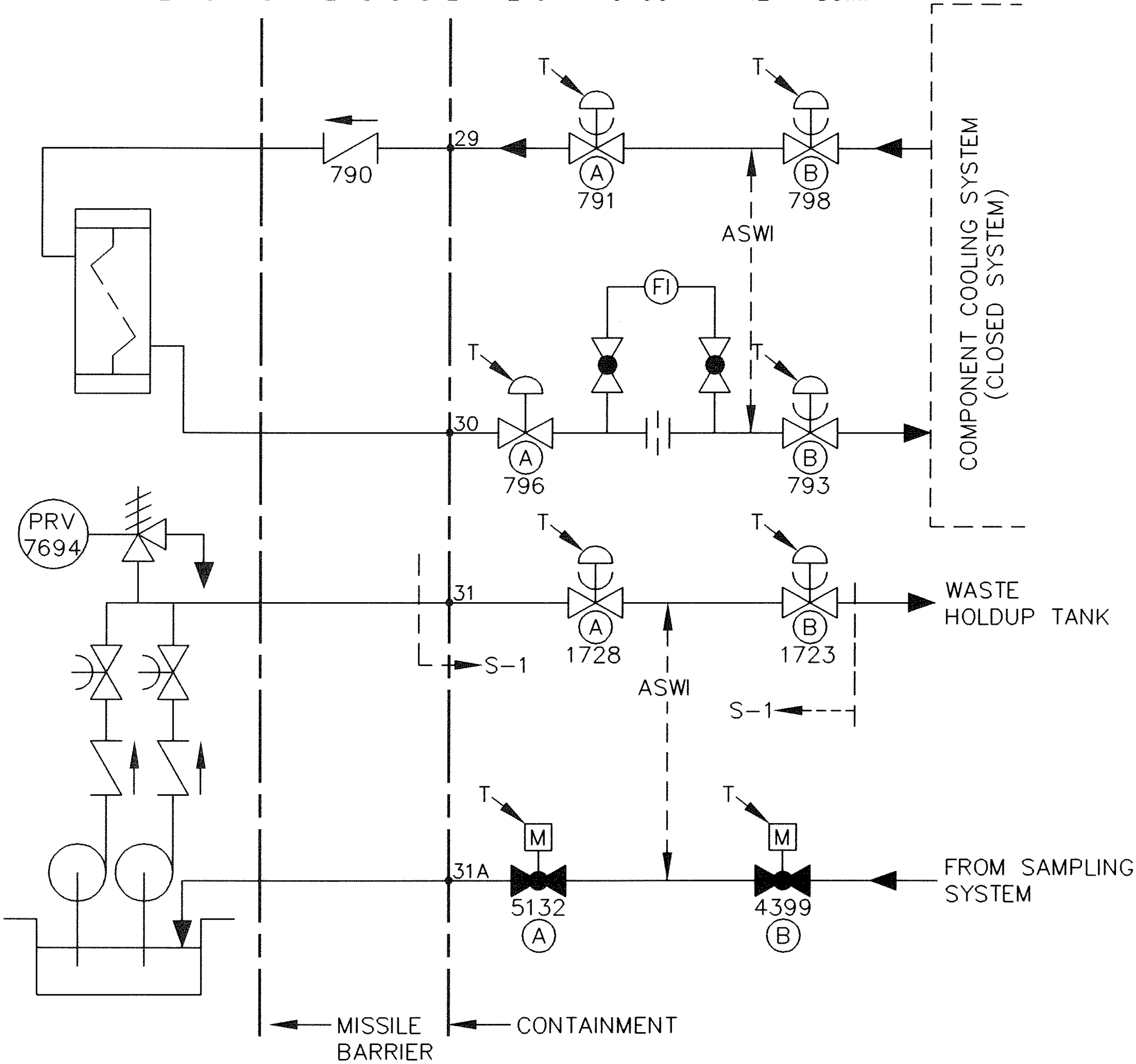
ITEM 27 RECIRCULATION PUMP COOLING WATER SUPPLY
ITEM 28 RECIRCULATION PUMP COOLING WATER RETURN



ENTIRE COMPONENT COOLING SYSTEM IS SEISMIC CLASS 1 DESIGN

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2-12 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3393	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

ITEM 29 EXCESS LETDOWN HEAT EXCHANGER COOLING WATER IN
ITEM 30 EXCESS LETDOWN HEAT EXCHANGER COOLING WATER OUT
ITEM 31 CONTAINMENT SUMP PUMP DISCHARGE
ITEM 31A SAMPLING SYSTEM RETURN TO CONTAINMENT SUMP



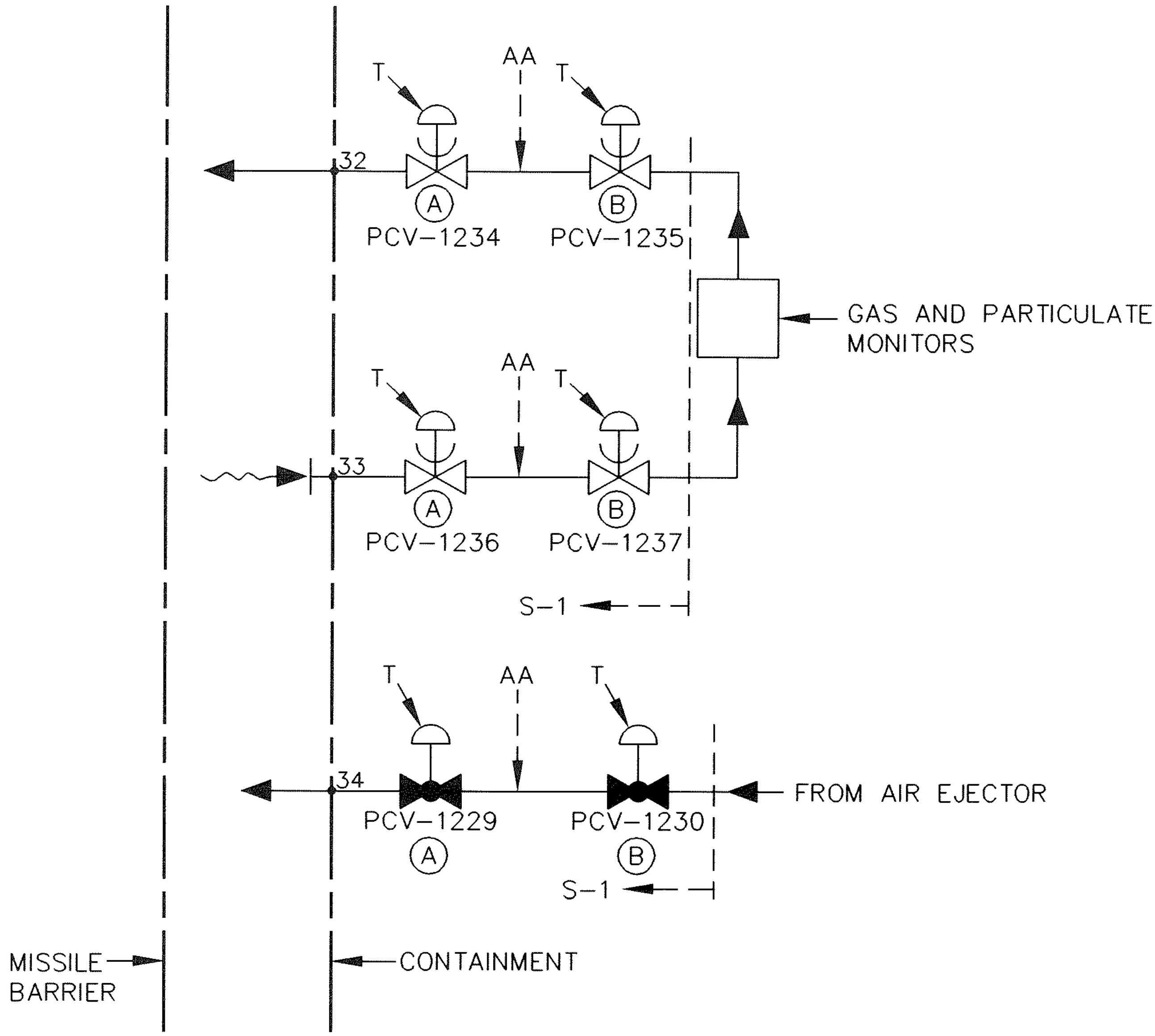
ENTIRE COMPONENT COOLING SYSTEM IS SEISMIC CLASS 1 DESIGN

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2-13 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3394	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

ITEM 32 CONTAINMENT AIR SAMPLE IN

ITEM 33 CONTAINMENT AIR SAMPLE OUT

ITEM 34 AIR EJECTOR DISCHARGE TO CONTAINMENT



AA – AUTOMATIC PRESSURIZATION WITH AIR FROM PENETRATION PRESSURIZATION SYSTEM

INDIAN POINT UNIT No. 2

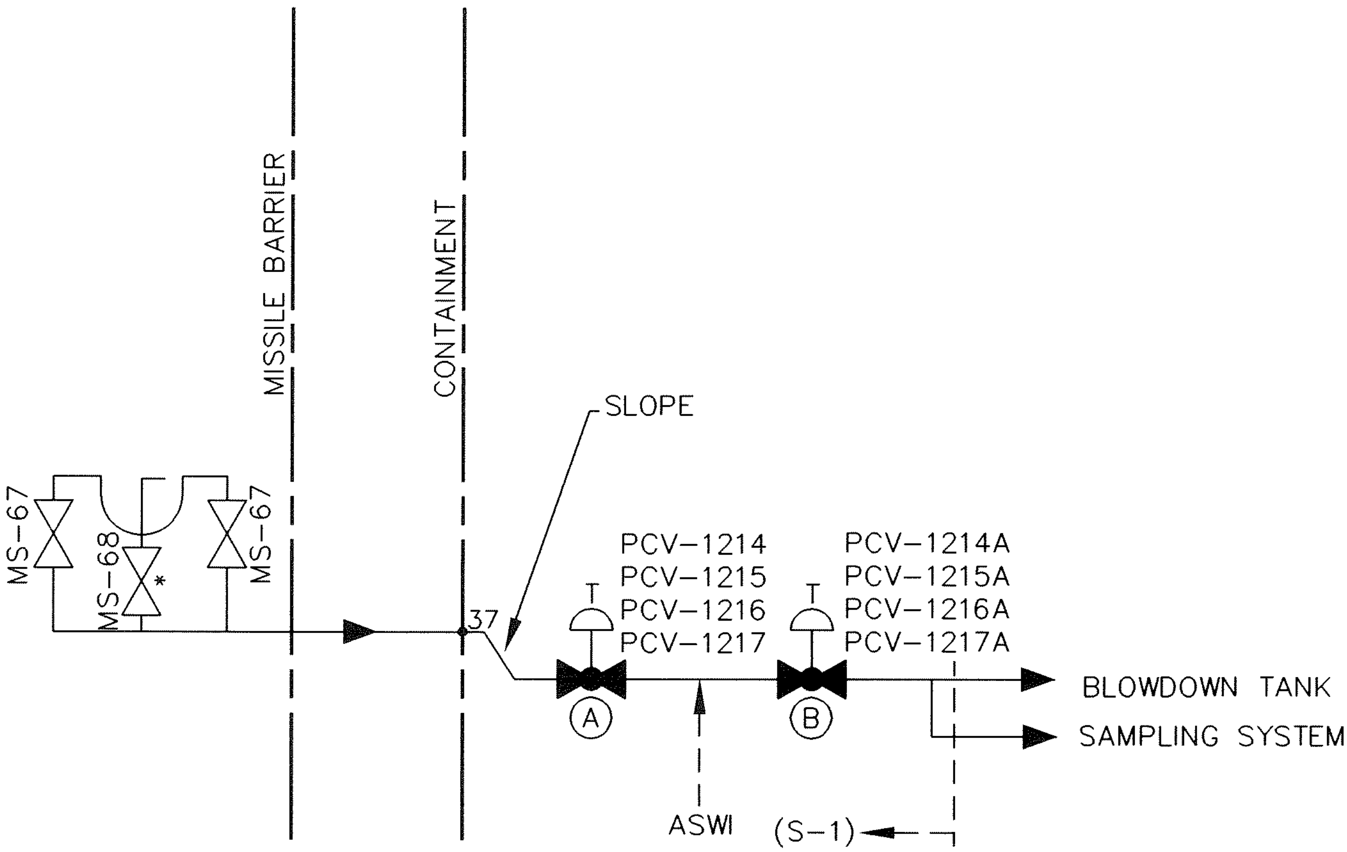
UFSAR FIGURE 5.2-14

CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS

MIC. No. 1999MC3395	REV. No. 17A

VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE

ITEM 37 STEAM GENERATOR BLOWDOWN/SAMPLE (4 PENETRATIONS)



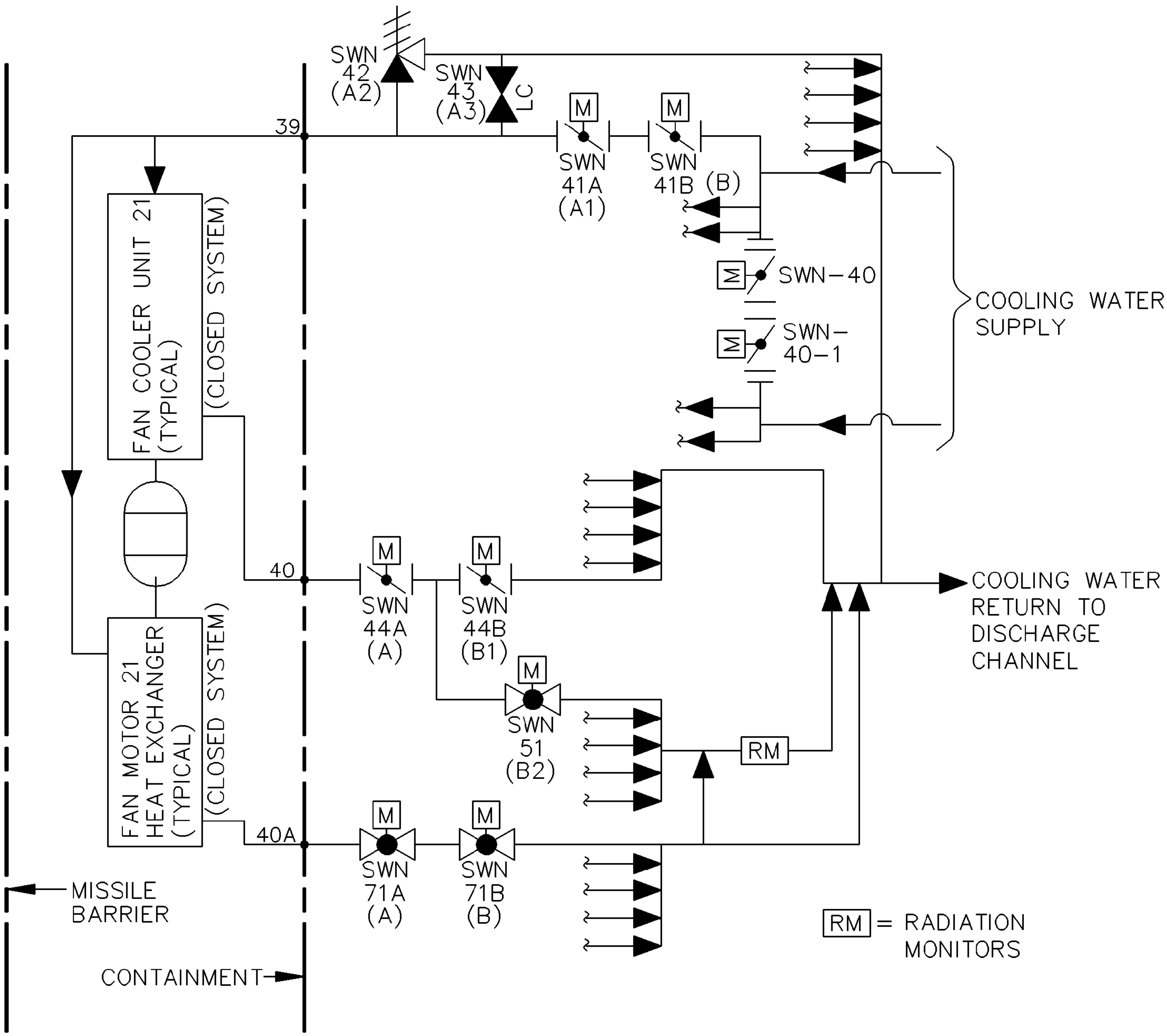
* THIS DRAIN LINE HAS BEEN REMOVED ON STEAM GENERATOR 21

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2-15 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3396	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

ITEM 39 10" VENTILATION SYSTEM COOLING WATER IN (5)

ITEM 40 10" VENTILATION SYSTEM COOLING WATER OUT (5)

ITEM 40A 2" VENTILATION SYSTEM MOTOR COOLING WATER OUT (5)



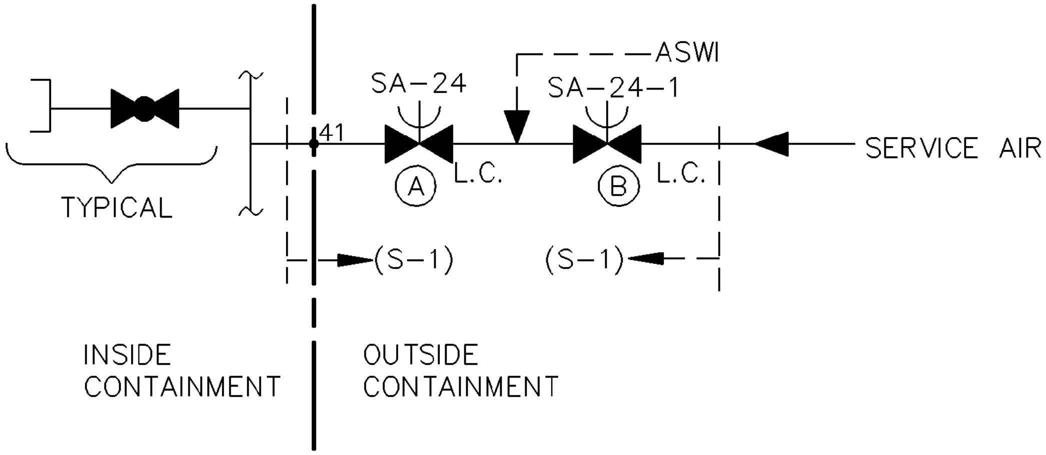
ENTIRE SYSTEM SHOWN IS SEISMIC CLASS 1 DESIGN

ALL MOV'S ARE DEENERGIZED OPEN

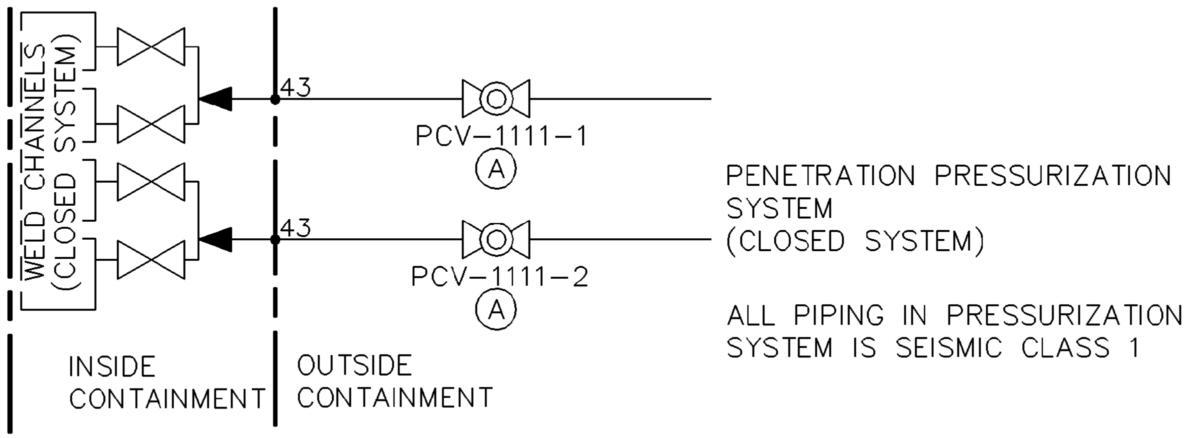
PLANT MANAGEMENT MAY DESIGNATE THE "A" OR THE "B" VALVE(S) IN THE 41, 44 AND 71 SERIES AS THE REQUIRED CONTAINMENT ISOLATION VALVE(S). DESIGNATION OF THE "B" VALVE(S) IN THE SWN-44 SERIES REQUIRES THE CODESIGNATION OF THE RESPECTIVE 51 SERIES VALVE(S) AS ADDITIONAL REQUIRED CONTAINMENT ISOLATION VALVE(S). CONTAINMENT ISOLATION VALVE SURVEILLANCE REQUIREMENTS MUST HAVE BEEN SATISFIED FOR THE DESIGNATED VALVE(S).

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2-16 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3397	REV. No. 17B
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

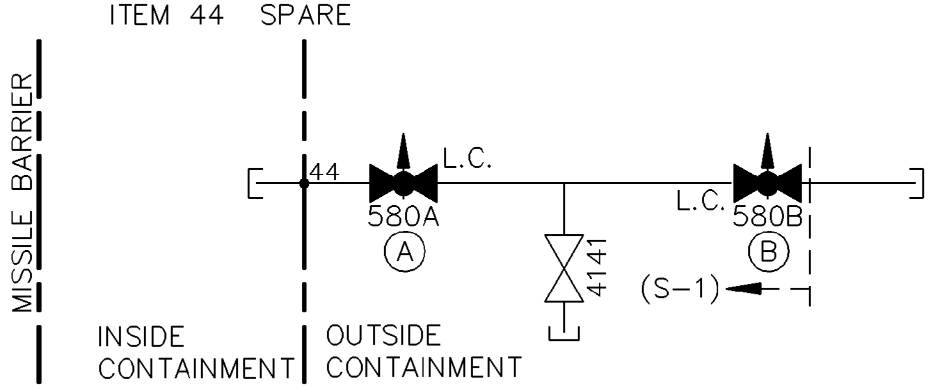
ITEM 41 SERVICE AIR



ITEM 43 WELD CHANNEL PRESSURIZATION AIR SUPPLY



ITEM 44 SPARE



INDIAN POINT UNIT No. 2

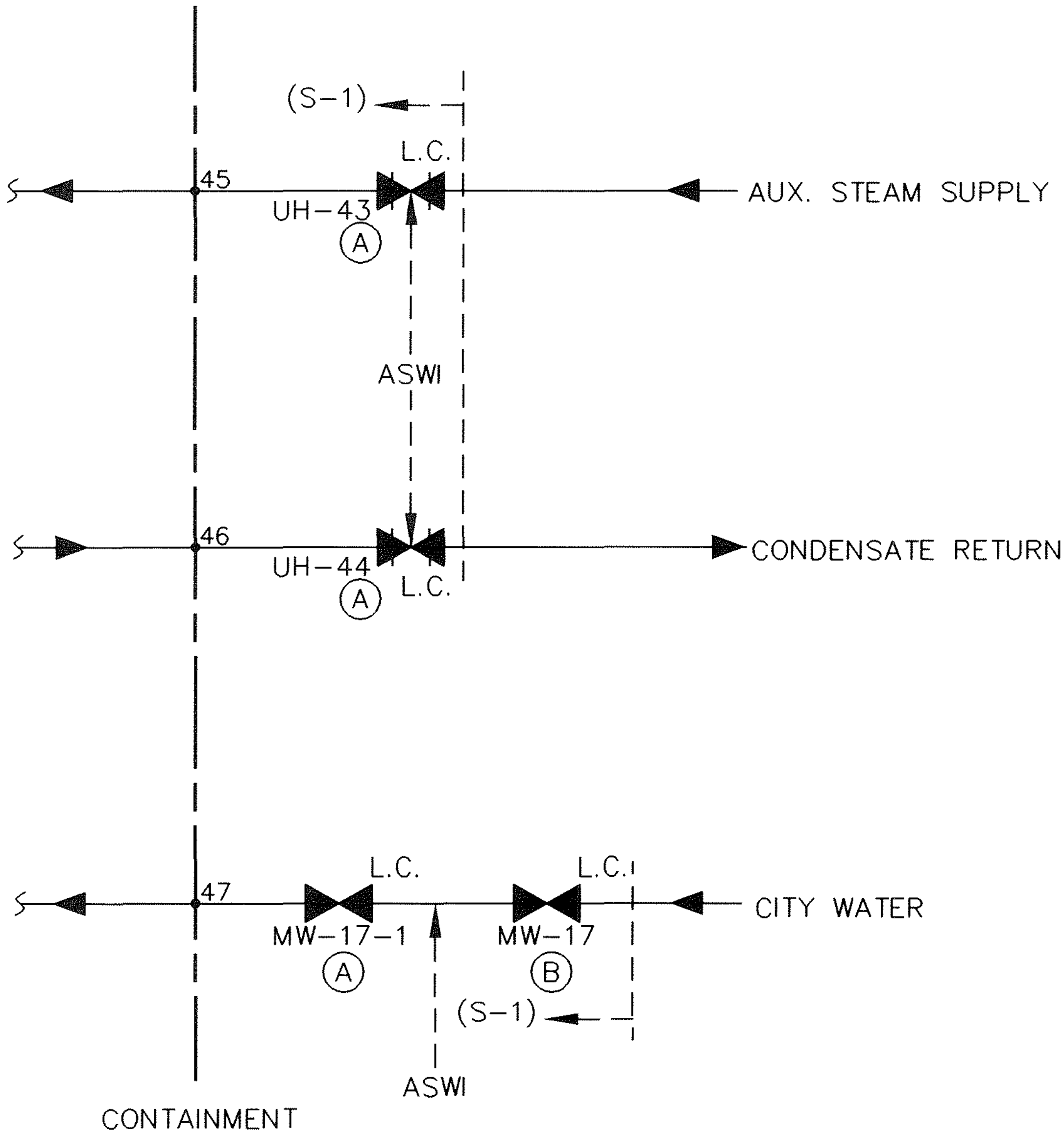
UFSAR FIGURE 5.2-17

CONTAINMENT ISOLATION SYSTEM
PENETRATION SCHEMATICS

MIC. No. 1999MC3398	REV. No. 17B

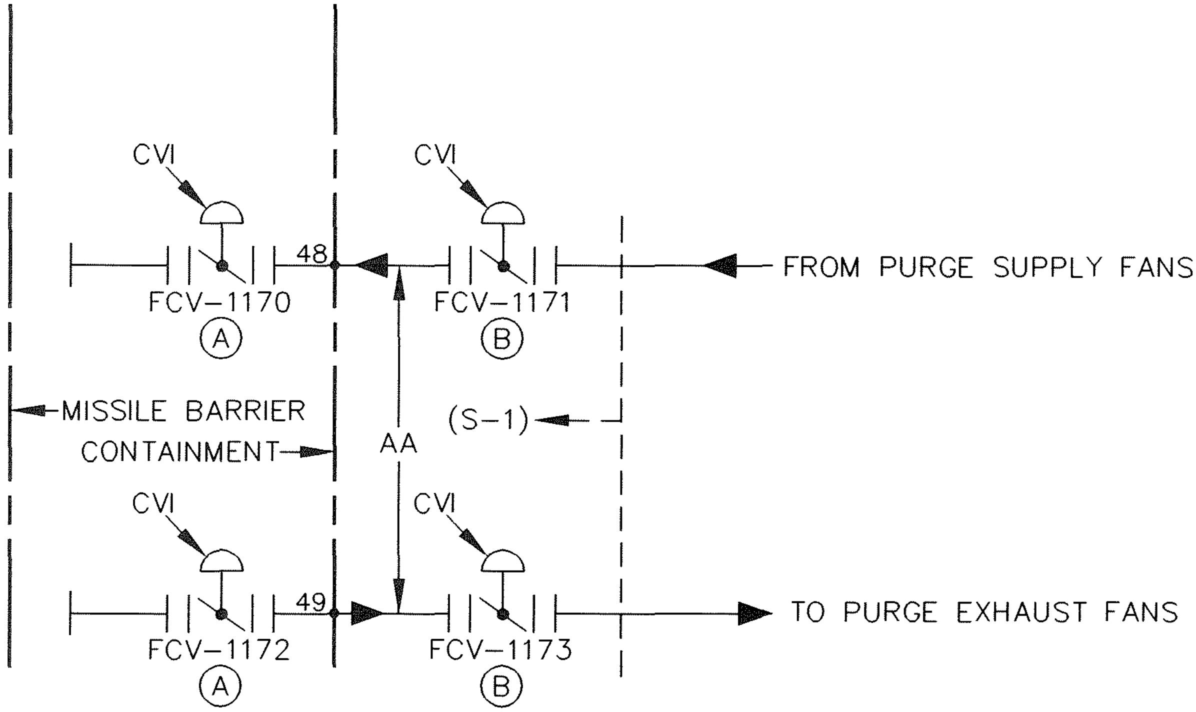
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE

ITEM 45 AUXILIARY STEAM SUPPLY
ITEM 46 AUXILIARY STEAM CONDENSATE RETURN
ITEM 47 CITY WATER

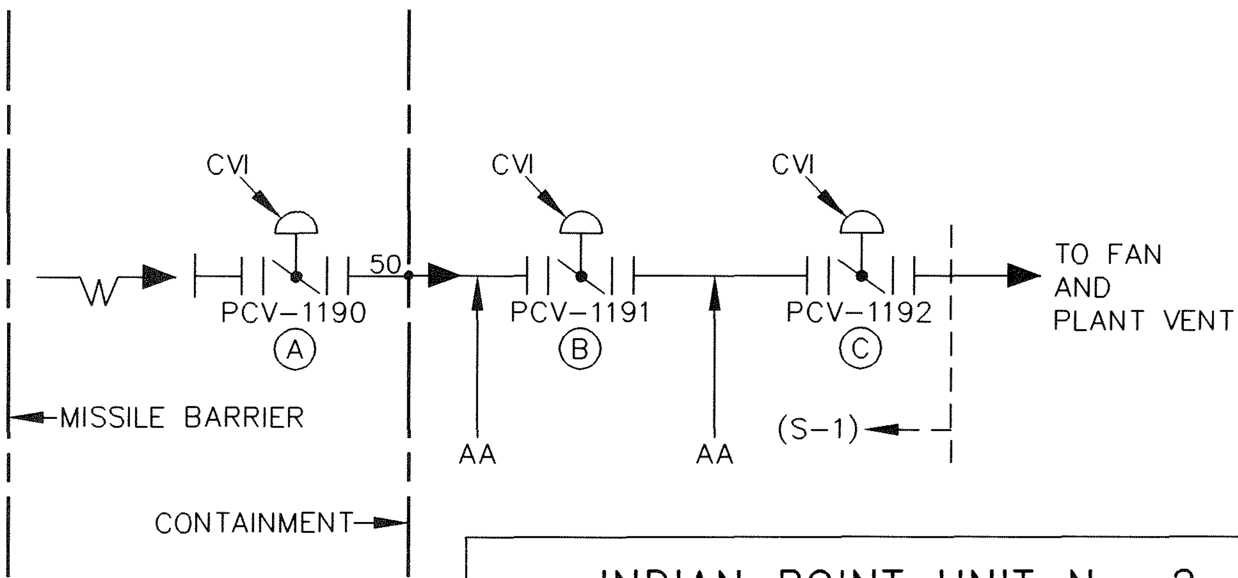


INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2-18 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3399	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

ITEM 48 PURGE SUPPLY DUCT
ITEM 49 PURGE EXHAUST DUCT



ITEM 50 CONTAINMENT PRESSURE RELIEF



INDIAN POINT UNIT No. 2

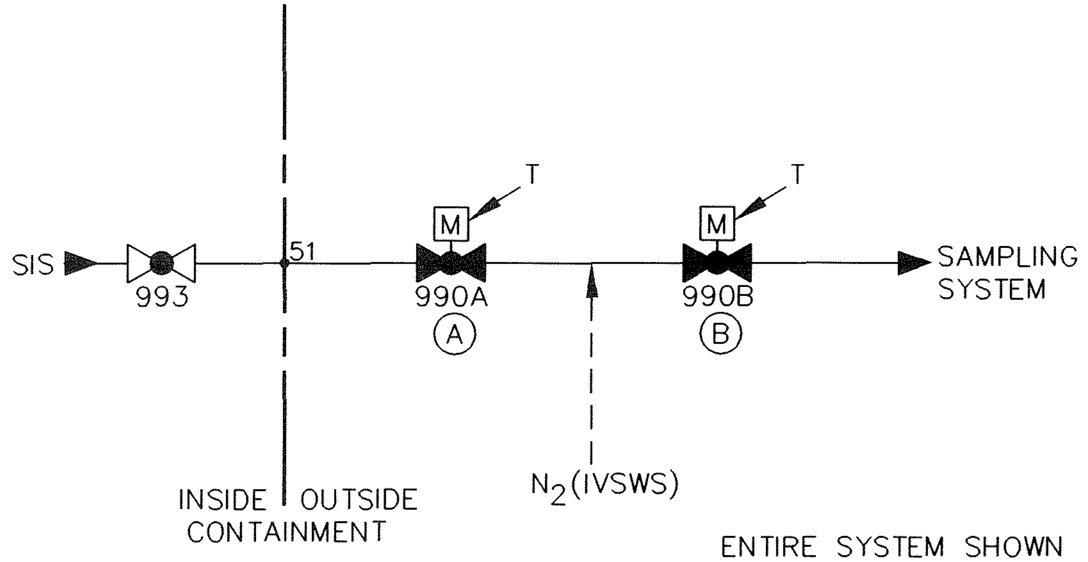
UFSAR FIGURE 5.2−19

CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS

MIC. No. 1999MC3400	REV. No. 17A

VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE

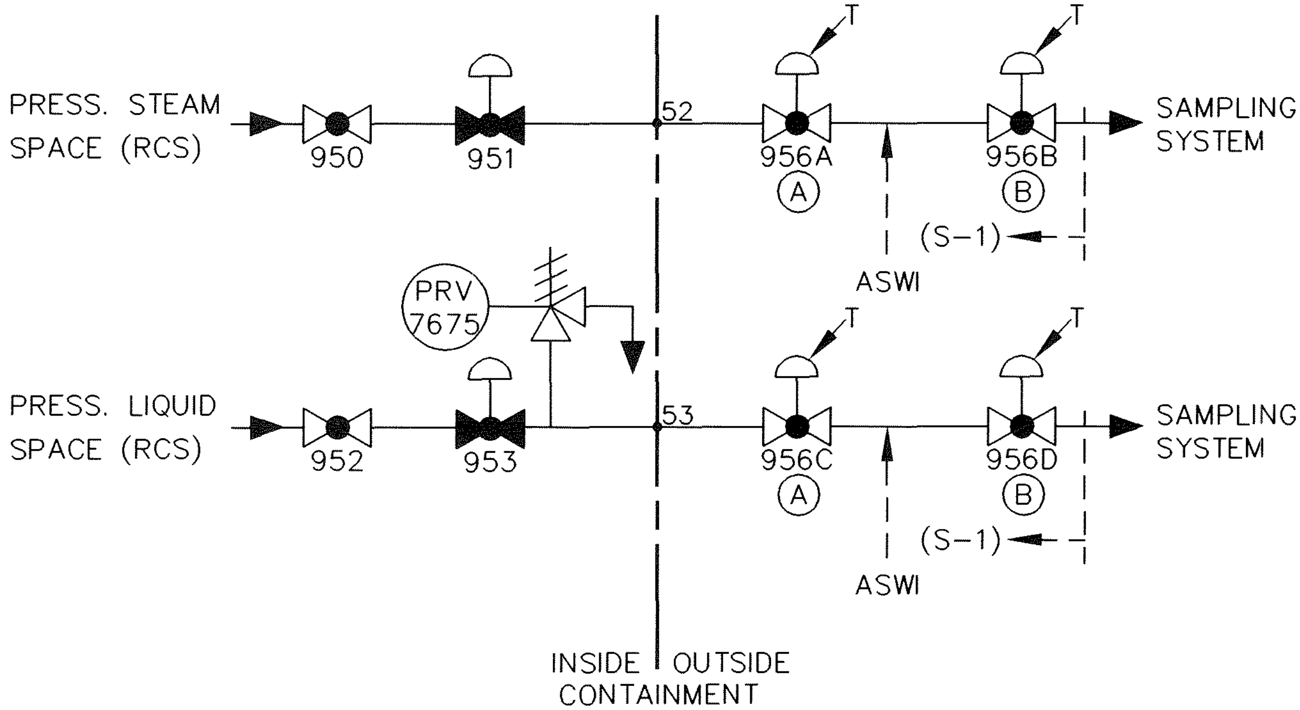
ITEM 51 RECIRCULATION PUMP DISCHARGE SAMPLE LINE



ENTIRE SYSTEM SHOWN
IS SEISMIC CLASS 1

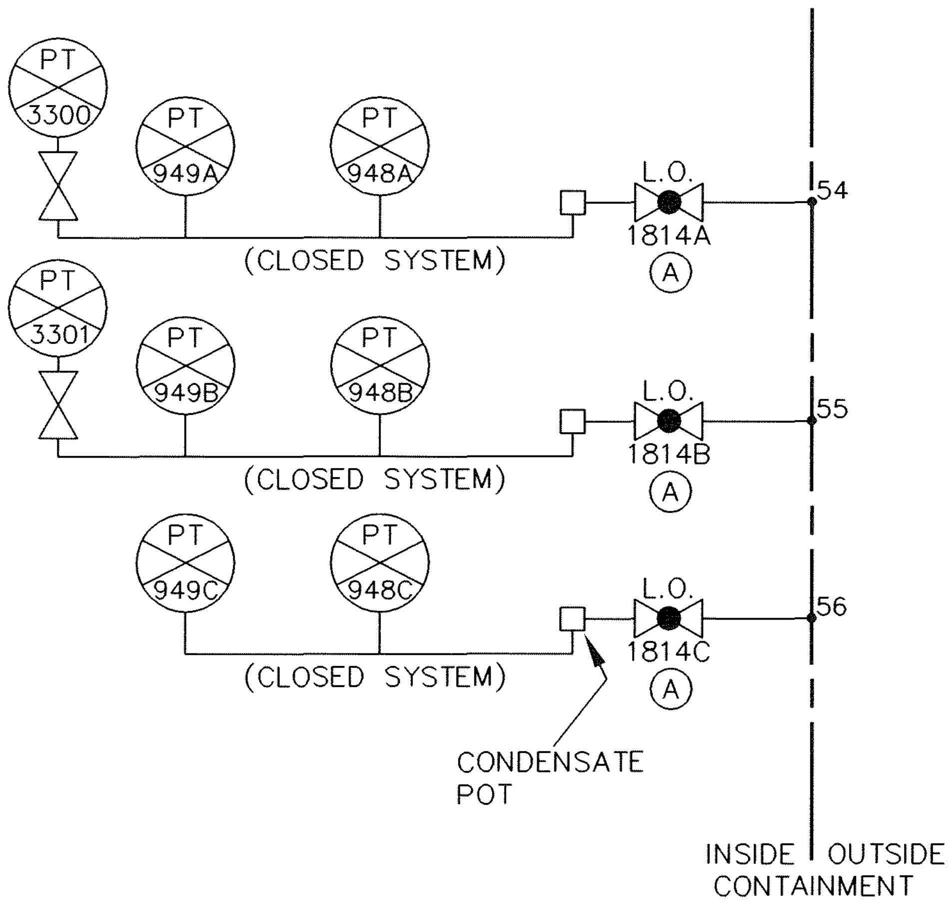
ITEM 52 PRESSURIZER STEAM SPACE SAMPLE

ITEM 53 PRESSURIZER LIQUID SPACE SAMPLE



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2-20 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3401	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

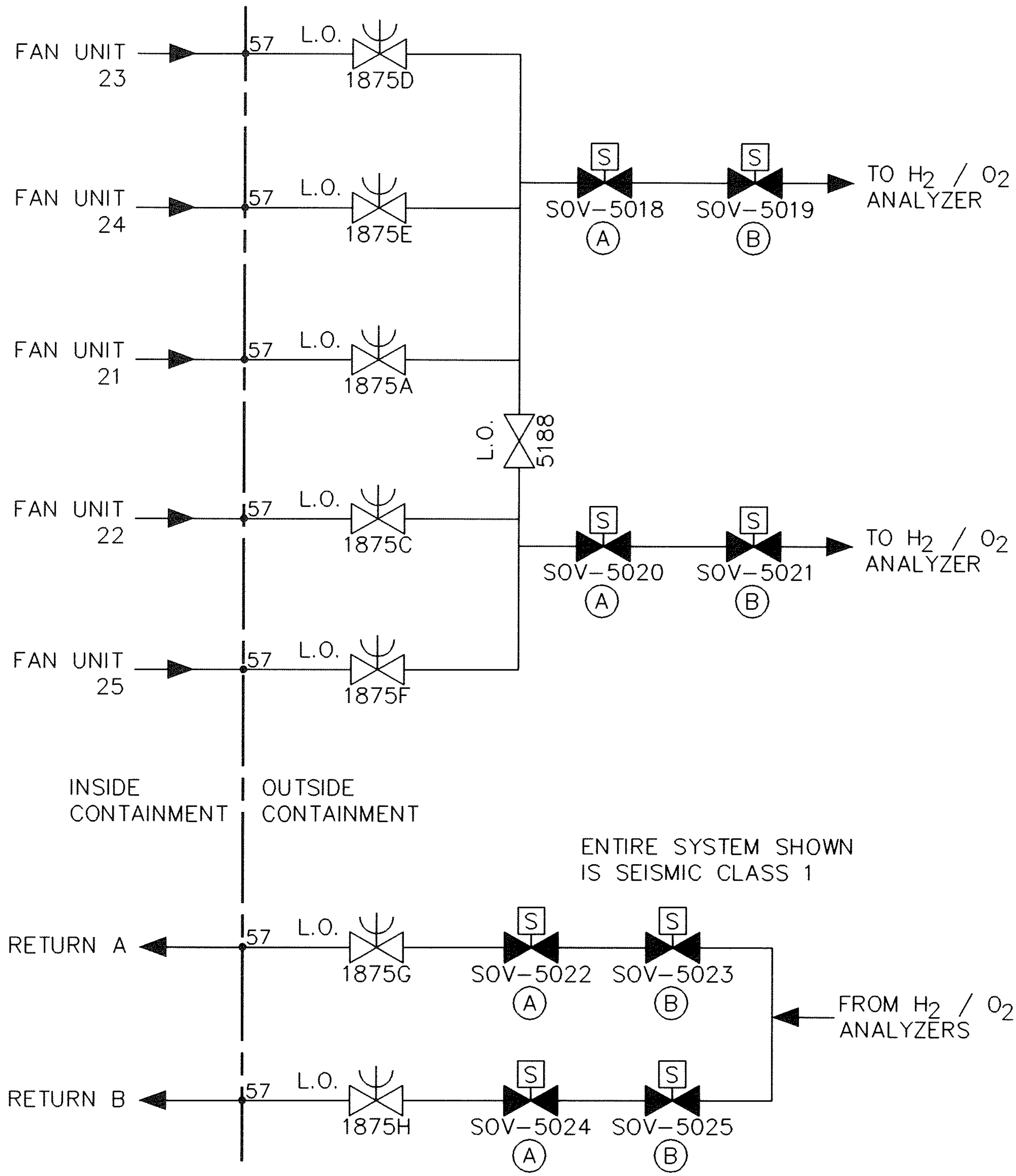
ITEMS 54, 55, AND 56 CONTAINMENT PRESSURE INSTRUMENTATION LINES



ENTIRE SYSTEM SHOWN IS SEISMIC CLASS 1

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2-21 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3402	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

ITEM 57 POSTACCIDENT CONTAINMENT SAMPLING LINES (SUPPLY AND RETURN)



INDIAN POINT UNIT No. 2

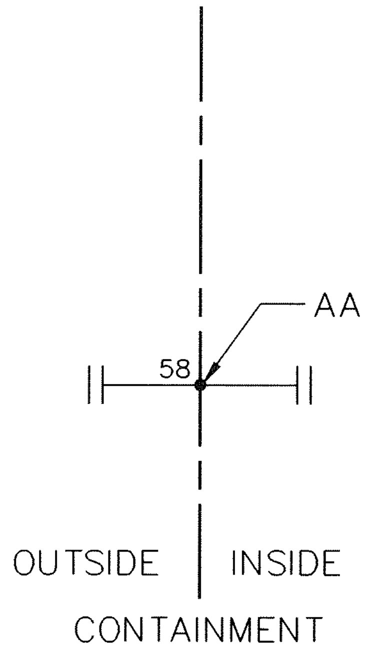
UFSAR FIGURE 5.2-22

CONTAINMENT ISOLATION SYSTEM
PENETRATION SCHEMATICS

MIC. No. 1999MC3403	REV. No. 17A

VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE

ITEM 58 SPARE



INDIAN POINT UNIT No. 2

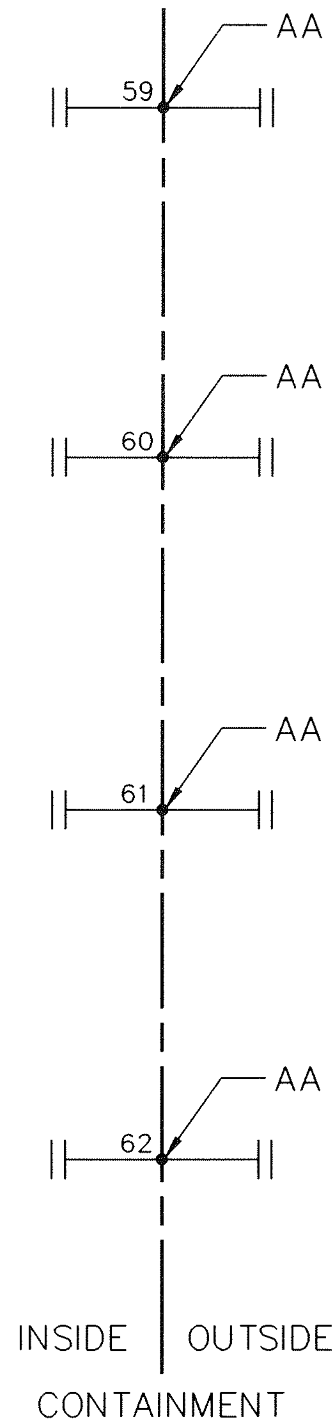
UFSAR FIGURE 5.2–23

CONTAINMENT ISOLATION SYSTEM
PENETRATION SCHEMATICS

MIC. No. 1999MC3404	REV. No. 17A

VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE

ITEMS 59, 60, 61, and 62 SPARE

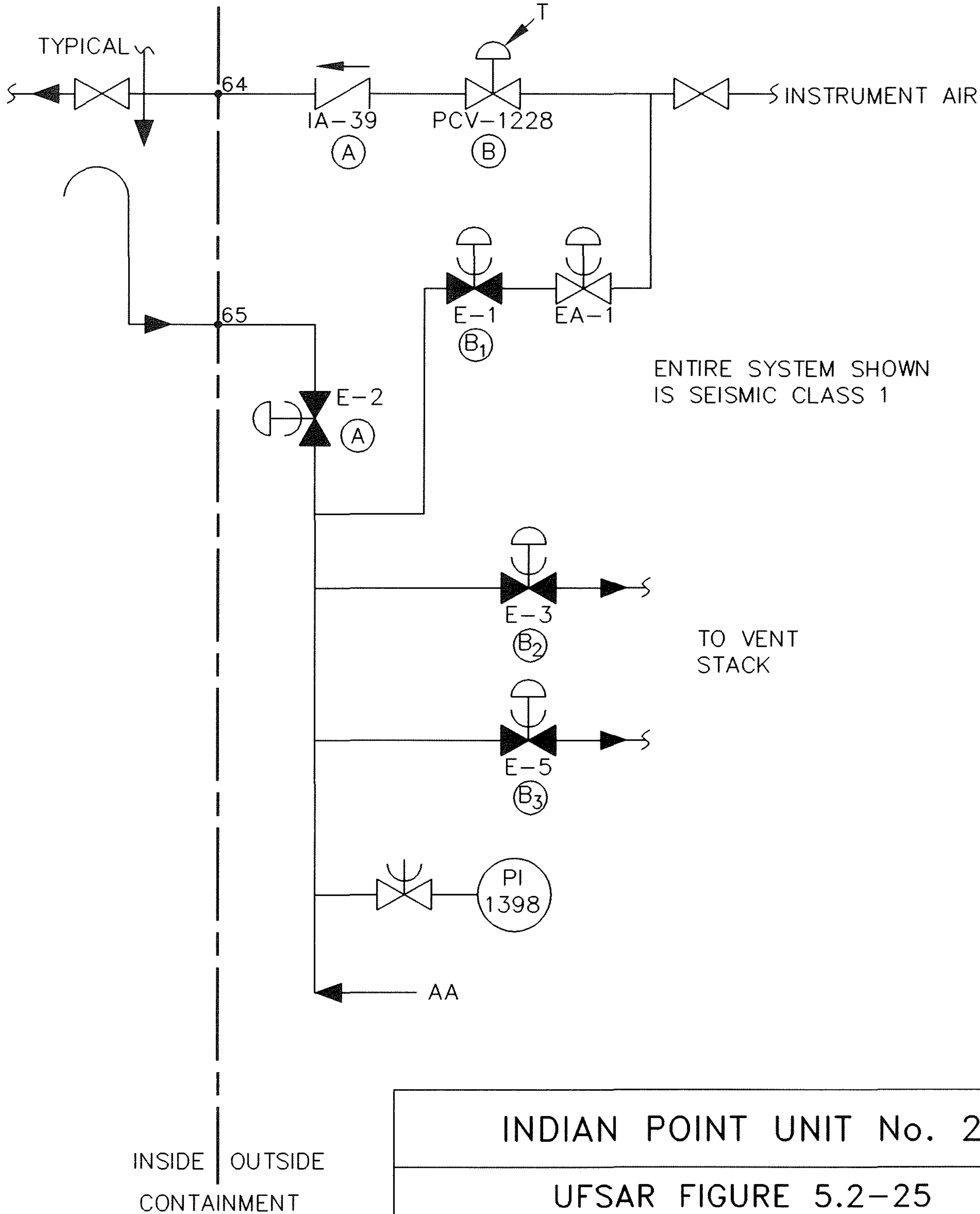


INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2-24 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3405	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

POSTACCIDENT (P.A.) CONTAINMENT VENTING SYSTEM

ITEM 64 INSTRUMENT AIR/P.A. VENTING SUPPLY LINE

ITEM 65 P.A. VENTING EXHAUST LINE



INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.2-25

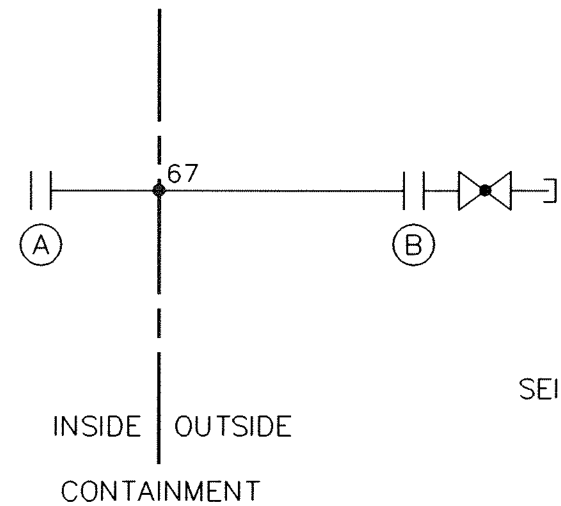
CONTAINMENT ISOLATION SYSTEM
PENETRATION SCHEMATICS

MIC. No. 1999MC3406	REV. No. 17A

VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE

ITEM 66 DELETED

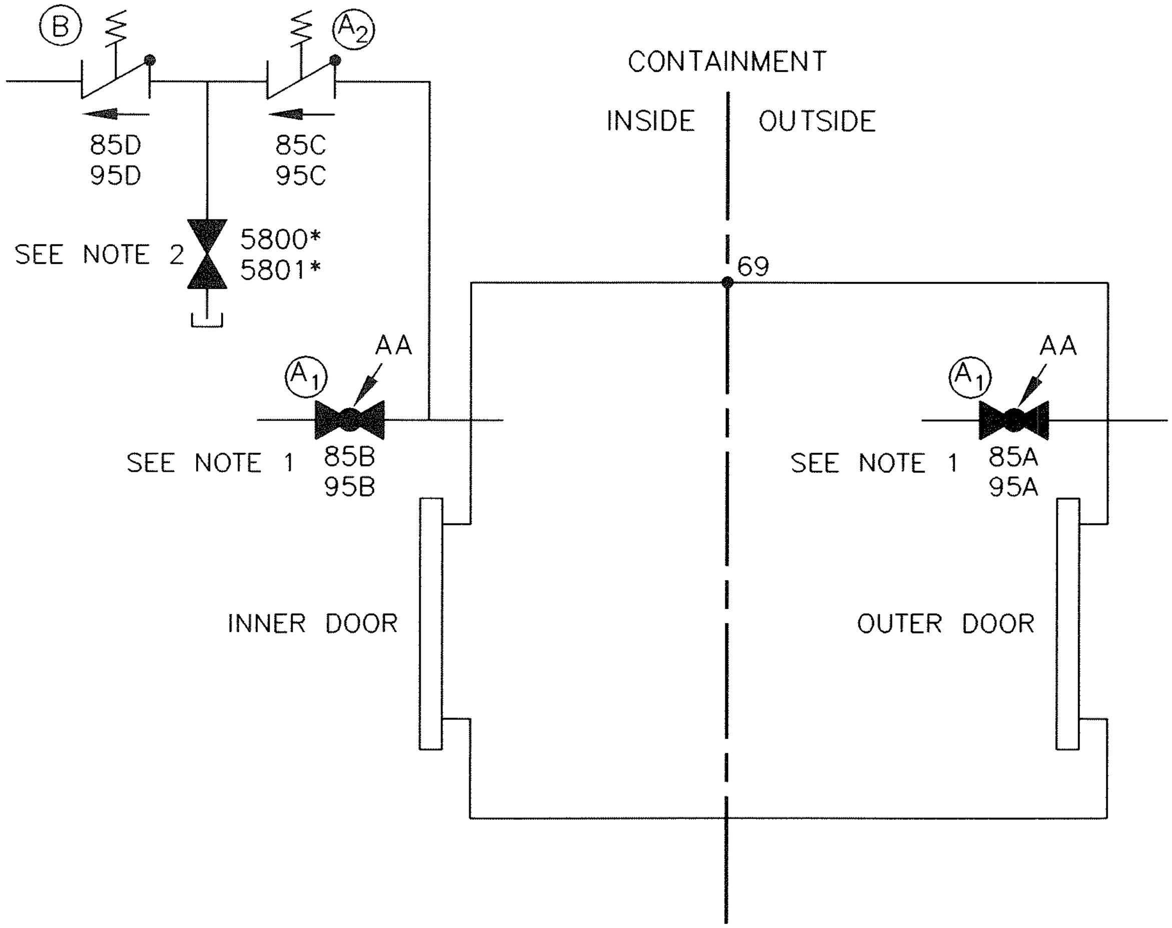
ITEM 67 CONTAINMENT LEAK TEST AIR LINE (2)



SEISMIC CLASS 1

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2–26 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3407	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

ITEM 69 PERSONNEL AIR LOCK (2)



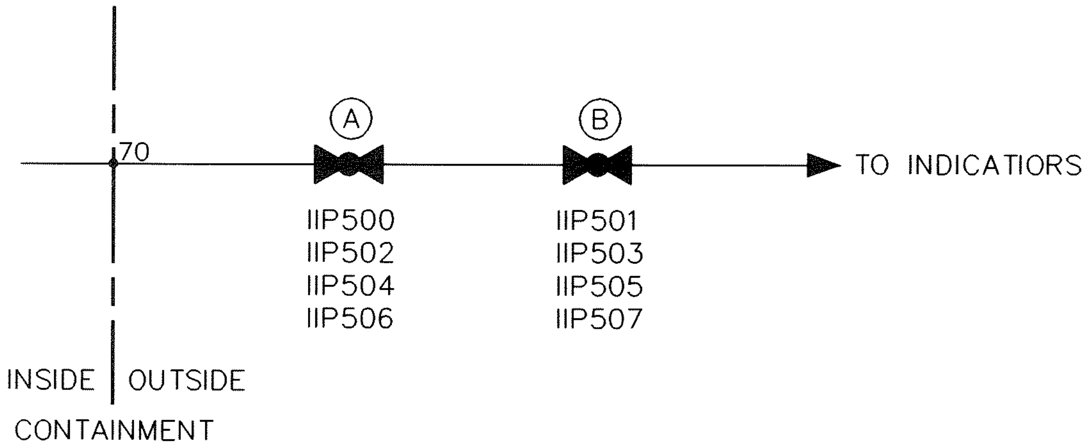
ENTIRE SYSTEM SHOWN IS SEISMIC CLASS 1 DEISGN

NOTE 1 : 85A & 95A MAY BE OPEN WHEN 85B & 95B ARE CLOSED.
85B & 95B MAY BE OPEN WHEN 85A & 95A ARE CLOSED.

2 : TEST VALVES 5800 & 5801 ARE LOCATED AT 83 FT. & 95 FT. AIRLOCKS RESPECTIVELY

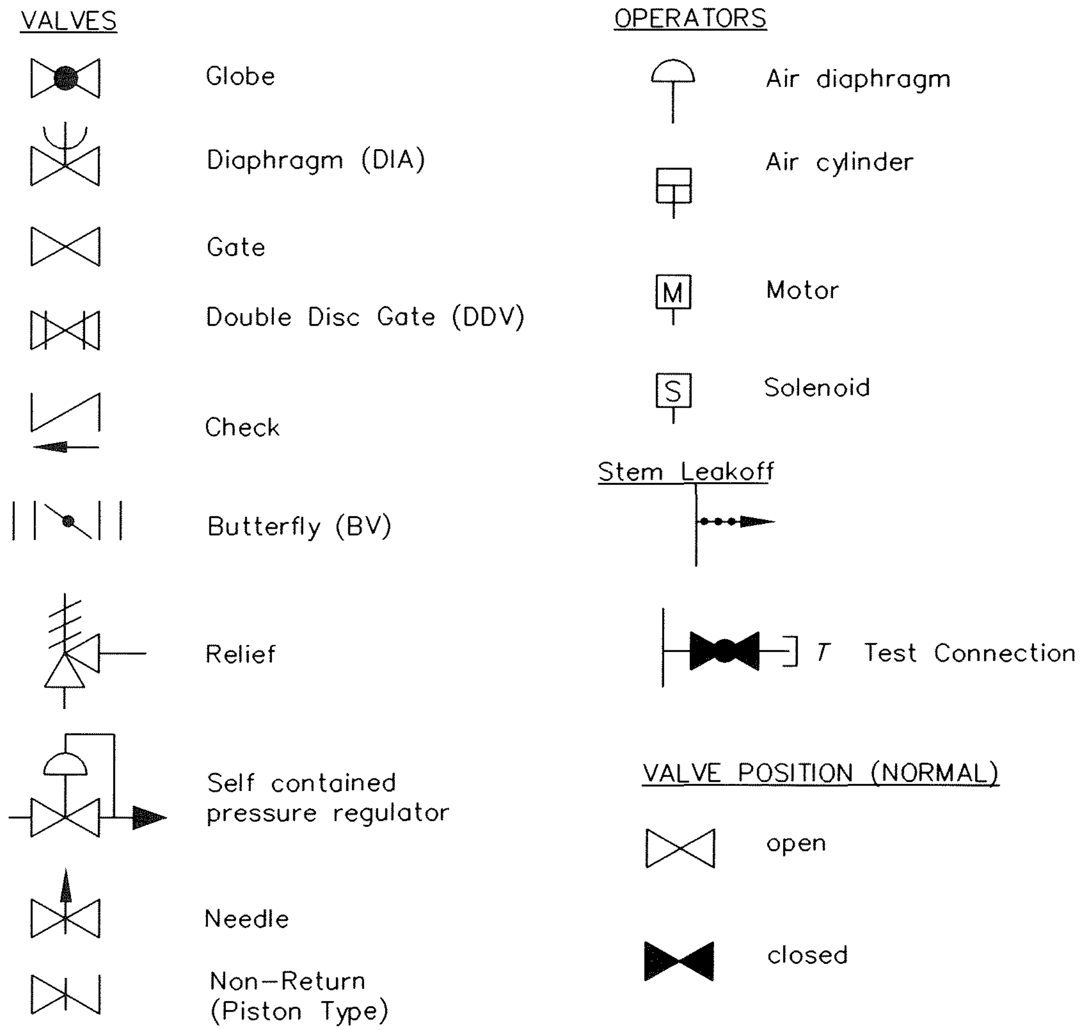
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 5.2-27 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS	
MIC. No. 1999MC3408	REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE	

ITEM 70 STEAM GENERATOR LEVEL INDICATION LINES (2)
PRESURIZER LEVEL INDICATION LINES (1)
PRESSURIZER PRESSURE INDICATION LINES (1)



INDIAN POINT UNIT No. 2
UFSAR FIGURE 5.2-28 CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS
MIC. No. 1999MC3409 — REV. No. 17A
VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE

LEGEND FOR SYMBOLS, CONTAINMENT ISOLATION SYSTEM



NOTATION

ASWI	–	AUTOMATIC SEAL WATER INJECTION
MSWI	–	MANUAL SEAL WATER INJECTION
AA	–	AUTOMATIC PRESSURIZATION WITH AIR
N_2	–	MANUAL PRESSURIZATION WITH NITROGEN
LO	–	LOCKED OPEN
LC	–	LOCKED CLOSED
P	–	TRIPPED CLOSED BY CONTAINMENT ISOLATION SIGNAL, PHASE A
T	–	TRIPPED CLOSED BY CONTAINMENT ISOLATION SIGNAL, PHASE B
S–1	–	SEISMIC CLASS I
S	–	OPEN S.I. Signal, Phase A
D.O.	–	DEENERGIZED OPEN
CVI	–	CONTAINMENT VENTILATION ISOLATION (AUTOMATIC SIGNAL)
IVSWS	–	ISOLATION VALVE SEAL WATER SYSTEM

INDIAN POINT UNIT No. 2

UFSAR FIGURE 5.2–29

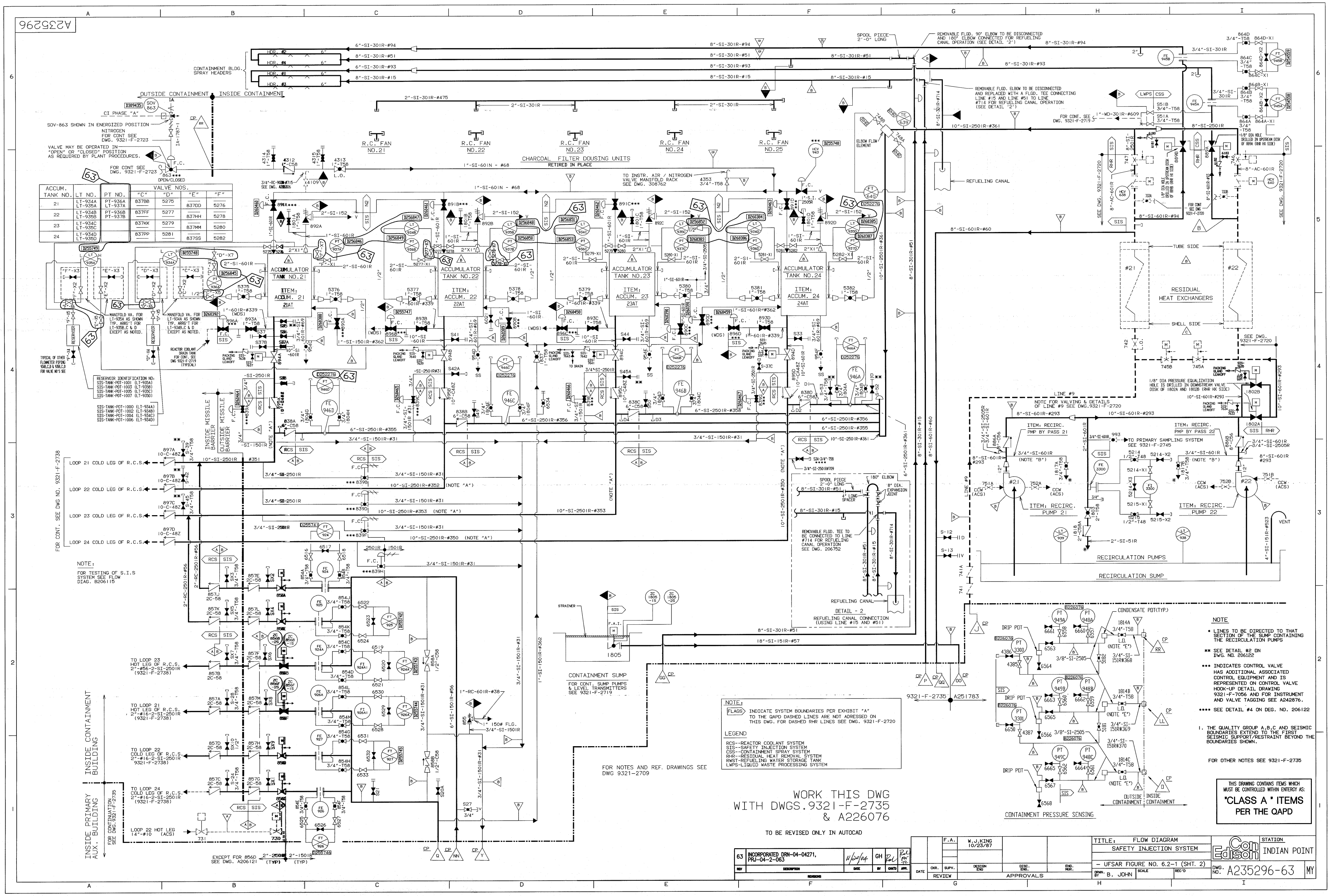
CONTAINMENT ISOLATION SYSTEM PENETRATION SCHEMATICS

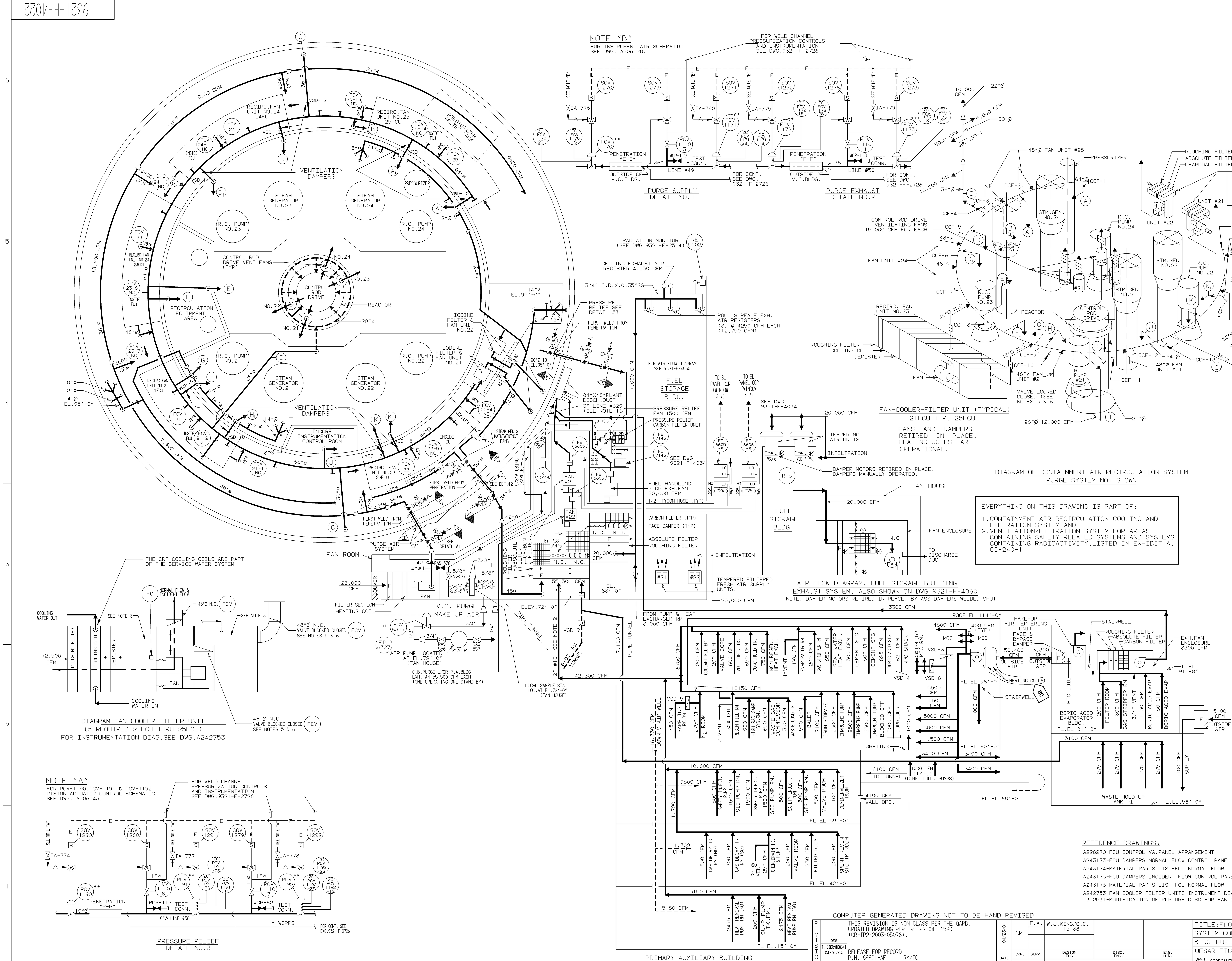
MIC. No. 1999MC3410 | REV. No. 17A

VALVE POSITIONS ARE NOT CONTROLLED BY THIS FIGURE

Figures 5.1-2, 5.1-3, 5.1-4, 5.1-5, 5.1-6, and 5.1-7
Supplied as plant drawings 9321-F-2501, 2502, 2503, 2506, 2507, and 2508.

Redacted by NRC staff as sensitive information.


A235296
CONTAINMENT BLDG. SPRAY HEADERS
OUTSIDE CONTAINMENT
INSIDE CONTAINMENT
SOV-863 SHOWN IN ENERGIZED POSITION
VALVE MAY BE OPERATED IN "OPEN" OR "CLOSED" POSITION AS REQUIRED BY PLANT PROCEDURES.
R.C. FAN NO.21
R.C. FAN NO.22
R.C. FAN NO.23
R.C. FAN NO.24
R.C. FAN NO.25
CHARCOAL FILTER DOUSING UNITS
RETIRED IN PLACE
ACCUMULATOR TANK NO.21
ACCUMULATOR TANK NO.22
ACCUMULATOR TANK NO.23
ACCUMULATOR TANK NO.24
REFUELING CANAL
TUBE SIDE
SHELL SIDE
RESIDUAL HEAT EXCHANGERS
ITEM: RECIRC. PUMP 21
ITEM: RECIRC. PUMP 22
RECIRCULATION PUMPS
RECIRCULATION SUMP
LOOP 21 COLD LEG OF R.C.S.
LOOP 22 COLD LEG OF R.C.S.
LOOP 23 COLD LEG OF R.C.S.
LOOP 24 COLD LEG OF R.C.S.
FOR CONT. SEE DWG NO. 9321-F-2738
NOTE: FOR TESTING OF S.I.S SYSTEM SEE FLOW DIAG. B206115
CONTAINMENT SUMP
CONTAINMENT PRESSURE SENSING
DETAIL - 2 REFUELING CANAL CONNECTION (USING LINE #15 AND #51)
NOTE: FLAGS INDICATE SYSTEM BOUNDARIES PER EXHIBIT "A" TO THE QAPD DASHED LINES ARE NOT ADRESSED ON THIS DWG. FOR DASHED RHR LINES SEE DWG. 9321-F-2720
LEGEND
RCS--REACTOR COOLANT SYSTEM
SIS--SAFETY INJECTION SYSTEM
CSS--CONTAINMENT SPRAY SYSTEM
RHR--RESIDUAL HEAT REMOVAL SYSTEM
RWST-REFUELING WATER STORAGE TANK
LWPS-LIQUID WASTE PROCESSING SYSTEM
FOR NOTES AND REF. DRAWINGS SEE DWG 9321-2709
WORK THIS DWG WITH DWGS.9321-F-2735 & A226076
TO BE REVISED ONLY IN AUTOCAD
NOTE
• LINES TO BE DIRECTED TO THAT SECTION OF THE SUMP CONTAINING THE RECIRCULATION PUMPS
** SEE DETAIL #2 ON DWG. NO. 206122
*** INDICATES CONTROL VALVE HAS ADDITIONAL ASSOCIATED CONTROL EQUIPMENT AND IS REPRESENTED ON CONTROL VALVE HOOK-UP DETAIL DRAWING 9321-F-7056 AND FOR INSTRUMENT AND VALVE TAGGING SEE A242876.
**** SEE DETAIL #4 ON DEG. NO. 206122
1. THE QUALITY GROUP A,B,C AND SEISMIC BOUNDARIES EXTEND TO THE FIRST SEISMIC SUPPORT/RESTRAINT BEYOND THE BOUNDARIES SHOWN.
FOR OTHER NOTES SEE 9321-F-2735
THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN ENTERGY AS: "CLASS A" ITEMS PER THE QAPD
63 INCORPORATED DRN-04-04271, PRJ-04-2-063
TITLE: FLOW DIAGRAM SAFETY INJECTION SYSTEM
- UFSAR FIGURE NO. 6.2-1 (SHT. 2)
DRWN. BY B. JOHN
W.J.KING 10/23/87
ConEdison
STATION INDIAN POINT
DWG. NO. A235296-63
APPROVALS
REVIEW



EVERYTHING ON THIS DRAWING IS PART OF:

1. CONTAINMENT AIR RECIRCULATION COOLING AND FILTRATION SYSTEM-AND
2. VENTILATION/FILTRATION SYSTEM FOR AREAS CONTAINING SAFETY RELATED SYSTEMS AND SYSTEMS CONTAINING RADIOACTIVITY,LISTED IN EXHIBIT A, CI-240-1

AIR DISTRIBUTION SCHEDULE

A + A1	= 58,675 CFM TO R.C.PUMP STM.GEN.& PRESSURIZER AREA.
B	= 2,000 CFM TO PRESS.RELIEF TANK AREA.
C + C1	= 60,000 CFM TO DOME AREA.
D + D1	= 51,275 CFM TO RC.PUMP & STM.GEN.AREA.
E	= 17,500 CFM ACROSS REFUELING POOL AREA.
F	= 10,000 CFM RECIRCULATOIN EQUIPMENT AREA.
G	= 5,000 CFM TO PASSAGE AREA.
H + H1	= 51,275 CFM TO R.C.PUMP & STM.GEN.AREA.
I	= 12,000 CFM AREA UNDER REACTOR.
J	= 5,000 CFM TO INCORE INSTRUMENTATION CONTROL ROOM.
K + K1	= 52,275 CFM TO R.C.PUMPS & STM.GEN.AREA.

LEGEND

N.O.	NORMALLY OPEN (FAIL CLOSED).
N.C.	NORMALLY CLOSED (FAIL OPEN).
	LOUVER
F	FILTER
	DAMPER
	BUTTERFLY VALVE-BUBBLE TIGHT
R	RADIATION
RC	RADIATION CONTROLLER
	RELIEF VALVE
VSD	VENTILATING SYSTEM DAMPER
FI	FLOW INDICATOR
M	MOTOR OPERATOR
	CARBON FILTER
	BACK DRAFT DAMPER
	AIR FOIL DAMPER
**	VALVES FURNISHED WITH DPDT LIMIT SWITCHES IN OPENED AND CLOSED POSITION.

NOTES:

1. 3"-#629 FROM POST ACCIDENT CONTAINMENT VENTING SYSTEM. SEE DWG.B208879.
2. 2"-#131 FROM GAS DECAY TANKS. SEE DWG.9321-F-2730.
3. ABSOLUTE & CHARCOAL FILTERS REMOVED IN F.C.U.'S #21 THRU 25.
4. THE QUALITY GROUP A,B,C AND SEISMIC BOUNDARIES EXTEND TO THE FIRST SEISMIC SUPPORT/RESTRAINT BEYOND THE BOUNDARIES SHOWN.
5. NORMALLY CLOSED INCIDENT FLOW CONTROL VALVES IN F.C.U.'S 21-25 ARE LOCKED CLOSED AND RETIRED IN PLACE
6. ALL F.C.U.'S INCIDENT INLET/OUTLET F.C.V.'S AND ASSOCIATED PANELS ARE RETIRED IN PLACE.

REFERENCE DRAWINGS:

A228270-FCU CONTROL VA.PANEL ARRANGEMENT
A243173-FCU DAMPERS NORMAL FLOW CONTROL PANEL & LOOP DIAG
A243174-MATERIAL PARTS LIST-FCU NORMAL FLOW
A243175-FCU DAMPERS INCIDENT FLOW CONTROL PANEL & LOOP DIAG
A243176-MATERIAL PARTS LIST-FCU NORMAL FLOW
A242753-FAN COOLER FILTER UNITS INSTRUMENT DIAGRAM
312531-MODIFICATION OF RUPTURE DISC FOR FAN COOLER UNITS

THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN ENTERGY AS:
"CLASS A" ITEMS
PER THE QAPD

COMPUTER GENERATED DRAWING NOT TO BE HAND REVISED

REVISION			DATE	CKR.	SUPV.	DESIGN ENG	DISC. ENG.	ENG. MGR.
	DES	THIS REVISION IS NON CLASS PER THE QAPD. UPDATED DRAWING PER ER-IP2-04-16520 (CR-IP2-2003-05078).	04/23/01	SM	F.A.	W.J.KING/G.C. 1-13-88		
	T. CZERNIEWSKI 04/01/04 ENG	RELEASE FOR RECORD P.N. 69901-AF RM/TC						
				REVIEW		APPROVALS		

TITLE:FLOW DIAGRAM VENTILATION SYSTEM CONTAINMENT,PRIMARY AUX. BLDG FUEL STG BLDG
UFSAR FIG. No.'s 5.3-1
DRWN. BY GIBBS&HILL | SCALE NONE | REC'D
Con Edison
STATION INDIAN POINT
DWG. NO. 9321-F-4022-60 MY

CHAPTER 6
ENGINEERED SAFETY FEATURES

6.0 INTRODUCTION

The central safety objective in reactor design and operation is the control of reactor fission products. The following methods are used to ensure this objective:

1. Core design to preclude release of fission products from the fuel (Chapter 3).

2. Retention of fission products in the reactor coolant for whatever leakage occurs (Chapters 4 and 6).

3. Retention of fission products by the containment for operational and accidental releases beyond the reactor coolant boundary (Chapters 5 and 6).

4. Optimizing fission product dispersal to minimize population exposure (Chapters 2 and 11).

The engineered safety features are the provisions in the plant that embody methods 2 and 3 above to prevent the occurrence or to ameliorate the effects of serious accidents.

The engineered safety features systems in this plant are the containment system, detailed in Chapter 5, the safety injection system, detailed in Section 6.2, the containment spray system, detailed in Section 6.3, the containment air recirculation cooling system, detailed in Section 6.4, the isolation valve seal-water system, detailed in Section 6.5, and the containment penetration and weld channel pressurization system, detailed in Section 6.6.

Evaluations of techniques and equipment used to accomplish the central objective including accident cases are detailed in Chapters 5, 6, and 14.

6.1 GENERAL DESIGN CRITERIA

Criteria applying in common to all engineered safety features are given in Section 6.1.1. Thereafter, criteria that are related to engineered safety features, but which are more specific to other plant features or systems, are listed and cross referenced in Section 6.1.2.

6.1.1 Engineered Safety Features Criteria

6.1.1.1 Engineered Safety Features Basis for Design

Criterion: Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. Such engineered safety features shall be designed to cope with any size reactor coolant piping break up to and including the equivalent of a circumferential rupture of any pipe in that boundary, assuming unobstructed discharge from both ends. (GDC 37)

The design, fabrication, testing, and inspection of the core, reactor coolant pressure boundary, and their protection systems give assurance of safe and reliable operation under all anticipated normal, transient, and accident conditions. However, engineered safety features are provided in

the facility to back up the safety provided by these components. These engineered safety features have been designed to cope with any size reactor coolant pipe break up to and including the circumferential rupture of any pipe assuming unobstructed discharge from both ends as discussed in Section 14.3.3.3. They are also designed to cope with any steam or feedwater line break up to and including the main steam or feedwater lines as discussed in Section 14.2.5.

Limiting the release of fission products from the reactor fuel is accomplished by the safety injection system, which by cooling the core, keeps the fuel in place and substantially intact and limits the metal water reaction to an insignificant amount.

The safety injection system consists of high- and low-head centrifugal pumps driven by electric motors and of passive accumulator tanks that are self-energized and which act independently of any actuation signal or power source.

The release of fission products from the containment is limited in three ways:

1. Blocking the potential leakage paths from the containment. This is accomplished by:

   a. A steel-lined, reinforced-concrete reactor containment with testable, double-sealed penetrations, and liner weld channels, the spaces of which are continuously pressurized above accident pressure and which form a virtually leaktight barrier to the escape of fission products should a loss-of-coolant accident occur. (Section 6.6.2 lists those portions of the Weld Channel Pressurization System that have been disconnected because repairs have been determined not to be practical.)

   b. Isolation of process lines by the containment isolation system, which imposes double barriers in each line that penetrates the containment except for lines used during the accident. Pipes penetrating the containment are sealed as shown in Table 5.2-1. This table presents the sealing method for all containment piping penetrations and valving.

2. Reducing the fission product concentration in the containment atmosphere. This is accomplished by containment spray, which removes elemental iodine vapor and particulates from the containment atmosphere by washing action. The spray is chemically treated during the recirculation phase to enhance iodine retention.

3. Reducing the containment pressure and thereby limiting the driving potential for fission product leakage. This is accomplished by cooling the containment atmosphere by the following independent systems of approximately equal heat removal capacity that together also function to ensure the containment design criteria is maintained even with an assumed single failure:

   a. Containment spray system.
   b. Containment air recirculation cooling system.

6.1.1.2 Reliability and Testability of Engineered Safety Features

Criterion: All engineered safety features shall be designed to provide such functional reliability and ready testability as is necessary to avoid undue risk to the health and safety of the public. (GDC 38)

A comprehensive program of plant testing was formulated for all equipment, systems, and system control vital to the functioning of engineered safety features. The program consisted of performance tests of individual pieces of equipment in the manufacturer's shop, and integrated tests of the systems as a whole. Periodic tests of the actuation circuitry and mechanical components ensure reliable performance, upon demand, throughout the life of the plant.

The initial tests of individual components and the integrated test of the system as a whole complemented each other to ensure the performance of the system as designed and to prove proper operation of the actuation circuitry.

Routine periodic testing of the engineered safety features components is performed, in accordance with the Technical Specifications.

6.1.1.3 Missile Protection

Criterion: Adequate protection for those engineered safety features, the failure of which could cause an undue risk to the health and safety of the public, shall be provided against dynamic effects and missiles that might result from plant equipment failures. (GDC 40)

A loss-of-coolant accident or other plant equipment failure might result in dynamic effects or missiles. For engineered safety features that are required to ensure safety in the event of such an accident or equipment failure, protection is provided primarily by the provisions that are taken in the design to prevent the generation of missiles. In addition, protection is also provided by the layout of plant equipment or by missile barriers in certain cases. (Refer to Section 5.1.2 for a discussion of missile protection.)

Injection paths leading to unbroken reactor coolant loops are protected against damage resulting from the maximum reactor coolant pipe rupture by layout and structural design considerations. Injection lines penetrate the main missile barrier, which is the crane wall, and the injection headers are located in the missile protected area between the crane wall and the containment wall. Individual injection lines, connected to the injection header, pass through the barrier and then connect to the loops. The separation of the individual injection lines is provided to the maximum extent practicable. The movement of the injection line, associated with a rupture of a reactor coolant loop, is accommodated by line flexibility and by the design of the pipe supports such that no damage outside the missile barrier is possible.

The containment structure is capable of withstanding the effects of missiles originating outside the containment and which might be directed toward it so that no loss-of-coolant accident can result.

All hangers and anchors are designed in accordance with USAS B31.1, Code for Pressure Piping. This code provides minimum requirements on material, design, and fabrication with ample safety margins for both dead and dynamic loads over the life of the equipment. Concrete

missile barriers, bumpers, walls and other concrete structures are designed in accordance with ACI 318-63, Building Code Requirements for Reinforced Concrete.

In 1989, the NRC approved changes to the design basis with respect to dynamic effects of postulated primary loop ruptures, as discussed in Section 4.1.2.4

6.1.1.4 Engineered Safety Features Performance Capability

Criterion: Engineered safety features, such as the emergency core cooling system and the containment heat removal system, shall provide sufficient performance capability to accommodate the failure of any single active component without resulting in undue risk to the health and safety of the public. (GDC 41)

Each engineered safety feature provides sufficient performance capability to accommodate any single failure of an active component and still function in a manner to avoid undue risk to the health and safety of the public.

The extreme upper limit of public exposure is taken as the levels and time periods presently outlined in 10 CFR 50.67 (i.e., 25 rem total effective dose equivalent (TEDE) at the exclusion radius for the worst two hour interval, 25 rem TEDE over the duration of the accident at the low-population-zone distance and 5 rem TEDE for the operators in the control room for the duration of the accident). The accident condition considered is the hypothetical case of a release of fission products per NUREG-1465. Also, the total loss of all outside power is assumed concurrently with this accident. With minimum engineered safety features systems functioning, the offsite exposure would be within 10 CFR 50.67 limits as discussed in Section 14.3.6.

Under these accident conditions, the containment air recirculation cooling system and the containment spray system are designed and sized so that both systems, each operating with partial effectiveness, are able to supply the necessary postaccident cooling capacity to ensure the maintenance of containment integrity, that is, keeping the pressure below design pressure at all times assuming that the core residual heat is released to the containment as steam. Partial effectiveness is defined as the operation of a system with at least one active component failure. Containment spray relies on a sufficient amount of passive sodium tetraborate stored in containment to raise the pH of the recirculating solution for continued iodine removal following an accident. The containment spray system alone is able to supply the post accident iodine removal required to restrict the offsite exposure to within 10 CFR 50.67 limits.

6.1.1.5 Engineered Safety Features Components Capability

Criterion: Engineered safety features shall be designed so that the capability of these features to perform their required function is not impaired by the effects of a loss-of-coolant accident to the extent of causing undue risk to the health and safety of the public. (GDC 42)

Instrumentation, pumps, fans, cooling units, valves, motors, cables, and penetrations located inside the containment are selected to meet the most adverse accident conditions to which they may be subjected. These items are either protected from containment accident conditions or are designed to withstand, without failure, exposure to the worst combination of temperature, pressure, and humidity expected during the required operational period.

In response to NRC Generic Letter 95-07, safety-related power-operated gate valves have been evaluated for susceptibility to pressure locking and thermal binding. The results of this evaluation identified that those potential conditions will not prevent the plant from achieving a safe shutdown, as all valves evaluated remain operable. This conclusion is based upon valve design, plant configuration during normal, accident, and post accident operating modes and sufficient actuator thrust to open the valve. The details of the system and valve evaluations are documented in Reference 1.

In response to NRC Generic Letter 96-06, isolated pipe line segments that penetrate containment have been analyzed to evaluate their susceptibility to overpressurization caused by thermal expansion of the contained fluid in the event of a design basis accident. The results show that potential overpressurization will not cause lines to fail; all remain operable. Those that are protected by safety relief devices were further evaluated for the effects of stuck-open relief valves under accident conditions. No failure modes were identified that would adversely affect the ability for safety-related systems to perform their intended functions during accidents.

The safety injection system pipes serving each loop are restrained in such a manner as to restrict potential accident damage to the portion of piping downstream of the crane wall that constitutes the missile barrier in each loop area. The restraints are designed to withstand, without failure, the thrust force of any branch line severed from the broken reactor coolant pipe and discharging fluid to the atmosphere and to withstand a bending moment equivalent to that which produces a failure of the piping under the action of free-end discharge to the atmosphere or a motion of the broken reactor coolant pipe to which the injection pipes are connected. This prevents possible failure at any point upstream from the support point, including the branch line connection into the piping header.

In 1989, the NRC approved changes to the design basis with respect to dynamic effects of postulated primary loop ruptures, as discussed in Section 4.1.2.4.

Designated valves that are located in areas that would have excessive radiation levels in the event of a release of fission products from the core are provided with capability for remote operation.

6.1.1.6 Accident Aggravation Prevention

Criterion: Protection against any action of the engineered safety features, which would accentuate significantly the adverse after effects of a loss of normal cooling shall be provided. (GDC 43)

The introduction of borated cooling water into the core results in a negative reactivity addition. The control rods insert and remain inserted.

The supply of water by the safety injection system to cool the core cladding does not produce significant metal-water reaction (<1.0-percent).

The delivery of cold safety injection water to the reactor vessel following accidental expulsion of reactor coolant does not cause further loss of integrity of the reactor coolant system boundary.

6.1.1.7 Sharing of Systems

Criterion: Reactor facilities may share systems or components if it can be shown that such sharing will not result in undue risk to the health and safety of the public. (GDC 4)

The engineered safety features at Indian Point 2 do not share systems or components with other units.

6.1.2 Related Criteria

The following are criteria, which although related to all engineered safety features, are more specific to other plant features or systems, and, therefore, are discussed in other chapters, as listed.

Name	Discussion
Quality Standards (GDC 1)	Chapter 4
Performance Standards (GDC 2)	Chapter 4
Records Requirements (GDC 5)	Chapter 4
Instrumentation and Control Systems (GDC 12)	Chapter 7
Engineered Safety Features Protection Systems (GDC 15)	Chapter 7
Emergency Power (GDC 39)	Chapter 8

REFERENCES FOR SECTION 6.1

1. NRC Letter dated May 20, 1999, Jeffrey F. Harold to A. Alan Blind, Subject: Safety Evaluation of Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," for Indian Point Nuclear Generating Unit No.2 (TAC M93473).

6.2 SAFETY INJECTION SYSTEM

At Indian Point Unit 2 the emergency core cooling function is performed by the safety injection system. Therefore, whenever the term "emergency core cooling system" or ECCS is referenced in the document it is synonymous with the safety injection system.

6.2.1 Design Basis

6.2.1.1 Emergency Core Cooling System Capability

Criterion: An emergency core cooling system with the capability for accomplishing adequate emergency core cooling shall be provided. This core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to acceptable amounts for all sizes of breaks in the reactor coolant piping up to the equivalent of a double-ended rupture of the largest pipe. The performance of such emergency core cooling system shall be evaluated conservatively in each area of uncertainty. (GDC 44)

Adequate emergency core cooling is provided by the safety injection system (which constitutes the emergency core cooling system) whose components operate in three modes. These modes are delineated as passive accumulator injection, active safety injection, and residual heat removal recirculation.

The primary purpose of the safety injection system is the automatic delivery of cooling water to the reactor core in the event of a loss-of-coolant accident. This limits the fuel clad temperature and thereby ensures that the core will remain intact and in place, with its essential heat transfer geometry preserved. This protection is afforded for:

1. All pipe break sizes up to and including the hypothetical instantaneous circumferential rupture of a reactor coolant loop, assuming unobstructed discharge from both ends.

2. A loss of coolant associated with the rod ejection accident.

3. A steam-generator tube rupture.

The basic design criteria for loss-of-coolant accident evaluations prior to codification under 10 CFR 50.46 were as follows:

1. The cladding temperature is to be less than:

    a. The melting temperature of Zircaloy-4.
    b. The temperature at which gross core geometry distortion, including clad fragmentation, may be expected.

2. The total core metal-water reaction will be limited to less than 1-percent.

These criteria ensure that the core geometry remains in place and substantially intact to such an extent that effective cooling of the core is not impaired.

Subsequently, the basic design criteria for loss-of-coolant accident calculations have been revised to those required under 10 CFR 50.46.

For any rupture of a steam pipe and the associated uncontrolled heat removal from the core, the safety injection system adds shutdown reactivity so that with a stuck rod, no offsite power, and minimum engineered safety features, there is no consequential damage to the reactor coolant system and the core remains in place and coolable as discussed in Section 14.2.5.

Redundancy and segregation of instrumentation and components are incorporated to ensure that postulated malfunctions will not impair the ability of the system to meet the design objectives. The system is effective in the event of loss of normal station auxiliary power coincident with the loss of coolant, and is tolerant of failures of any single component or instrument channel to respond actively in the system. During the recirculation phase, the system is tolerant of a loss of any part of the flow path since backup alternative flow path capability is provided as described in Section 6.2.3.3.

The ability of the safety injection system to meet its capability objectives is presented in Section 6.2.3. The analysis of the accidents is presented in Chapter 14.

6.2.1.2 Inspection of Emergency Core Cooling System

Criterion: Design provisions shall, where practical, be made to facilitate inspection of physical parts of the emergency core cooling system, including reactor vessel internals and water injection nozzles. (GDC 45)

Design provisions are made to the extent practical to facilitate access to the critical parts of the reactor vessel internals, pipes, valves, and pumps for visual or boroscopic inspection for erosion, corrosion, and vibration-wear evidence and for nondestructive test inspection where such techniques are desirable and appropriate.

6.2.1.3 Testing of Emergency Core Cooling System Component

Criterion: Design provisions shall be made so that components of the emergency core cooling system can be tested periodically for operability and functional performance. (GDC 46)

The design provides for periodic testing of active components of the safety injection system for operability and functional performance.

Power sources are arranged to permit individual actuation of each active component of the safety injection system.

The safety injection pumps and residual heat removal pumps can be tested periodically during plant operation using the minimum flow recirculation lines provided. The residual heat removal pumps are used every time the residual heat removal loop is put into operation. All remote-operated valves can be exercised, and actuation circuits can be tested either during normal operation or routine plant maintenance.

6.2.1.4 Testing of Emergency Core Cooling System

Criterion: Capability shall be provided to test periodically the operability of the emergency core cooling system up to a location as close to the core as is practical. (GDC 47)

An integrated system test can be performed when the plant is cooled down and the residual heat removal loop is in operation. This test would not introduce flow into the reactor coolant system, but would demonstrate the operation of the valves, pump circuit breakers, and automatic circuitry upon the initiation of safety injection.

Level and pressure instrumentation is provided for each accumulator tank, and accumulator tank pressure and level are continuously monitored during plant operation. Flow from the tanks can be checked at any time using test lines as described in Section 6.2.5.3.1.

6.2.1.5 Testing of Operational Sequence of Emergency Core Cooling System

Criterion: Capability shall be provided to test initially, under conditions as close as practical to design, the full operational sequence that would bring the emergency core cooling system into action, including the transfer to alternate power sources. (GDC 48)

The design provides for the capability to test initially, to the extent practical, the full operational sequence up to the design conditions for the safety injection system to demonstrate the state of readiness and capability of the system. Details of the operational sequence testing are presented in Section 6.2.5.

6.2.1.6 Codes and Classifications

Table 6.2-1 lists the codes and standards to which the safety injection system components are designed.

6.2.1.7 Service Life

All portions of the system located within the containment are designed to operate without benefit of maintenance and without loss of functional performance for the duration of time the component is required. Per the 12/06/04 NRC generic SER on NEI 04-07 (Reference 5), evaluations of PWR post accident emergency recirculation performance to address the potential impact of debris blockage per GSI-191 (Reference 4) and Generic Letter 2004-02 (Reference 3) will use a mission time of 30 days.

6.2.2 System Design And Operation

6.2.2.1 System Description

Adequate emergency core cooling following a loss-of-coolant accident is provided by the safety injection system shown in Plant Drawing 9321-2735 [Formerly UFSAR Figure 6.2-1]. Plant Drawing 235296 [Formerly UFSAR Figures 6.2-2] and Figures 6.2-2 through 6.2-5 depict how this system concept is translated into plant layout design. The system components operate in the following possible modes:

1. Injection of borated water by the passive accumulators.

2. Injection by the safety injection pumps drawing borated water from the refueling water storage tank.

3. Injection by the residual heat removal pumps also drawing borated water from the refueling water storage tank.

4. Recirculation of spilled reactor coolant, injected water, and containment spray system drainage back to the reactor from the recirculation sump by the recirculation pumps. The residual heat removal pumps provide backup recirculation capability through the independent containment sump as described in Section 6.2.3.3.

The initiation signal for core cooling by the safety injection pumps and the residual heat removal pumps is the safety injection signal, which is described in Section 7.2.3.2.3.

6.2.2.1.1 Injection Phase

The principal components of the safety injection system, which provide emergency core cooling immediately following a loss of coolant are the accumulators (one for each loop), the three safety injection (high-head) pumps, and the two residual heat removal (low-head) pumps. The safety

safety injection and residual heat removal pumps are located in the primary auxiliary building.

The accumulators, which are passive components, discharge into the cold legs of the reactor coolant piping when pressure decreases to the minimum Technical Specification value, thus rapidly ensuring core cooling for large breaks. They are located inside the containment, but outside the crane wall, therefore each is protected against possible missiles.

The safety injection signal opens certain of the safety injection system isolation valves, provides confirmatory open signals to system isolation valves that are normally open, and starts the safety injection pumps and residual heat removal pumps.

The three safety injection pumps (high-head) deliver borated water to two separate discharge headers. The flow from the discharge headers can be injected into the four cold legs and two hot legs of the reactor coolant system. The motor-operated isolation valves in the four cold-leg injection lines are open during normal plant operation. The motor-operated isolation valves in the two hot-leg injection lines are closed during normal plant operation. The hot-leg injection lines are provided for later use during hot-leg recirculation following a reactor coolant pressure boundary break. The high-head safety injection system is configured with two cold leg injection lines physically connected to the reactor coolant pressure boundary and the other two lines connected to the accumulator discharge lines upstream of the pressure boundary. Since a small break in the reactor coolant pressure boundary can include a cold leg injection line, safety injection flow capability can be limited by the resulting flow from only three intact cold leg injection lines. Depending on the assumed single failure, either two or three safety injection pumps can be operating. To maximize the fraction of safety injection flow delivered to the reactor coolant system with a broken cold leg injection line, the four cold leg injection lines are flow balanced to within an allowable range. The resulting system flow capability is sufficient for the makeup of coolant following a small break that does not immediately depressurize the reactor coolant system to the accumulator discharge pressure. Credit is not taken for operator action to isolate a broken cold leg injection line.

For large breaks, the reactor coolant system would be depressurized and voided of coolant rapidly (about 26 sec for the largest break as shown in Figure 14.3-12) and a high flow rate is required to recover quickly the exposed fuel rods and limit possible core damage as discussed in Section 14.3.3.3.1. To achieve this objective, one residual heat removal pump and two safety injection pumps are required to deliver borated water to the cold legs of the reactor coolant loops. Two residual heat removal and three safety injection pumps are available to provide for an active component failure. Delivery from these pumps supplements the accumulator discharge. Since the reactor coolant system backpressure is relatively low (rapid depressurization for large breaks), a broken injection line would not appreciably change the flows in the other injection line's delivery to the core.

The residual heat removal pumps take suction from the refueling water storage tank. In addition, the charging pumps of the chemical and volume control system are available but are not required to augment the flow of the safety injection system.

Because the injection phase of the accident is terminated before the refueling water storage tank is completely emptied, all pipes are kept sufficiently filled with water before recirculation is initiated to ensure the systems remain operable and perform properly. Water level indication and alarms on the refueling water storage tank give the operator ample warning to terminate the injection phase. Additional level indicators and alarms are provided in the recirculation and

containment sumps, which also give backup indication when injection can be terminated and recirculation initiated.

6.2.2.1.2 Recirculation Phase

After the injection operation, coolant spilled from the break and water collected from the containment spray are cooled and returned to the reactor coolant system by the recirculation system.

When the break is large, depressurization occurs due to the large rate of mass and energy loss through the break to containment. In the event of a large break, the recirculation flow path is within the containment. The system is arranged so that the recirculation pumps take suction from the recirculation sump in the containment floor and deliver spilled reactor coolant and borated refueling water back to the core through the residual heat exchangers. The system is also arranged to allow either of the residual heat removal pumps to take over the recirculation function. The residual heat removal pumps would only be used if backup capacity to the internal recirculation loop is required as described in Section 6.2.3.3. Water is delivered from the containment to the residual heat removal pumps from the separate containment sump inside the containment.

For small breaks, the depressurization of the reactor coolant system is augmented by steam dump from and auxiliary feedwater addition to the steam generators. For the smaller breaks in the reactor coolant system where recirculated water must be injected against higher pressures for long-term cooling, the system is arranged to deliver the water from residual heat removal heat exchanger 21 to the high-head safety injection pump suction and by this external recirculation route to the reactor coolant loops. If this flow path is unavailable, an alternate flow path is provided as indicated in Table 6.2-11. Thus, if depressurization of the reactor coolant system proceeds slowly, the safety injection pumps may be used to augment the flow-pressure capacity of the recirculation pumps in returning the spilled coolant to the reactor. In this system configuration, the recirculation pump (or residual heat removal pump) provides flow and net positive suction head to the operating safety injection pumps. To prevent safety injection pump flow in excess of its maximum allowable (i.e., runout) limit, variable flow orifices are installed at the discharge of the safety injection pumps and the hot and cold leg motor-operated isolation valves are preset with mechanical stops based on data from operational flow testing to limit system maximum flow capability.

The recirculation pumps, the residual heat removal heat exchangers, piping, and valves vital to the function of the recirculation loop are located in a missile-shielded space inside the polar crane support wall on the west side of the reactor primary shield.

There are two sumps within the containment, the recirculation sump and the containment sump. Both sumps collect liquids discharged into the containment during the injection phase of the design-basis accident.

As part of the resolution of GSI-191 and Generic Letter 2004-02, various flow channeling barriers are installed in the Vapor Containment, EL 46'-0" to force the recirculation flow into the Reactor Cavity Sump area, up and out the Incore Instrumentation Tunnel, through the Crane Wall via the three nominal 20 inch square openings and into the annulus area outside the Crane Wall. The recirculation flow will migrate towards the Recirculation Sump Strainer or the Containment Sump Strainer depending on which pump(s) are operating. Flow channeling barriers are installed on the reactor Cavity Platform, EL 29'-4", around the Incore Instrumentation Tunnel, on the

Instrumentation Tunnel, on the Recirculation Sump Trenches, at the Containment Sump, and on Crane Wall penetrations up to the flood level. Flow channeling barrier doors are installed in the Northeast and Northwest quadrant openings of the Crane Wall. In addition, flow channeling barrier doors are installed in the North and South entrances to the Recirculation Sump area. Perforated plate is installed on the RHR Heat Exchanger Platform, EL 66'-0" to preclude debris from washing through the existing grating and into the Recirculation Sump area. Forcing the recirculation flow path into the Reactor Cavity Sump area (a low velocity zone) allows the larger debris an opportunity to settle.

The Recirculation Sump and Containment Sump strainers consist of a matrix of multi-tube top-hat modules, which are fabricated from perforated stainless steel plate and mounted in the horizontal position. The perforated plate has 3/32" diameter holes sized to limit downstream affects. The top-hat modules have four (4) layers of perforated surfaces for straining debris from the sump fluid. Typical Recirculation Sump and Containment Sump strainer top-hat modules consist of a 12-1/2" diameter outer perforated tube with a respective 10-1/2" diameter inner perforated tube and a second set of tubes, which consist of a 7-1/2" diameter outer perforated tube with a respective 5-1/2" diameter inner perforated tube. The top-hat modules feature an internal vortex suppressor, which prevents air ingestion into the piping system. Stainless steel mesh has been installed between each pair of perforated plate tubes to minimize fiber bypass through the strainers. The top-hat modules are attached to strainer water boxes. Frame structures supporting sections of grating are installed above the Internal Recirculation and Containment Sump strainers including the sump strainer extension in the containment annulus providing for additional vortex suppression function. The Containment Sump Level Detection System is discussed in Section 6.7.2.13.

The Recirculation Sump relies on two, connected water boxes with 249 top-hat modules in the sump pit for the purpose of preventing particles greater than 3/32" in diameter from entering the suction of the recirculation pumps. The recirculation sump strainer has effective surface area of ~3,156 square feet and an effective interstitial volume of ~476 cubic feet. Water will enter the top-hat modules through the perforated plates and flow through the stainless steel mesh inside either of the two (2) annuli flow paths within each top-hat module. Upon exiting the top-hat modules, water will flow into either of the two connected strainer water boxes, flow over the Recirculation Sump weir wall and into the Recirculation Pump Bay towards the pumps.

The Containment Sump relies on a water box with 23 top-hat modules in the Containment Sump pit and a plenum extension out into the annulus with a water box with an additional 40 top-hat modules for the purpose of preventing particles greater than 3/32" in diameter from entering the Containment Sump suction line to the RHR Pumps. The Containment Sump strainer has a combined effective surface area of ~1182 square feet (~412 sq. ft. for sump pit and ~770 sq. ft. for the annulus extension) and an effective combined interstitial volume of ~161 cubic feet. Water will enter the top-hat modules through the perforated plates and flow through the stainless steel mesh inside either of the two (2) annuli flow paths within each top-hat module. Upon exiting the top-hat modules, water will flow into the strainer water box, which is connected to the Containment Sump suction line and the RHR System. The containment sump level detection system is discussed in Section 6.7.1.2.13.

Each sump strainer is qualified to handle the post LOCA design basis accident debris loads predicted by the mechanistic evaluations required by GL 2004-02. There are two classifications for the debris generated by an RCS break: 1) conventional debris (e.g., insulation, tags, coatings, dust and dirt), 2) chemical debris (principally the precipitation of Aluminum based compounds Sodium Aluminum Silicate and Aluminum Oxy Hydroxide) which are conservatively

predicted by use of a model detailed in WCAP-16530-NP-A (Reference 6). An Argonne National Laboratory (ANL) formula was used to predict the post-LOCA chemical precipitation temperature. The precipitation temperature is determined from the post-accident containment sump pool conditions (Aluminum concentration, temperature and pH). Chemical precipitants are not predicted to develop prior to the required switchover to hot-leg recirculation. Consequently, the internal recirculation sump strainer qualification uses predicted head losses associated with conventional debris loads up to the switchover to hot-leg recirculation and then conventional and chemical debris loads after the transfer to hot-leg recirculation occurs when reduced sump flow rates are expected to be less than two HHSI pumps at runout (2 x 675 GPM).

Per the 12/06/04 NRC generic SER on NEI 04-07 (Reference 5), evaluations of PWR post accident emergency recirculation performance to address the potential impact of debris blockage per GSI-191 (Reference 4) and Generic Letter 2004-02 (Reference 3) will use a mission time of 30 days. The internal recirculation sump strainers are qualified for a GL 2004-02 defined 30-day mission time. The containment sump strainers were qualified using the same methodology but are qualified from 24 hours post large break LOCA until the end of the 30 day mission time. The containment sump is not required to handle the same full debris loads at the start of the recirculation phase as the recirculation sump since the only postulated failure that would require its use is a passive failure, which is only postulated after 24 hours into the event (reference Technical Specification Amendment #257). However, to maintain redundancy for the more probable small break LOCAs, the containment sump strainers have been qualified for 6 inch diameter breaks and smaller from the start of the recirculation phase. A condition of Amendment #257 was that the Emergency Operating Procedures continue to utilize the containment sump as an alternative path should both the recirculation sump trains become unavailable.

As identified above in the strainer descriptions, both sump strainers are constructed of concentric cylindrical tubes perforated with 3/32 inch diameter holes and have stainless steel mesh behind the perforations to reduce the quantity of fine fibers able to pass through the strainer perforations, although some fine fibers and particulates may still pass through the strainer. The passive sump strainers provide adequate protection to the downstream components from the majority of the accident generated debris. As part of the resolution of GL 2004-02, an analysis of the components downstream of the strainers required for accident mitigation was performed to ensure satisfactory operation for the defined 30-day mission time. Pumps, isolation and throttle valves, orifices, instrument connections, and piping were examined using the guidance provided by revision 1 of WCAP-16406 (Reference 7) with justification provided for any methodology deviations. All equipment was found to have sufficient clearance: to allow passage of debris, to limit blockage to an acceptable level, and / or to have sufficient resistance to wear as to not affect there function for the defined 30-day mission time. Chemical effects on the fuel elements were also examined and not predicted to interfere with heat transfer per analysis based on WCAP-16793-NP (Reference 8).

The low-head external recirculation loop via the containment sump line and the residual heat removal pumps provides backup recirculation capability to the low-head internal recirculation loop. The containment sump line has two remote motor-operated normally closed valves located outside the containment and a remote motor-operated butterfly valve inside containment. The high-head external recirculation flow path via the high-head safety injection pumps is required for the range of small-break sizes for which the reactor coolant system pressure remains in excess of the shutoff head of the recirculation pumps at the end of the injection phase. The recirculation pumps, or residual heat removal pumps if backup capability is required, are also used to provide

required, are also used to provide flow to the high-head safety injection pumps during hot leg recirculation.

The external recirculation flow paths within the primary auxiliary building are designed so that external recirculation can be initiated immediately after the accident. Those portions of the safety injection system outside of the containment, which are designed to circulate, under postaccident conditions, radioactivity contaminated water collected in the containment meet the following requirements:

1. Shielding to limit radiation levels.
2. Collection of discharges from pressure-relieving devices into closed systems.
3. Means to detect and control radioactivity leakage into the environs.

These criteria are met by minimizing leakage from the system. External recirculation loop leakage is discussed in Section 6.2.3.8. The radiological consequences of external recirculation loop leakage following a design basis accident are presented in Section 14.3.6.6. Detection and control of leakage from external recirculation loop components is also discussed in Section 6.7.

One recirculation pump and one residual heat exchanger of the recirculation system provide sufficient cooled recirculated water to keep the core flooded with water by injection through the cold-leg connections while simultaneously providing, sufficient containment recirculation spray flow to reduce containment airborne activity. These systems are kept sufficiently filled with water to ensure the systems remain operable and perform properly. Three of the five fan cooler units prevent the containment pressure from rising above design limit. Analysis demonstrates that flow will be determined by system resistance provided by the physical configuration of the recirculation piping and components, and will be hydraulically balanced such that sufficient flow is established to the core and the spray header. Only one pump and one heat exchanger are required to operate for this capability at the earliest time recirculation spray is initiated. With both recirculation pumps in operation and both spray header valves open, a recirculation spray flow rate can be established such that no containment cooling fans (Section 6.4) are required. Likewise with five containment cooling units in operation, no containment spray is required to maintain containment pressure below its design limit. The system is also arranged to allow either of the residual heat removal pumps to take over the recirculation function following a passive failure as defined in Section 6.2.3.3. This design ensures that heat removal from the core and containment is effective in the event of a pipe or valve body rupture.

6.2.2.1.3 Cooling Water

The service water system (Section 9.6) provides cooling water to the component cooling loop, which in turn cools the residual heat exchangers, both of which are part of the auxiliary coolant systems (Section 9.3). Three non-essential service water pumps are available to take suction from the river and discharge to the two component cooling heat exchangers. Three component cooling pumps are available to discharge through their heat exchangers and deliver to the two residual heat exchangers. During the recirculation phase following a loss-of-coolant-accident, only one residual heat removal heat exchanger, one recirculation or residual heat removal pump, one non-essential service water pump, one component cooling water pump, one auxiliary component cooling water pump, and one component cooling water heat exchanger are required to meet the core-cooling function. The auxiliary component cooling water pump is required only to support the function of the recirculation pump. With the exception of the residual heat removal heat exchangers and the recirculation pumps, all of the cited equipment is located outside of the

outside of the containment.

6.2.2.1.4 Changeover From Injection Phase to Recirculation Phase

Assuming that the three high-head safety injection pumps, the two residual heat removal pumps, and the two containment spray pumps (Section 6.3) are running at their maximum capacity, the time sequence, from the time of the safety injection signal, for the changeover from injection to recirculation in the core of a large rupture is as follows:

1. In approximately 15 min, sufficient water has been delivered to provide the required net positive suction head to start the recirculation pumps.

2. In approximately 20 min, (a) low-level alarms on the refueling water storage tank sound, and (b) the redundant containment and recirculation sump level indicators show the sump water level. The alarm(s) serve to alert the operator to start the switchover to the recirculation mode. The redundant containment and recirculation sump level indicators provide verification that the refueling water storage tank water has been delivered during the injection phase, as well as giving consideration to the case of a spurious (i.e., early) refueling water storage tank low-level alarm. The operator would see on the control board that the redundant sump level indications are at the appropriate points; switchover via the eight-switch sequence is performed at that time.

3. With the initiation of the eight-switch sequence (i.e., switch No. 1), only one spray pump will continue in operation. This spray pump will continue to draw from the refueling water storage tank until the level drops below 2 feet.

Recirculation pump motors are 2-ft 2-in. above the highest water level after the addition of the injected water to the spilled coolant.

The changeover from injection to recirculation takes place when the level indicator or level alarms on the refueling water storage tank indicate that the fluid has been injected. The level indicators in the containment sump will also verify that the level is sufficient within the containment. The sequence is followed regardless of which power supply is available. All switches are grouped together on the safeguard control panel. The component position lights verify when the function of a given switch has been completed. Should an individual component fail to respond, the operator can take corrective action to secure appropriate response from the backup component. The manual switchover by the operator:

1. Terminates safety injection signal in order that the control logic permits manipulation of the system (at any time following completion of the auto-start sequence).

2. Closes switches one and three (removes and isolates unnecessary loads from the diesels).

   Switch One:

   a. Trips one (i.e., pump 22) of three high-head safety injection pumps if all three are operating (no action if two are operating), and isolates the pump suction to

to the refueling water storage tank if the tripped pump is the middle safety injection pump (i.e. pump 22).

b. Trips one containment spray pump if both are operating (no action if one is operating).

c. Closes isolation valves at the inoperative spray pump discharge.

Switch Three:

a. Trips both residual heat removal pumps.

b. Sends close signals to isolation valves in the residual heat removal pump suction and discharge headers, which are administratively reenergized later in the sequence. (Technical Specifications require the motor operators for these valves to be deenergized.)

3. Closes switch two (establishes cooling flow for residual heat removal heat exchangers)

   a. Starts one service water pump, non-essential header (the second or third pump is given a start signal if the first or second pump fails to start).

   b. Starts one component cooling pump (the second or third pump is given a start signal if the first or second pump fails to start).

4. Isolates one RHR heat exchanger flow path. (if both are open)

5. Closes switch four (initiates internal recirculation flow).

   a. Opens valves on discharge of recirculation pumps.

   b. Starts recirculation pump 21 (if pump 21 fails to start, uses manual start on pump 22). (Pump 22 control switch is adjacent to switch four).

6. Checks flow to reactor coolant system via the low-head injection lines to ensure minimum flow requirements are established. If minimum flow requirements are established, the closes switch seven and switch eight to establish low-head recirculation.

   If minimum flow requirements are not established, then closes switch six and switch eight to establish high-head recirculation.

7. Close switch six (supplies recirculation for reactor coolant system pressures greater than 150 psig, which impedes flow via the low-head injection lines).

   a. Opens valves 888A and 888B to provide a flow path from the recirculation pump discharge to the high-head safety injection pump suction.

   b. Activates the low-pressure alarm circuit off of PT-947. If for some reason PT-947 alarm is not activated by this switch (RS-6), the operator can switch "HI

HEAD PUMP LO SUCTION PRESS ALARM" to activate this alarm. This latter switch is on the safeguards panel.

c. Closes valves 842 and 843 (high-head pump test line) (if their control feed interlock switches were first placed to the "OFF" position), and sends a close signal to valves 746 and 747 (residual heat removal heat exchanger discharge).

8. Closes switch seven (removes the two running safety injection pumps from service since they are no longer needed).

9. Closes switch eight (completes the isolation of the safety injection system and containment spray system lines from the refueling water storage tank).

a. Closes the valve on the spray test line.

b. Sends a close signal to the valve in the safety injection pumps suction line from the refueling water storage tank, which is administratively reenergized later in the sequence. (Control power for this valve is deenergized in accordance with Technical Specifications requirements).

10. Close switch five (Establishes additional cooling capability if adequate power is available i.e. all diesel breakers are either open or racked out, or at least one breaker from each of the three diesels is racked in and closed).

a. Starts second service water pump, non-essential header (the third pump is given the start signal if the second pump fails to start).

b. If (a) completed, starts second component cooling pump (the third pump is given the start signal if the second pump fails to start).

c. If (b) completed, starts recirculation pump 22 (unless already running). *[**Note**: running two (2) recirculation pumps is restricted to Low Head Recirculation. If High Head Recirculation is required, operator action is taken to prevent two recirculation pumps from operating simultaneously.]*

Although the listed switches are manual, each automatically causes the operations listed. An indicating lamp is provided to show the operator when the operations of a given switch have been performed. In addition, lamps indicating completion of the individual functions for a given switch are provided. These lamps are adjacent to the switches. Should an individual component fail to respond, the operator can take corrective action to secure appropriate response from controls within the control room.

Remote-operated valves for the injection phase of the safety injection system (Plant Drawings 9321-2735 and 235296 [Formerly Figure 6.2-1]), which are under manual control (i.e., valves, which normally are in their ready position and do not receive a safety injection signal) have their positions indicated on a common portion of the control board. At any time during operation when one of these valves is not in the ready position for injection, it is shown visually on the board. Table 6.2-2 lists the instrumentation readouts on the control board and assessment panel, which the operator can monitor during recirculation. In addition, an audible annunciation alerts the operator to the condition.

6.2.2.1.5 Location of the Major Components Required for Recirculation

The residual heat removal pumps are located in the residual heat removal pump room, which is on the basement floor of the primary auxiliary building (elevation 15-ft). The residual heat exchangers are on a platform above the basement floor in the containment building (elevation 66-ft).

The recirculation pumps are directly above the recirculation sump in the containment building (elevation 46-ft).

The component cooling pumps and heat exchangers are located in the primary auxiliary building (elevations 68-ft and 80-ft, respectively).

The auxiliary component cooling water pumps are located in the Mezzanine area of the piping penetration area at elevation 68-ft.

The service water pumps are located at the river water intake structure, and the redundant piping to the component cooling heat exchangers is run underground, until it surfaces just prior to its penetrating the Primary Auxiliary Building exterior wall.

6.2.2.2 Steam Line Break Protection

A large break of a steam system pipe rapidly cools the reactor coolant causing insertion of reactivity into the core and the depressurization of the system. Compensation is provided by the injection of boric acid from the refueling water storage tank. The analysis of the steam line rupture accident is presented in section 14.2.5.

6.2.2.3 Components

All associated components, piping, structures, and power supplies of the safety injection system are designed to seismic Class I criteria.

All components inside the containment are capable of withstanding or are protected from differential pressure that may occur during the rapid pressure rise to 47 psig in 10 sec.

Electrical equipment that has been determined to be important to safety and located in potentially harsh environments are environmentally qualified to ensure performance of their safety function under postaccident temperature, pressure, and humidity conditions.

Emergency core cooling components are either austenitic or an equivalent corrosion-resistant stainless steel, and hence, are compatible with the spray solution over the full range of exposure in the postaccident regime. Corrosion tests performed with simulated spray indicated negligible attack, both generally and locally, in stressed and unstressed stainless steel at containment and emergency core cooling system conditions. These tests are discussed in Reference 1.

The quality standards of all safety injection system components are given in summary form in Table 6.2-3.

6.2.2.3.1 Accumulators

The accumulators are pressure vessels filled with borated water and pressurized with nitrogen gas. During normal plant operation, each of the four accumulators is isolated from the reactor coolant system by two check valves in series. Should the reactor coolant system pressure fall below the accumulator pressure, the check valves open and borated water is forced into the cold legs of the reactor coolant system. Mechanical operation of the swing-disk check valves is the only action required to open the injection path from the accumulators to the core via the cold leg.

The level of borated water in each accumulator tank is adjusted remotely as required during normal plant operations. Refueling water is added using the accumulator topping pump (or safety injection pump 22 or 23). Water level is reduced by draining to the reactor coolant drain tank or through the chemistry sampling panel. Samples of the solution in the tanks are taken at the sampling station for periodic checks of boron concentration. Pressure is adjusted by adding nitrogen as required.

The accumulators are passive engineered safety features since the gas forces injection and no external source of power or signal transmission is needed to obtain fast-acting, high-flow capability when injection is required. One accumulator is attached to each of the four cold legs of the reactor coolant system.

The design capacity of the accumulators is based on the assumption that flow from one of the accumulators spills onto the containment floor through the ruptured loop. The flow from the three remaining accumulators provides sufficient water to fill the volume outside of the core barrel below the nozzles, the bottom plenum, and approximately one-half the core.

To assure the independence of the accumulators from each other, operating procedures require that only one liquid fill valve and only one nitrogen stop valve can be open at a time when reactor temperature is equal to or greater than 350°F.

The accumulators are carbon steel, internally clad with stainless steel and designed to ASME Section III, Class C. Connections for remotely draining or filling the fluid space, during normal plant operation, are provided.

Redundant level and pressure indicators are provided with readouts on the control board. Each indicator is equipped with high- and low-level alarms.

The accumulator design parameters are given in Table 6.2-4.

6.2.2.3.2 Boron Injection Tank

The boron injection tank has been removed.

6.2.2.3.3 Refueling Water Storage Tank

In addition to its normal duty to supply borated water to the refueling canal for refueling operations, this tank provides borated water to the safety injection pumps, the residual heat removal pumps, and the containment spray pumps for the Loss-of-Coolant Accident. These systems are kept sufficiently filled with water to ensure the systems remain operable and perform properly. During plant operation, this tank is aligned to these pumps.

The capacity of the refueling water storage tank is based on the requirement for filling the refueling canal; a minimum of 345,000 gal is required by the Technical Specifications to be maintained in the refueling water storage tank. This capacity provides an amount of borated water to assure:

1. A sufficient volume of water on the floor to permit the initiation of recirculation (246,000 gal).

2. A volume water sufficient to allow time for completing the switchover to recirculation and securing High Head Injection and Containment Spray Flow from the RWST (60,000 gal).

3. A sufficient volume of water to allow for instrument inaccuracies, additional margin, and for water that is physically unavailable from the bottom of the tank (39,000 gal).

4. The RWST water volume injected into containment, when added to accumulator discharge to the reactor coolant system, assures no return to criticality with the reactor at cold shutdown and no control rods inserted into the core.

The water in the tank is borated to ensure a minimum shutdown margin as discussed in Section 14.1.5.2.1. The maximum boric acid concentration is approximately 1.4 wt percent boric acid. At 32^{o}F, the solubility limit of boric acid is 2.2-percent. Therefore, the concentration of boric acid in the refueling water storage tank is well below the solubility limit at 32^{o}F. Steam heating is provided for the tank, and the outside lines are heat traced to maintain the temperature above freezing.

Each of two redundant channels of refueling water storage tank level instrumentation provide level indication and low-level alarms in the central control room. In addition, a third instrument provides local level indication.

The design parameters are presented in Table 6.2-6.

6.2.2.3.4 Pumps

The three high-head safety injection pumps for supplying borated water to the reactor coolant system are horizontal centrifugal pumps driven by electrical motors. Parts of the pump in contact with borated water are stainless steel or an equivalent corrosion-resistant material. Each safety injection pump is sized at 50-percent of the capacity required to meet the design criteria outlined in Section 6.2.1. The design parameters are presented in Table 6.2-7; Figure 6.2-6 gives the performance characteristics of these pumps.

A minimum flow bypass line is provided on each pump discharge to recirculate flow to the refueling water storage tank in the event the pumps are started with the normal flow paths blocked. Valves in the minimum flow bypass line (which are normally open) are equipped with motor operators. If either valve closes, an alarm annunciates in the control room. Power is de-energized to prevent spurious valve closure.

The safety injection pump bearing oil is cooled by CCW circulating water pumps using component cooling water as a heat sink. The CCW circulating water pumps are directly

connected to the injection pump motor shaft. The pump seals are designed to operate during the injection phase without forced component cooling water flow. During the recirculation phase, cooling water is supplied by the component cooling system or alternately from the primary water system. Emergency backup is available via connections to the city water system.
The two residual heat removal (low-head) pumps of the auxiliary coolant system are used to inject borated water at low pressure into the reactor coolant system. The two recirculation pumps are used to recirculate fluid from the recirculation sump back to the reactor, to the spray headers, or to suction of the safety injection pumps. The recirculation pumps will only be required to operate during the recirculation phase. In addition, the recirculation pumps are required to be operable for a period of one year. Per the 12/06/04 NRC generic SER on NEI 04-07 (Reference 5), evaluations of PWR post accident emergency recirculation performance to address the potential impact of debris blockage per GSI-191 (Reference 4) and Generic Letter 2004-02 (Reference 3) will use a mission time of 30 days. All four of these pumps are of the vertical centrifugal type driven by electric motors. The recirculation pumps are open suction, well-type pumps. Parts of the pumps, which contact the borated water solution during recirculation are stainless steel or an equivalent corrosion-resistant material. A minimum flow bypass line is provided on the discharge of the residual heat removal heat exchangers to recirculate cooled fluid to the suction of the residual heat removal pumps should these pumps be started with their normal flow paths blocked. There are two normally open motor-operated valves in this line. The control power to the two normally open motor operated valves is locked open. The emergency procedures ensure that the RHR pumps are not run in parallel for extended time periods with RCS pressure at or above their shutoff head. A minimum flow bypass, discharging back into the recirculation sump, is provided to protect the recirculation pumps should these flow paths be blocked. Valves in these lines are manually operated and are in the open position during normal plant operation. Figures 6.2-7 and 6.2-8 give the performance characteristics of these pumps. The design parameters are presented in Table 6.2-7.

The recirculation pump motors are air-to-water cooled in a similar manner as the containment cooling fan motors described in section 6.4.2.2.5, item 2. The motor fans are integral to the recirculation pump motor shafts. Cooling water to the motor heat exchanger is component cooling water. The sump water cools the pump bearings. The two auxiliary component cooling water pumps are started during the injection phase. However, their function during this phase is not required to protect the recirculation pump motors from the containment atmosphere. Since the recirculation pumps do not operate during injection, their motors do not experience any self-heating. Without this self-induced heat up, the motor’s functional capabilities and EQ characteristics have been shown in motor qualification testing to be unaffected by the post-LOCA environment. Even with an auxiliary component cooling water pump running, effectively no motor cooling occurs during the injection phase because the air circulating fans integral to the motor that drive cooling air through the heat exchanger and motor are not operating.

Details of the component cooling pumps and service water pumps, which serve the safety injection system, are presented in Section 9.3 and 9.6, respectively.

The pressure-retaining parts of the high-head safety injection pumps are castings conforming to ASTM A-296, Grade CA-15 or ASME SA-487, Grade CA-6NM. The pressure-retaining parts of the residual heat removal pumps and the recirculation pumps are castings conforming to ASTM A-351, Grade CF-8A (chromium content 21.0 to 22.5%) and ASTM A-351, Grade CF-8, respectively. Stainless steel forgings are procured per ASTM A-182, Grade F304 or F316, or ASTM A-336, Class F8 or F8M, and stainless steel plate is constructed to ASTM A-240, Type 304 or 316. All bolting material conforms to ASTM A-193. Material such as Monel is used at

points of close running clearances in the pumps to prevent galling and to ensure continued performance ability in high-velocity areas subject to erosion.

All pressure-retaining parts of the pumps were chemically and physically analyzed, and the results were checked to ensure conformance with the applicable ASTM specification. In addition, all pressure-retaining parts of the pump were liquid-penetrant inspected in accordance with Appendix VIII of Section VIII of the ASME Boiler and Pressure Vessel Code. The acceptance standard for the liquid-penetrant test is USAS B31.1, Code for Pressure Piping, Case N-10.

The pump design was reviewed with special attention to the reliability and maintenance aspects of the working components. Specific areas include the evaluation of the shaft seal and bearing design to determine whether adequate allowances had been made for shaft deflection and clearances between stationary parts.

Where welding of pressure-containing parts was necessary, a welding procedure including joint detail was submitted for review and approval by Westinghouse. The procedure included evidence of qualification necessary for compliance with Section IX of the ASME Boiler and Pressure Vessel Code Welding Qualifications. This requirement also applied to any repair welding performed on pressure-containing parts.

The pressure-containing parts of the pump were assembled and hydrostatically tested to 1.5 times the design pressure for 30 min.

Each pump was given a complete shop performance test in accordance with Hydraulic Institute Standards. The pumps were run at design flow and head, shutoff head, and three additional points to verify performance characteristics. Where net positive suction head is critical, this value was established at design flow by means of adjusting suction pressure.

An accumulator topping pump is provided to fill the accumulators rather than using safety injection pump 22 or 23. The pump is a double-diaphragm type with a capacity of approximately 5 gpm (293 gph). It is located in the safety injection system pump area and is operated from a local key-locked push button switch. The topping pump is capable of withstanding the safe shutdown earthquake but does not operate following safety injection actuation.

6.2.2.3.5 Heat Exchangers

The two residual heat exchangers of the auxiliary coolant system are sized for the cooldown of the reactor coolant system. Table 6.2-8 gives the design parameters of the heat exchangers. During the recirculation phase following a loss-of-coolant-accident, only one residual heat removal heat exchanger is required to ensure that heat removal requirements from the core and containment are met.

The ASME Boiler and Pressure Vessel Code has strict rules regarding the wall thicknesses of all pressure-containing parts, material quality assurance provisions, weld joint design, radiographic and liquid-penetrant examination of materials and joints, and hydrostatic testing of the unit as well as requiring final inspection and stamping of the vessel by an ASME Code inspector.

The designs of the heat exchangers also conform to the requirements of the Tubular Exchanger Manufacturers Association (TEMA) for Class R heat exchangers. Class R is the most rugged

class of TEMA heat exchangers and is intended for units where safety and durability are required under severe service conditions. Items such as tube spacing, flange design, nozzle location, baffle thickness and spacing, and impingement plate requirements are set forth by TEMA standards.

In addition to the above, additional design and inspection requirements were imposed to ensure rugged, high-quality heat exchangers such as the following:

1. Confined-type gaskets, general construction and mounting brackets suitable for the plant seismic design requirements.

2. Tubes and tube sheet capable of withstanding full shell-side pressure and temperature with atmospheric pressure on the tube side, ultrasonic inspection in accordance with Paragraph N-324.3 of Section III of the ASME Code of all tubes before bending, penetrant inspection in accordance with Paragraph N-627 of Section III of the ASME Code of all welds and hot- or cold-formed parts.

3. A hydrostatic test duration of not less than 30 min, the witnessing of hydro and penetrant tests by a qualified inspector, a thorough final inspection of the unit for good workmanship and the absence of any gouge marks or other scars that could act as stress concentration points, and a review of the radiographs and of the certified chemical and physical test reports for all materials used in the unit.

The residual heat exchangers are conventional vertical shell and U-tube type units. The tubes are seal welded to the tube sheet. The shell connections are flanged to facilitate shell removal for inspection and cleaning of the tube bundle. Each unit has an SA-285, Grade C, carbon steel shell; an SA-234 carbon steel shell end cap; SA-213, Type 304, stainless steel tubes; an SA-240, Type 304, stainless steel channel; an SA-240, Type 304, stainless steel channel cover; and an SA-240, Type 304, stainless steel tube sheet.

6.2.2.3.6 Valves

All parts of valves used in the safety injection system in contact with borated water are austenitic stainless steel or an equivalent corrosion-resistant material. The motor operators on the injection line isolation valves are capable of rapid operation. All valves required for the initiation of safety injection or isolation of the system have remote position indication in the control room.

Valving is specified for exceptional tightness. All valves, except those which perform a control function, are provided with backseats that are capable of limiting leakage. The estimated leakage of backseated valves outside containment is provided in Table 6.2-9. Those valves, which are normally open are backseated, except when operational considerations do not allow. [***Note** - The following valves may not be backseated based on operational requirements: 744, 850A, 850B, 851A, 851B, 883, 885A, 885B, 887A, 887B, 888A, 888B and 958.*] Normally closed globe valves are installed with recirculation flow under the seat to prevent the leakage of recirculated water through the valve stem packing. Relief valves are totally enclosed. Control and motor-operated valves, 2.5-in. and above, which are exposed to recirculation flow, are generally provided with double-packed stuffing boxes and stem leakoff-connections that are piped to the waste disposal system.

The check valves that isolate the safety injection system from the reactor coolant system are installed immediately adjacent to the reactor coolant piping to reduce the probability of a safety injection line rupture causing a loss-of-coolant accident.

A relief valve is installed in the safety injection pump discharge header discharging to the pressurizer relief tank to prevent overpressure in the lines that have a lower design pressure than the reactor coolant system. RV-855 is a thermal relief valve which protects the Safety Injection System piping and components from overpressurization due to thermal expansion of fluid in the system or from in-leakage of reactor coolant. The setpoint of RV-855 was changed to 1670 psig to ensure that the valve does not lift when operating the SI system at a pressure near the shutoff head of the SI pumps.

The gas relief valves on the accumulators protect them from pressures in excess of the design value.

6.2.2.3.7 Motor-Operated Butterfly Valve (Containment Sump Valve)

The pressure-containing parts (body, disks) of the valves employed in the safety injection system are designed per criteria established by USAS B16.5 or MSS SP-67 specifications. The materials of construction for these parts are procured per ASTM A182, F316 or A351, Grade CF8M or CF8. All material in contact with the primary fluid, except the packing and the liner, is austenitic stainless steel or an equivalent corrosion-resistant material. The liner is EPT-NORDEL (Du Pont). The pressure-containing cast components are radiographically inspected as outlined in ASTM E-71, Class 1 or Class 2. The body and disk are liquid-penetrant inspected in accordance with ASME Boiler and Pressure Vessel Code, Section VIII, Appendix VIII. The liquid-penetrant acceptable standard is as outlined in USAS B31.1, Case N-10.

The entire assembled unit is hydrotested as outlined in MSS SP-67, with the exception that the test is maintained for a minimum period of 30 min. The motor operator is evaluated in accordance with the GL 89-10 Motor Operated Valve Program to assure its capability to meet the required stem torque for opening and closing.

The shaft material is ASTM A276, Type 316, condition B, or precipitation hardened 17-4 pH stainless steel procured and heat treated to Westinghouse specifications. These materials are selected because of their corrosion-resistant, high-tensile properties, and their resistance to surface scoring by the packing.

The motor operator is located above the maximum sump fluid level and therefore is never submerged. The motor operator is extremely rugged and is noted throughout the power industry for its reliability. The unit incorporates a hammer-blow feature that allows the motor to impact the disks away from the fore or backseat upon opening or closing. This hammer-blow feature not only impacts the disk but allows the motor to attain its operational speed.

The valve is assembled, hydrostatically tested, seat-leakage tested (fore and back), operationally tested, cleaned, and packaged per specifications. All manufacturing procedures employed by the valve supplier such as welding, repair welding, and testing are submitted to Westinghouse for approval.

The valve operator completes its cycle from one position to the other in approximately 120 sec.

Valves that must function against system pressure are designed such that they function with a pressure drop equal to full system pressure across the valve disk.

6.2.2.3.8 Motor-Operated Gate Valves

The pressure-containing parts (body, bonnet, and disks) of the valves employed in the safety injection system are designed per criteria established by USAS B16.5 or MSS SP-66 specifications. The materials of construction for these parts are procured per ASTM A182, F316 or A351, Grade CF8M or CF8. All material in contact with the primary fluid, except the packing, is austenitic stainless steel or an equivalent corrosion-resistant material. The pressure-containing cast components are radiographically inspected as outlined in ASTM E-71, Class 1 or Class 2. The body, bonnet, and disks are liquid-penetrant inspected in accordance with ASME Boiler and Pressure Vessel Code, Section VIII, Appendix VIII. The liquid-penetrant acceptable standard is as outlined in USAS B31.1, Case N-10.

When a gasket is employed, the body-to-bonnet joint is designed per ASME Boiler and Pressure Vessel Code, Section VIII, or USAS B16.5 with a fully trapped, controlled compression, spiral wound gasket with provisions for seal welding, or of the pressure seal design with provisions for seal welding. The body-to-bonnet bolting and nut materials are procured per ASTM A193 and A194, respectively.

The entire assembled unit is hydrotested as outlined in MSS SP-61, with the exception that the test is maintained for a minimum period of 30 min. The seating design is of the Darling parallel disk design, the Crane flexible wedge design, or the equivalent. These designs have the feature of releasing the mechanical holding force during the first increment of travel. Thus, the motor operator is evaluated in accordance with the GL 89-10 Motor Operated Valve Program to assure its capability to meet the required stem thrust for opening and closing. The disks are guided throughout the full disk travel to prevent chattering and to provide ease of gate movement. The seating surfaces are hard faced (Stellite No. 6 or equivalent) to prevent galling and reduce wear.

The stem material is ASTM A276, Type 316, condition B, or precipitation hardened 17-4 pH stainless steel procured and heat treated to Westinghouse specifications. These materials are selected because of their corrosion-resistant, high-tensile properties, and their resistance to surface scoring by the packing. The valve stuffing box was originally designed with a lantern ring leakoff connection with a minimum of a full set of packing below the lantern ring and a maximum of one-half of a set of packing above the lantern ring; a full set of packing is defined as a depth of packing equal to 1-1/2 times the stem diameter. An alternate packing arrangement may be installed in these valves upon approval for substitution. Experience with designs utilizing live load and graphite packing has been favorable.

The motor operator is extremely rugged and is noted throughout the power industry for its reliability. The unit incorporates a hammer-blow feature that allows the motor to impact the disks away from the fore or backseat upon opening or closing. This hammer-blow feature not only impacts the disk but allows the motor to attain its operational speed.

The valve is assembled, hydrostatically tested, seat-leakage tested (fore and back), operationally tested, cleaned, and packaged per specifications. All manufacturing procedures employed by the valve supplier such as hard facing, welding, repair welding, and testing are submitted to Westinghouse for approval.

For those valves that must function on the safety injection signal, approximately 10-sec operation is required. For all other valves in the system, the valve operator completes its cycle from one position to the other in approximately 120 sec. Operating times greater than these values are permitted on a case by case basis if properly justified by an individual safety evaluation.

Valves that must function against system pressure are designed such that they function with a pressure drop equal to full system pressure across the valve disk.

6.2.2.3.9 Manual Valves

The stainless steel manual globe, gate, and check valves are designed and built in accordance with the requirements outlined in the motor-operated valve description above with the following exceptions:

1. Alternate materials, evaluated to be equivalent, have been used in some replacement valves.

2. Liquid-penetrant inspection of the body, bonnet, and disks to ASME V Article 6 with acceptance per ASME III has been used on some replacement valves.

The carbon steel valves are built to conform with USAS B16.5. The materials of construction of the body, bonnet, and disk conform to the requirements of ASTM A105, Grade II; A181, Grade II; or A216, Grade WCB or WCC. Alternate materials, evaluated to be equivalent, have been used in some replacement valves. The carbon steel valves pass only nonradioactive fluids and are subjected to hydrostatic test as outlined in MSS SP-61, except that the test pressure is maintained for at least 30 min/in. of wall thickness. Since the fluid controlled by the carbon steel valves is not radioactive, the double packing and seal weld provisions are not provided.

6.2.2.3.10 Accumulator Check Valves

The pressure-containing parts of this valve assembly are designed in accordance with MSS SP-66. All parts in contact with the operating fluid are of austenitic stainless steel or of equivalent corrosion-resistant materials procured to applicable ASTM or WAPD specifications. The cast pressure-containing parts are radiographed in accordance with ASTM E-94 and the acceptance standard as outlined in ASTM E-71, Class 1 or Class 2. The cast pressure-containing parts, machined surfaces finished hard facings, and gasket bearing surfaces are liquid-penetrant inspected per the ASME Boiler and Pressure Vessel Code, Section VIII, and the acceptance standard is as outlined in USAS B31.1, Code Case N-10. The final valve is hydrotested per MSS SP-61, except that the test pressure is maintained for at least 30 min. The seat leakage is conducted in accordance with the manner prescribed in MSS SP-61, except that the acceptable leakage is 2 cm^3/hr-in. nominal pipe diameter.

The valve is designed with a low-pressure drop configuration with all operating parts contained within the body, which eliminates those problems associated with packing glands exposed to boric acid. The clapper arm shaft is manufactured from 17-4 pH stainless steel heat treated to Westinghouse specifications. The clapper arm shaft bushings are manufactured from Stellite No. 6 material. The various working parts are selected for their corrosion-resistant, tensile, and bearing properties.

The disk and seat rings are manufactured from forgings. The mating surfaces are hard faced with Stellite No. 6 to improve the valve seating life. The flexible disc-hinge connection permits the disc to completely contact the seat even if there is minor seat movement.

The valves are intended to be operated in the closed position with a normal differential pressure across the disk of approximately 1650 psi. The valves remain in the closed position except for testing and safety injection. Since the valve will normally not be required to operate in the open condition, hence be subjected to impact loads caused by sudden flow reversal, it is expected that this equipment will not have difficulties performing its required functions.

When the valve is required to function, a differential pressure of less than 25 psig will shear any particles that may attempt to prevent the valve from functioning. Although the working parts are exposed to the boric acid solution contained within the reactor coolant loop, a boric acid "freeze up" is not expected with this low a concentration.

The experience derived from the check valves employed in the safety injection system of the Carolina-Virginia Tube Reactor (CVTR) in a similar system has indicated that the system is reliable and workable. The CVTR emergency injection system, maintained at atmospheric conditions, was separated from the main coolant piping by one 6-in. check valve. A leak detection pit was provided in the CVTR to accumulate any leakage coming back through the check valve. A level alarm provided a signal on excessive leakage. There was a gas volume in the upper space of the loop. The pressure differential was 1500 psi and the system was stagnant. The valve was located 2 to 3-ft from the main coolant piping, which resulted in some heatup and cooldown cycling. The CVTR went critical late in 1963. Since that time and up to initial operation of Indian Point Unit 2, the level alarm in the detection pit had never gone off due to check valve leakage.

6.2.2.3.11 Relief Valves

The accumulator relief valves are sized to pass nitrogen gas at a rate in excess of the accumulator gas fill line delivery rate. The relief valves will also pass water in excess of the maximum expected leak rate of 1.0 gpm identified in Technical Specifications with leakage being from the reactor coolant system into an accumulator through an accumulator discharge line. The accumulators are provided with level and pressure alarms. Operator response to inleakage causing these alarms to actuate would preclude the need for the relief valves to perform in a water relief capacity.

The safety injection test line relief valve is provided to relieve any overpressure, that might build up in the high-head safety injection piping due to thermal expansion of fluid in the system or from leakage from the reactor coolant system past the SI header check valves. The valve will pass a nominal 15 gpm (2.25 x 10^5cm^3/hr), which is far in excess of the manufacturing design in leakage rate from the reactor coolant system of 24 cm^3/hr.

6.2.2.3.12 Leakage Limitations of Valves

Valving is specified for exceptional tightness.

Normally open valves have backseats that limit leakage as shown in Table 6.2-9. Normally closed globe valves are installed with recirculation flow under the seat to prevent stem leakage from the more radioactive fluid side of the seat.

Motor-operated valves, which are exposed to recirculation flow, are generally provided with double-packed stuffing boxes and stem leakoff connections that are piped to the waste disposal system.

The specified leakage across the valve disk required to meet the equipment specification and hydrotest requirements is as follows:

1. Conventional globe - 3 cm^3/hr-in. of nominal pipe size.

2. Gate valves - 3 cm^3/hr-in. of nominal pipe size; 10 cm^3/hr-in. for 300- and 150-lb USA Standard.

3. Motor-operated gate valves - 3 cm^3/hr-in. of nominal pipe sizes; 10 cm^3/hr-in. for 300- and 150-lb USA Standard.

4. Check valves - 3 cm^3/hr-in. of nominal pipe size; 10 cm^3/hr-in. for 300- and 150-lb USA Standard.

5. Accumulator check valves - 2 cm^3/hr-in. of nominal pipe size; relief valves are totally enclosed.

Leakage from components of the recirculation loop including valves, is given in Table 6.2-9.

6.2.2.3.13 Piping

All safety injection system piping in contact with borated water is austenitic stainless steel. Piping joints are welded except for the flanged connections at the safety injection pumps, the recirculation pumps, and valve 741A.

The piping beyond the accumulator stop valves is designed for reactor coolant system conditions (2485 psig, 650^{o}F). All other piping connected to the accumulator tanks is designed for 700 psig and 400^{o}F.

The safety injection pump and residual heat removal pump suction piping (210 psig at 300^{o}F) from the refueling water storage meets net positive suction head requirements of the pumps.

The safety injection high-pressure branch lines (1500 psig at 300^{o}F) are designed for high-pressure losses to limit the flow rate out of the branch line, which may have ruptured at the connection to the reactor coolant loop. The system design incorporates the ability to isolate the safety injection pumps on separate headers such that full flow from at least one pump is ensured should a branch line break.

The piping is designed to meet the minimum requirements set forth in (1) the USAS B31.1 Code (1955) for the Pressure Piping, (2) Nuclear Code Case N-7, (3) USAS Standards B36.10 and B36.19, and (4) ASTM Standards with supplementary standards plus additional quality control measures.

Minimum wall thicknesses are determined by the USAS Code (1955) formula found in the power piping Section 1 of the USAS Code (1955) for Pressure Piping. This minimum thickness has been increased to account for the manufacturer's permissible tolerance of –12.5-percent on the

nominal wall. Purchased pipe and fittings have a specified nominal wall thickness that is no less than the sum of that required for pressure containment, mechanical strength, and manufacturing tolerance.

Thermal and seismic piping flexibility analyses have been performed. Special attention is directed to the piping configuration at the pumps with the objective of minimizing pipe imposed loads at the suction and discharge nozzles. Piping is supported to accommodate expansion due to temperature changes during the accident.

Piping between valves 730 and 731 (Line 10) has 6" thick insulation to assure operability of these valves during design basis accident conditions.

Pipe and fittings materials are procured in conformance with all requirements of ASTM and USAS specifications. All materials are verified for conformance to specification and documented by certification of compliance to ASTM material requirements. Specifications impose additional quality control upon the suppliers of pipes and fittings as listed below.

1. Check analyses are performed on both the purchased pipe and fittings.

2. Pipe branch lines 2.5-in. and larger between the reactor coolant pipes and the isolation stop valves conform to ASTM A376 and meet the supplementary requirement S6 for ultrasonic testing. Fittings conform to the requirements of ASTM A403. Fittings 2.5-in. and above have requirements for ultrasonic testing inspection similar to S6 of A376.

Shop fabrication of piping subassemblies is performed by reputable suppliers in accordance with specifications that define and govern material procurement, detailed design, shop fabrication, cleaning, inspection, identification, packaging, and shipment.

Welds for pipes sized 2.5-in. and larger are butt welded. Reducing tees are used where the branch size exceeds one-half of the header size. Branch connections of sizes that are equal to or less than one-half of the header size are of a design that conforms to the USAS rules for reinforcement set forth in the USAS B31.1 Code for Pressure Piping. Bosses for branch connections are attached to the header by means of full penetration welds.

All welding is performed by welders and welding procedures qualified in accordance with the ASME Boiler and Pressure Vessel Code, Section IX, Welding Qualifications. The shop fabricator is required to submit all welding procedures and evidence of qualifications for review and approval prior to release for fabrication. All welding materials used by the shop fabricator must have prior approval.

All high-pressure piping butt welds containing radioactive fluid at greater than 600°F temperature and 600 psig pressure or equivalent are radiographed. The remaining piping butt welds are randomly radiographed. The technique and acceptance standards are those outlined in UW-51 of the ASME Boiler and Pressure Vessel Code, Section VIII. In addition butt welds are liquid-penetrant examined in accordance with the procedure of ASME Boiler and Pressure Vessel Code, Section VIII, Appendix VIII, and the acceptance standard as defined in the USAS Nuclear Code Case N-10. Finished branch welds are liquid-penetrant examined on the outside, and where size permits, on the inside root surfaces.

A postbending solution anneal heat treatment is performed on hot-formed stainless steel pipe bends. Completed bends are then completely cleaned of oxidation from all affected surfaces. The shop fabricator is required to submit the bending, heat treatment, and cleanup procedures for review and approval-prior to release for fabrication.

General cleaning of completed piping subassemblies (inside and outside surfaces) is governed by basic ground rules set forth in the specifications. For example, these specifications prohibit the use of hydrochloric acid and limit the chloride content of service water and demineralized water.

The packaging of the piping subassemblies for shipment is done so as to preclude damage during transit and storage. Openings are closed and sealed with tight fitting covers to prevent the entry of moisture and foreign material. Flange facings and weld end preparations are protected from damage by means of wooden cover plates and securely fastened in position. The packing arrangement proposed by the shop fabricator is subject to approval.

6.2.2.3.14 Pump and Valve Motors Outside Containment

Motor electrical insulation systems are supplied in accordance with IEEE, and NEMA standards and are tested as required by standards.

Temperature rise design selection is such that normal long life is achieved even under accident loading conditions.

Criteria for motors of the safety injection system require that under any anticipated mode of operation the motor nameplate rating is not exceeded. The pump motors have a 1.15 service factor for normal operation. Design and test criteria ensure that motor loading does not exceed the application criteria.

6.2.2.3.15 Pump and Valve Motors Inside Containment

Motors for the recirculation pumps were originally specified to operate in an ambient condition of saturated steam of 270°F and 47 psig pressure for 1 day, followed by indefinite operation at 155°F and 5 psig in a steam atmosphere. These ambient conditions and operating times have been updated and are maintained by the ongoing Environmental Qualification Program discussed in Section 7.1.4. As part of this program, the recirculation pump motors are qualified to withstand containment environmental conditions following the loss of coolant accident so that the pumps can perform their required function during the recovery period (one year). These motors are of a similar design as the containment fan cooler motors. Refer to Section 6.4.2.2.5 for a description and evaluation of the motor design. Per the 12/06/04 NRC generic SER on NEI 04-07 (Reference 5), evaluations of PWR post accident emergency recirculation performance to address the potential impact of debris blockage per GSI-191 (Reference 4) and Generic Letter 2004-02 (Reference 3) will use a mission time of 30 days.

The motors for the valves inside containment are designed to withstand containment environment conditions following the loss-of-coolant accident so that the valves can perform the required function during the recovery period.

Periodic operation of the motors and tests of the insulation ensure that the motors remain in a reliable operating condition.

Although the motors that are provided only to drive engineered safety features equipment are normally run only for tests, the design loading and temperature rise limits are based on accident conditions. Normal design margins are specified for these motors to make sure the expected lifetime includes allowance for the occurrence of accident conditions.

6.2.2.3.16 Valve Motor Operators

Environmental Qualification

As part of the original plant design, a program of environmental qualifications was performed on valve motor operators important to plant safety. Tests to demonstrate the adequacy of valve motor operators to be functional after exposure to temperature, pressure, and radiation were conducted in two groups.

The first group test was the exposure of valve motor operators to both temperature and pressure. Two suppliers, Philadelphia Gear Corporation Limitorque Division and Crane Company Teledyne Division, conducted simulated containment pressure and temperature tests as follows with pressure and temperature similar to that predicted for the incident:

1. Operator located inside a pressure vessel with the operator exposed to approximately 330°F at 90 psig.

2. Operator cycled approximately 3 times under simulated valve operating loads.

3. Pressures and temperatures reduced in step change to 285°F at 60 psig, 219°F at 20 psig, and 152°F at atmosphere or less.

4. Operator cycled approximately 3 times at each of the levels of change. Full recordings of pertinent data were taken throughout the tests.

5. Unit was examined after completion of test and operating data compared to data prior to exposure.

The second group test was the radiation test on a motor from the valve operator.

1. Two production line motors were used for this test; one exposed to 1.5×10^8 rads of gamma radiation for an approximate period of 1 month, the other motor used for the final comparative analysis.

2. Both units were tested for coil resistance, insulation meggering both before and after motor vibration, and reversing operations.

More recently, a program of environmental requalifications of items important to plant safety has been initiated using the "Division of Operating Reactors" or NUREG-0588 guidelines. See Section 7.1.4 for a discussion of this ongoing program.

In response to IE Information Notice 86-03, all limitorque motor operators on the EQ Master List (see Section 7.1.4) were inspected and serviced to assure that wiring, limit switches and torque switches have been environmentally qualified.

In response to the IE Bulletin 85-03, the operability of key safety Motor Operated Valves was verified with associated full differential pressure.

6.2.2.4 Electrical Supply

Details of the normal and emergency power sources for the safety injection system are presented in Chapter 8.

6.2.2.5 Protection Against Dynamic Effects

The injection lines penetrate the containment adjacent to the primary auxiliary building. For most of the routing, these lines are outside the crane wall, and hence are protected from missiles originating within these areas. Each line penetrates the crane wall near the injection point to the reactor coolant pipe. In this manner, maximum separation and hence protection it provided in the coolant loop area.

Coolant loop supports are designed to restrict the motion in one loop due to rupture in another loop to about one-tenth of an inch, whereas the attached safety injection piping can sustain a 3-in. displacement without exceeding the working stress range. The analysis assumes that the injection flow to the ruptured loop is spilled on the containment floor.

In 1989, the NRC approved changes to the design basis with respect to dynamic effects of postulated primary loop ruptures, as discussed in Section 4.1.2.4

All hangers and anchors are designed in accordance with USAS B31.1, Code for Pressure Piping. This code provides minimum requirements on materials, design, and fabrication with ample safety margins for both dead and dynamic loads over the life of the equipment. Concrete missile barriers, bumpers, walls, and other concrete structures are designed in accordance with ACI 318-63, Building Code Requirements for Reinforced Concrete. Specifically, these standards require the following:

1. All materials used are in accordance with ASTM specifications that establish quality levels for the manufacturing process, minimum strength properties, and for test requirements that ensure compliance with the specifications.

2. Welding processes and welders must be qualified for each class of material welded and for types and positions of welds.

3. Maximum allowable stress values are established, which provide an ample safety margin on both yield strength and ultimate strength.

6.2.3 Design Evaluation

6.2.3.1 Range of Core Protection

The measure of effectiveness of the safety injection system is the ability of the pumps and accumulators to keep the core flooded or to reflood the core rapidly when the core has been uncovered for postulated large area ruptures. The result of this performance is to limit sufficiently any increase in clad temperature below a value where emergency core cooling objectives are met (Section 6.2.1). The sequence of events involving safety injection actuation for small and large breaks of a reactor coolant pipe are presented in Section 14.3.2.

6.2.3.2 System Response

To provide protection for large area ruptures in the reactor coolant system, the safety injection system must respond rapidly to reflood the core following the depressurization and core voiding that is characteristic of large area ruptures. The accumulators act to perform the rapid reflooding function with no dependence on the normal or emergency power sources and also with no dependence on the receipt of an actuation signal.

The operation of this system with three of the four available accumulators delivering their contents to the reactor vessel (one accumulator spilling through the break) prevents fuel clad melting and limits metal-water reaction to an insignificant amount (<1-percent).

The function of the safety injection and residual heat removal pumps is to complete the refill of the vessel and ultimately return the core to a subcooled state. Moreover, there is sufficient excess water delivered by the accumulators to tolerate a delay in starting the pumps.

Initial response of the injection systems is automatic, with appropriate allowances for delays in the actuation of circuitry and active components. The active portions of the injection systems are automatically actuated by the safety injection signal (Chapter 7). In addition, manual actuation of the entire injection system and individual components can be accomplished from the control room. In the analysis of system performance, delays in reaching the programmed trip points and in the actuation of components are conservatively established on the basis that only emergency onsite power is available.

The starting sequence of the safety injection and residual heat removal pumps and the related emergency power equipment is discussed in sections 7.2 and 8.2.3.4 and their analyzed performance is discussed in the various Chapter 14 safety analyses.

6.2.3.3 Single-Failure Analysis

A single active failure analysis is presented in Table 6.2-10. All credible active system failures are considered. This analysis is based on the worst single failure (generally a pump failure) in both the safety injection and residual heat removal pumping systems. The analysis shows that the failure of any single active component will not prevent fulfilling the design function. The analysis of the loss-of-coolant accident presented in Section 14.3 is consistent with this single-failure analysis.

In addition to active failures, an alternative flow path is available to maintain core cooling if any part of the recirculation flow path becomes unavailable due to a single passive failure. This is evaluated in Table 6.2-11. The procedure followed to establish the alternative flow path also isolates the spilling line. A valve is provided in the containment recirculation line to the residual heat removal pumps to isolate this line should it be required.

Therefore, the ECCS design incorporates redundancy of components such that neither a single active component failure during the injection phase nor an active or passive failure during the recirculation phase will degrade the ECCS function. Only active failures are assumed to occur within the first 24 hours following the initiating event.

Failure analyses of the component cooling and service water system under loss-of-coolant accident conditions are described in Sections 9.3 and 9.6, respectively.

6.2.3.4 Reliance on Interconnected Systems

During the injection phase, the high-head safety injection pumps do not depend on any portion of other systems, with the exception of the suction line from the refueling water storage tank and the component cooling loop as a heat sink for bearing and lube oil cooling. During the recirculation phase of the accident for small breaks, suction to the high-head safety injection pumps is provided by the recirculation pumps or, should backup capability be required, the residual heat removal pumps. The residual heat removal (low-head) pumps are normally used during reactor shutdown operations. Whenever the reactor is at power, the pumps are aligned for emergency duty.

6.2.3.5 Shared Function Evaluation

Table 6.2-12 is an evaluation of the main components, which have been previously discussed, and a brief description of how each component functions during normal operation and during the accident.

6.2.3.6 Passive Systems

The accumulators are a passive safety feature in that they perform their design function in the total absence of an actuation signal or power source. The only moving parts in the accumulator injection train are in the two check valves.

The working parts of the check valves are exposed to fluid of relatively low boric acid concentration. Even if some unforeseen deposition accumulated, a reversed differential pressure of about 25 psi can shear any particles in the bearing that may tend to prevent valve functioning. This is demonstrated by calculation.

The isolation valve at each accumulator is only closed when the reactor is intentionally depressurized or momentarily for testing when pressurized. The isolation valve is normally open and an alarm in the control room sounds if the valve is inadvertently closed. It is not expected that the isolation valve will have to be closed due to excessive leakage through the check valves.

The check valves operate in the closed position with a nominal differential pressure across the disk of approximately 1650 psi. They remain in this position except for testing or when called upon to function. Since the valves operate normally in the closed position and are therefore not subject to the abuse of flowing operation or impact loads caused by sudden flow reversal and seating, they do not experience any wear of the moving parts, and therefore function as required.

When the reactor coolant system is being pressurized during the normal plant heatup operation, the check valves can be tested for leakage as soon as there is about 150 psi differential across the valve. This test confirms the seating of the disk and whether or not there has been an increase in the leakage since the last test. When this test is completed, the discharge line test valves are opened and the reactor coolant system pressure increase is continued. There should be no increase in leakage from this point on since increasing reactor coolant pressure increases the seating force and decreases the probability of leakage.

The accumulators can accept leakage from the reactor coolant system without effect on their availability. Table 6.2-13 indicates what inleakage rates, over a given time period, require readjusting the level at the end of the time period. In addition, these rates are compared to the maximum allowed leak rates for manufacturing acceptance tests (20 cm^3/hr i.e., 2 cm^3/hr/in).

Inleakage at a rate of 5 cm^3/hr-in., 2.5 times test, would require that the accumulator water volume be adjusted approximately once every 30 months. This would indicate that level adjustments can be scheduled for normal refueling shutdowns and that this work can be done at the operator's convenience. At a leak rate of 30 cm^3/hr-in. (15 times the acceptance leak rate), the water level will have to be readjusted approximately once every 5 to 6 months. This readjustment will take about 2 hr maximum.

The accumulators are located inside the reactor containment and protected from the reactor coolant system piping and components by a missile barrier. Accidental release of the gas charge in the accumulator would cause an increase in the containment pressure. This release of gas has been included in the containment pressure analysis for the large break loss-of-coolant accidents, (Section 14.3.3.3 and 14.3.5.1.1).

During normal operation, the flow rate through the reactor coolant piping is approximately 5 times the maximum flow rate from the accumulator during injection. Therefore, fluid impingement on reactor vessel components during operation of the accumulator is not restricting.

6.2.3.7 Emergency Flow to the Core

Special attention is given to factors that could adversely affect the accumulator and safety injection flow to the core. These factors are as follows:

1. Steam binding in the core, including flow blockage due to loop sealing.
2. Loss of accumulator water during blowdown.
3. Short circuiting of the accumulator from the core to another part of the reactor coolant system.
4. Loss of accumulator water through the breaks.

All of the above are considered in the analysis of the Loss of Coolant Accident which is discussed in Section 14.3.

6.2.3.8 External Recirculation Loop Leakage

Table 6.2-9 summarizes the maximum potential leakage from the leak sources of the external recirculation loop, which goes through the residual heat removal pumps, a residual neat exchanger, and the high-head safety injection pumps. In the analysis, a maximum leakage is assumed from each leak source. For conservatism, 3 times the maximum expected leak rate from the pump seals was assumed, even though the seals are acceptance tested to essentially zero leakage, and a leakage of 10 drops/min was assumed from each flange although each flange would be adjusted to essentially zero leakage. The total maximum potential leakage resulting from all sources is 999 cm^3/hr to the auxiliary building atmosphere and 21 cm^3/hr to the drain tank.

During external recirculation, significant margin exists between the design and operating conditions of the residual heat removal system components, as shown in Table 6.2-14. In

addition, during normal plant cooldown, operation of the residual heat removal system is initiated when the primary system pressure and temperature have been reduced to below 365 psig (the upper limit to prevent RHR system overpressurization) and 350°F, respectively. Even assuming a conservative maximum RHR System pressure of 232 psig and a conservative maximum RHR System temperature of 277°F during recirculation as shown in Table 6.2-14, significant margin also exists between normal operating and accident conditions. In view of the above margins, it is considered that the leakage rates tabulated in Table 6.2-9 are conservative. The radiological consequences of external recirculation loop leakage following a design basis accident are presented in Section 14.3.6.6.

6.2.3.9 Pump Net Positive Suction Head Requirements

6.2.3.9.1 Residual Heat Removal Pumps

The net positive suction head (NPSH) of the residual heat removal pumps is evaluated for normal plant shutdown operation and the operation of both the injection and recirculation phases of the design-basis accident.

The residual heat removal pumps are used as backup to the internal recirculation pumps in the event of failures to the normal recirculation path; this duty provides the pumps with the minimum NPSH condition. For the design case of [Deleted] one pump recirculating through one heat exchanger path, the available NPSH exceeds the NPSH required, assuming saturated fluid and no operator action to throttle back the flow. There is no postulated failure that requires both RHR pumps to operate in the recirculation phase.

6.2.3.9.2 Safety Injection Pumps

The NPSH for the safety injection pumps is evaluated for both the injection and recirculation phase operation of the design-basis accident. The end of injection phase operation gives the limiting NPSH requirement; the NPSH available is determined from the elevation head and vapor pressure of the water in the refueling water storage tank and the pressure drop in the suction piping from the tank to the pumps. At the end of the injection phase, greater than 20-percent NPSH margin is available assuming all three pumps running together with two residual heat removal pumps at maximum flow conditions permitted by the system alignment.

6.2.3.9.3 Recirculation Pumps

The NPSH for the recirculation pumps is evaluated for recirculation operation. The NPSH available is determined considering the elevation head of the water above the pump NPSH reference line (eye of the 1st stage impeller) in the sump, level drawdown due to the flow path in the containment, fluid temperature adjustments, and strainer head losses. The NPSH determination met the requirements of GL 2004-02. The containment water level is confirmed to be above the minimum level required for NPSH, prior to starting the recirculation pumps during the changeover from the injection phase to the recirculation phase. The RWST level is confirmed to be less than 2 feet prior to stopping the remaining operating containment spray pump and establishing simultaneous recirculation flow to the core and the spray headers. This maximizes the available NPSH to the recirculation pumps in this mode of operation.

The internal recirculation pumps are conventional vertical condensate pumps and are of double suction design, requiring less NPSH. At the initiation of the recirculation phase, the NPSH requirement is met with one or both pumps operating at the pump design flow. When

simultaneous recirculation flow to the core and spray headers is established, the available NPSH requirement will be met at expected pump flows. [Deleted]

6.2.4 Minimum Operating Conditions

The Technical Specifications establish limiting conditions regarding the operability of the system when the reactor is critical.

6.2.5 Inspections and Tests

6.2.5.1 Inspection

All components of the safety injection system are inspected periodically to demonstrate system readiness.

The pressure-containing components are inspected for leaks from pump seals, valve packing, flanged joints, and safety valves during system testing.

Current requirements for safety injection system surveillance are discussed in Sections 3.5 and 5.5.6 of the facility Technical Specifications and in UFSAR Section 1.12, "Inservice Inspection and Testing Programs".

6.2.5.2 Preoperational Testing

6.2.5.2.1 Component Testing

Preoperational performance tests of the components were performed in the manufacturer's shop. The pressure-containing parts of the pump were hydrostatically tested in accordance with Paragraph UG-99 of Section VIII of the ASME Code. Each pump was given a complete shop performance test in accordance with Hydraulic Institute Standards. The pumps were run at design flow and head, shutoff head, and at additional points to verify performance characteristics. Net positive suction head was established at design by means of adjusting suction pressure for a representative pump. This test was witnessed by qualified Westinghouse personnel.

The remote-operated valves in the safety injection system are motor-operated. Shop tests for each valve included a hydrostatic pressure test, leakage tests, a check of opening and closing time, and verification of torque switch and limit switch settings. The ability of the motor operator to move the valve with the design differential pressure across the gate was demonstrated by opening the valve with an appropriate hydrostatic pressure on one side of the valve.

The recirculation piping and accumulators were initially hydrostatically tested at 150-percent of design pressure.

The service water and component cooling water pumps were tested prior to initial operation.

6.2.5.2.2 System Testing

An initial functional test of the core cooling portion of the safety injection system was conducted during the hot-functional testing of the reactor coolant system before initial plant startup. The purpose of the initial systems test was to demonstrate the proper functioning of instrumentation

and actuation circuits and to evaluate the dynamics of placing the system in operation. This test was performed following the flushing and hydrostatic testing of the system.

The functional test was performed with the water level below the safety injection setpoint in the pressurizer and with the reactor coolant system initially cold and at low pressure. The safety injection system valving was set initially to simulate the system alignment for plant power operation.

To initiate the test, the safety injection block switch was moved to the unblock position to provide control power allowing the automatic actuation of the safety injection relays from low-pressure signals from the pressurizer instrumentation. Simultaneously, the breakers supplying outside power to the 480-V buses were tripped manually and operation of the emergency diesel system automatically commenced. The high-head safety injection pumps and the residual heat removal pumps were started automatically following the prescribed diesel loading sequence. The valves were operated automatically to align the flow path for injection into the reactor coolant system.

The rising water level in the pressurizer provided indication of system delivery. Flow into the reactor coolant system terminated with the filling of the pressurizer, and the operation of the safety injection systems was terminated manually in the control room.

This functional test provided information to confirm valve operating times, pump motor starting times, the proper automatic sequencing of load addition to the emergency diesels, and delivery rates of injection water to the reactor coolant system.

The functional test was repeated for the various modes of operation needed to demonstrate performance at partial effectiveness, that is, to demonstrate the proper loading sequence with two of the three emergency diesels and to demonstrate the correct automatic starting of a second pump should the first pump fail to respond. These latter cases were performed without delivery of water to the reactor coolant system, but included starting of all pumping equipment involved in each test.

The systems were accepted only after the demonstration of proper actuation and after the demonstration of flow delivery and shutoff head within design requirements.

Flow was introduced into the reactor coolant loops through the accumulator discharge line to demonstrate the operability of the check valves and remotely actuated stop valve, and to confirm length to diameter (L/D) ratios of accumulator discharge lines used in the calculation.

6.2.5.3 Post-operational Testing

6.2.5.3.1 Component Testing

Routine periodic testing of the safety injection system components and all necessary support systems at power is done. No inflow to the reactor coolant system will occur whenever the reactor coolant pressure is above the safety injection pump shutoff head. If such testing indicates a need for corrective maintenance, the redundancy of equipment in these systems permits such maintenance to be performed without shutting down or reducing load under conditions defined in the Technical Specifications. These conditions include such matters as the period within which the component is to be restored to service and the capability of the remaining equipment to meet safety limits within such a period.

Test Circuits are provided to examine periodically the leakage back through the check valves and to ascertain that these valves seat whenever the reactor system pressure is increased. The recirculation pumps are normally in a dry sump. These pumps can only be started and allowed to reach full speed with the plant at cold shutdown. Flow testing of these pumps is performed during refueling operations by filling the recirculation sump and directing the flow back to the sump through the valve on the discharge of the pump. The service water and component cooling pumps not running during normal operation may be tested by alternating with the operating pumps.

The contents of the accumulators and the refueling water storage tank are sampled periodically to determine the boron concentration.

6.2.5.3.2 System Testing

System testing is conducted during plant shutdown to demonstrate proper automatic operation of the safety injection system. A test signal is applied to initiate automatic action and verification made that the safety injection and residual heat removal pumps receive start signals. The test demonstrates the operation of the valves, pump circuit breakers and automatic circuitry. Isolation valves in the injection line will be blocked closed so that flow is not introduced into the reactor coolant system. The test is considered satisfactory if control board indication and visual observations indicate all components have operated and sequenced properly.

The safety injection piping up to the final isolation valve is maintained sufficiently full of borated water at refueling water concentration while the plant is in operation to ensure the system remain operable and perform properly. The safety injection pumps recirculate refueling water through the injection lines via a small test line provided for this purpose.

Flow in each of the safety injection headers and in the main flow line for the residual heat removal pumps is monitored by a local flow indicator. Pressure instrumentation is also provided for the main flow paths of the safety injection and residual heat removal pumps. Accumulator isolation valves are blocked closed for this test.

The high pressure safety injection pumps are run and the variable orifices and injection line valves are adjusted to balance flowrates within the specified range.

The eight-switch sequence for recirculation operation is tested to demonstrate proper sequencing of valves and pumps. The recirculation pumps are blocked from starting during this test.

The external recirculation flow paths are hydrotested during periodic retests at the operating pressures. This is accomplished by running each pump, which could be used during external recirculation (safety injection and residual heat removal pumps) and checking the discharge and recirculation test lines. The suction lines are tested by running the residual heat removal pumps and opening the flow path to the safety injection pumps in the same manner as described above.

During the above test, all system joints, valve packings, pump seals, leakoff connections, or other potential points of leakage can be visually examined. Valve gland packing, pump seals, and flanges are adjusted or replaced as required to reduce the leakage to acceptable proportions. For power-operated valves, final packing adjustments are made, and the valves are

are put through an operating cycle before a final leakage examination is made.

The entire recirculation loop, except the recirculation line to the residual heat removal pumps, is pressurized during periodic testing of the engineered safety features components. The recirculation line to the residual heat removal pump is capable of being hydrotested during plant shutdown, and it is also leak-tested at the time of the periodic retests of the containment.

REFERENCES FOR SECTION 6.2

1. M. J. Bell, et al., Investigations of Chemical Additives for Reactor Containment Spray Systems, WCAP-7153, Westinghouse Electric Corporation, March 1968.

2. Deleted

3. NRC Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors”, dated September 13, 2004.

4. NRC Generic Safety Issue (GSI) 191, “Assessment of Debris Accumulation on PWR Sump Performance.”

5. NRC Generic SER dated 12/06/04 on the NEI 04-07 Guidance Report entitled, “Pressurized Water Reactor Sump Performance Evaluation Methodology.”

6. A. E. Lane, et al., Evaluation of Post Accident Chemical Effects in Containment Sump Fluids to Support GSI-191, WCAP-16530-NP-A, Westinghouse Electric Corporation LLC, March 2008.

7. T. S. Andreychek, et al., Evaluation of Downstream Sump Debris Effects in Support of GSI-191, WCAP-16406-P-A, Revision 1; Westinghouse Electric Corporation LLC, March 2008.

8. T. S. Andreychek, et al., Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fliud, WCAP-16793-NP, Revision 0; Westinghouse Electric Corporation LLC, May 2007.

TABLE 6.2-1
Safety Injection System – Code Requirements

Component	Code
Refueling Water Storage Tank	AWWA D100-65
Residual Heat Exchanger	
Tube Side	ASME Section III Class C

Shell Side	ASME Section VIII
Accumulators	ASME Section III Class C
Valves	USAS B16.5 (1955)
Piping	USAS B31.1 (1955)

TABLE 6.2-2 (Sheet 1 of 2)
Instrumentation Readouts On The Control Board
For Operator Monitoring During Recirculation

Valves

System	Valve No.
SIS	MOV 1802 A, B
SIS	MOV 885 A, B
SIS	MOV 889 A, B
SIS	MOV 888 A,B
SIS	MOV 866 A, B, C, D
SIS	MOV 851 A, B
SIS	MOV 856 A, B, C, D,E,F
SIS	MOV 882
SIS	MOV 842
SIS	MOV 843
ACS	MOV 744
ACS	MOV 745 A,B
ACS	MOV 746
ACS	MOV 747
ACS	MOV 1810
ACS	HCV 638
ACS	HCV 640

Instruments

System	Channel No
SIS	FI 945 A, B
SIS	FI 946 A, B, C,D
SIS	FI 924
SIS	FI 925
SIS	FI 926
SIS	FI 927
SIS	LI 938
SIS	LI 939
[Deleted]	
SIS	LI 941
SIS	LT 3300
SIS	LT 3301
SIS	LT 3302
[Deleted]	
SIS	LT 3304
SIS	PI 922
SIS	PI 923
SIS	PI 947
ACS	FI 640
ACS	LI 628

ACS	TR 636
RCS	LI 459

TABLE 6.2-2 (Sheet 2 of 2)
Instrumentation Readouts On The Control Board
For Operator Monitoring During Recirculation

Instruments (continued)

System	Channel No.
RCS	LI 460
RCS	LI 461
RCS	LI 462

Pumps

System	Pumps
SIS	Safety Injection
SWS	Service Water
ACS	Component Cooling
CS	Containment Spray
RS	Recirculation
ACS	Residual Heat Removal

Key:

ACS - Auxiliary Coolant System
CS - Containment Spray System
RCS - Reactor Coolant System
RS - Recirculation
SIS - Safety Injection System
SWS - Service Water System

TABLE 6.2-3 (Sheet 1 of 3)
Quality Standards Of Safety Injection System Components

Residual Heat Exchanger

A. Tests and inspections
   1. Hydrostatic test
   2. Radiograph of longitudinal and girth welds (tube side only)
   3. Ultrasonic testing of tubing or eddy current tests
   4. Dye penetrant test of welds
   5. Dye penetrant test of tube to tube sheet welds
   6. Gas leak test of tube to tube sheet welds before hydro and expanding of tubes

B. Special manufacturing process control
   1. Tube to tube sheet weld qualifications procedure
   2. Welding and NDT and procedure review
   3. Surveillance of supplier quality control and product

Component Cooling Heat Exchanger

A. Tests and inspections
   1. Hydrostatic Test
   2. Dye penetrant test of welds

B. Special Manufacturing Process Control
   1. Welding and NDT and procedure review
   2. Surveillance of supplier quality control and product

Safety Injection, Recirculation, and Residual Heat Removal Pumps

A. Test and inspections
   1 Performance test
   2 Dye penetrant of pressure retaining parts$_1$
   3. Hydrostatic test

B. Special manufacturing process control
   1. Weld, NDT, and inspection procedures for review
   2. Surveillance of suppliers quality control system and product

Accumulators

A. Tests and inspection
   1. Hydrostatic test
   2. Radiography of longitudinal and girth welds
   3. Dye penetrant/magnetic particle of weld

B. Special manufacturing process control
   1. Weld, fabrication, NDT, and inspection procedure review
   2. Surveillance of suppliers quality control and product

TABLE 6.2-3 (Sheet 2 of 3)
Quality Standards Of Safety Injection System Components

Valves

A. Tests and inspections
   1. 200 psi and 200°F or below (cast or bar stock)
      a. Dye penetrant test
      b. Hydrostatic test
      c. Seat leakage test
   2. Above 200 psi and 200°F
      a. Forged valves (2-1/2-in. and larger)
         (1) Ultrasonic tests of billet prior to forging
         (2) Dye penetrant 100-percent of accessible areas after forging
         (3) Hydrostatic test
         (4) Seat leakage test

      b. Case valves
         (1) Radiograph 100-percent [2]
         (2) Dye penetrant all accessible areas [2]
         (3) Hydrostatic test
         (4) Seat leakage

   3. Functional tests required for:
      a. Motor operated valves
      b. Auxiliary relief valves

B. Special manufacturing process control
   1. Weld, NDT, performance testing, assembly and inspection procedure review
   2. Surveillance of suppliers quality control and product
   3. Special weld process procedure qualification (e.g., hard facing)

Piping

A. Tests and inspections
   Class 1501 and below
   Seamless or welded. If welded 100-percent radiography is required, shop-fabricated and field-fabricated pipe weld joints are inspected as follows:

   2501R – 610R: 100-percent radiographic inspection and penetrant examination
   301R – 302R: 20-percent random radiographic inspection
   151R – 152R: 100-percent liquid penetrant examination

B. Special manufacturing process control
   Surveillance of suppliers quality control and product

Refueling Water Storage Tank

A. Tests and inspections
   1. Vacuum box test of tank bottom seams

TABLE 6.2-3 (Sheet 3 of 3)
Quality Standards Of Safety Injection System Components

2. Hydrostatic test of tank
3. Hydrostatic test of tank heater coil
4. Spot radiography of longitudinal and girth welds

B. Special manufacturing process control
1. Weld, fabrication, NDT, and inspection procedure review
2. Surveillance of suppliers quality control and product
3. Material chemical and physical properties certification

Notes:
1. Except Internal Recirculation Pump.
2. For valves with radioactive service only.

TABLE 6.2-4
Accumulator Design Parameters

Number	4
Type	Stainless steel lined/ carbon steel
Design pressure, psig	700
Design temperature, °F	300
Operating temperature, °F	100-150
Normal operating pressure, psig	Note 2
Total volume, ft^3 (each)	1100
Water volume at operating conditions, ft^3 (each)	Note 2
Minimum boron concentration (as boric acid), ppm	2000
Relief valve setpoint, psig[1]	700

Notes:
1. The relief valves have soft seats and are designed and tested to ensure exceptional tightness.

2. Minimum and maximum operating pressure and volume are controlled by Technical Specifications.

TABLE 6.2-5
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TABLE 6.2-6
Refueling Water Storage Tank Design Parameters

Number:	1
Material:	Stainless Steel
Nominal Capacity, gal.	350,000
Volume Required by Technical Specifications (solution), gal.	345,000
Normal pressure, psig	Atmospheric
Operating temperature, ^{o}F	40-110
Design pressure, psig	Atmospheric
Design temperature, ^{o}F	120
Boron concentration (as boric acid), ppm	2400 (minimum)
Type of heating	Steam

TABLE 6.2-7
Pump Design Parameters

Safety injection pump

Number	3
Design pressure, discharge, psig	1750
Design pressure, suction, psig	200
Design temperature, °F	285
Design flow rate, gpm	400
Maximum flow rate, gpm	650
Design head, ft	2500
Shutoff head, ft	3550
Material	Martensitic stainless steel
Motor, hp	400
Type	Horizontal centrifugal

Recirculation pump

Number of pumps	2
Type	Vertical centrifugal
Design pressure, discharge, psig	250
Design temperature, °F	300
Design flow, gpm	3000
Design head, ft	360
Material	Austenitic stainless steel
Maximum flow rate, gpm	4428
Shutoff head, ft	476
Motor, hp	350

Residual heat removal pump

Number of pumps	2
Type	Vertical centrifugal
Design pressure, discharge, psig	600
Design temperature, °F	400
Design flow, gpm	3000
Design head, ft	350
Material	Austenitic stainless steel
Maximum flow rate, gpm	5500
Shutoff head, ft	390
Motor, hp	400

TABLE 6.2-8
Residual Heat Exchangers Design Parameters

Heat Exchangers

Number		2
Design heat duty, Btu/hr (Normal)		30.8×10^6
Design UA_1, Btu/hr-°F		1.2×10^6
Design cycles (85°F – 350°F)		200
Type		Vertical shell and U-tube
Normal condition	Tube side	Shellside
Design pressure, psi	600	150
Design flow, lb/hr	1.44×10^6	2.46×10^6
Inlet temperature, °F	135	88.3
Outlet temperature, °F	113.5	100.8

Notes:
1. Total heat transfer coefficient x Area.

TABLE 6.2-9
Estimated External Recirculation Loop Leakage[3]

Items	Number of Units	Type of Leakage Control and Unit Leakage Rate	Leakage to Atmosphere (cm^3/hr)	Leakage to Tank (cm^3/hr)
Residual heat removal pumps (low-head safety injection)	2	Mechanical seal with leakoff-drop/min	0[1]	6
High-head safety injection pumps	3	Same as residual heat removal	0[1]	9
Flanges:		Gasket-adjusted to zero leakage following any test - 10 drops/min per flange		
a. Pump	15		450[1]	0
b. Valves - Bonnet to Body (larger than 2-in.)	16		480[2]	0
Valves - stem leakoffs	6	Backseated, double packing with leakoff 1 cm^3/hr-in.stem diameter	0[2]	6
Misc. small valves	23	Flanged body packed stems - 1 drop/min	69[2]	0
		Totals	999	21

Notes:

1. Total estimated leakage from RHR and SI pump mechanical seals and flanges is 450 cc/hr.
2. The total leakage estimated from all sources including valve stem leakage, packing leakoffs, flanged body packed stems and other potential sources (pumps and flanges) of External Recirculation Loop is 999 cc/hr.
3. Actual measured leakage is limited by Technical Specifications. The radiological consequences of external recirculation loop leakage following a design basis accident are presented in Section 14.3.6.6.

TABLE 6.2-10 (Sheet 1 of 3)
Single Active Failure Analysis – Safety Injection System

Component[1]	Malfunction	Comments
A. Accumulator (injection phase)	Deliver to broken loop	Totally passive system with one accumulator per loop. Evaluation based on three accumulators delivering to the core and one spilling from ruptured loop.
B. Pump: (injection phase)		
1. Safety injection	Fails to start	Three provided. Evaluation based on operation of two.
2. Residual heat removal	Fails to start	Two provided. Evaluation based on operation of one.
3. Essential service water	Fails to start	Three provided. Evaluation based on operation of two.
4. Component cooling[2]	Fails to start	A total of 1 of 3 required during recirculation.
5. Nonessential service water[2]	Fails to start	A total of 1 of 3 required during recirculation.
6. Recirculation[2]	Fails to start	Two provided. One required to operate during recirculation.
7. Auxiliary component cooling pump	Fails to start	Two provided. One required to operate during recirculation.

TABLE 6.2-10 (Sheet 2 of 3)
Single Active Failure Analysis – Safety Injection System

Component	Malfunction	Comments
C. Automatically operated valves: (open on safety injection signal) (injection phase)		
1. Safety injection line isolation valve at the loops	Fails to open (if closed)	Active failure to open is not credible since the injection valves are maintained in the open position when the reactor is critical.
2. Residual heat removal line isolation valve at residual heat exchanger discharge	Fails to open	Two parallel lines, one valve in either line is required to open.
3. Isolation valve on component cooling water line from residual heat exchangers	Fails to open	Two parallel lines, one valve in either line is required to open.
D. Valves operated from control room for recirculation: (recirculation phase)		
1. Recirculation sump internal recirculation isolation	Fails to open	Two lines in parallel, one valve in either line is required to open.
2. Safety injection pump suction valve at residual heat exchanger discharge	Fails to open	Two parallel lines, one valve in either line required to open.
3. Isolation valve on the mini-flow line returning to the refueling water storage tank	Fails to close	Two valves in series, one required to close.
4. Isolation at suction header from refueling water storage tank to safety injection pumps	Fails to close	Two valves in series, one required to close (one valve is a check valve).

TABLE 6.2-10 (Sheet 3 of 3)
Single Active Failure Analysis – Safety Injection System

Component	Malfunction	Comments
5. Residual heat removal pump recirculation line	Fails to close	Two valves in series, one required to close.
6. Residual heat removal pump discharge line	Fails to close	Two valves in series, one required to close (one valve is a check valve).

Notes:
1. The status of all active components of the safety injection system is indicated on the main control board. Reference is made to Table 6.2-2.
2. Recirculation phase.

TABLE 6.2-11
Single Passive Failure Analysis
(Loss Of Recirculation Flow Path)[5]

Flow Path	Indication Of Loss Of Flow Path	Alternative Flow Path[1]
Low head recirculation		
From recirculation sump to low-head injection header via the recirculation pumps and the residual heat exchangers	1. Insufficient flow in low-head injection lines (one flow monitor in each of the four low-head injection lines[2])	From recirculation sump to high-head injection header via the recirculation pumps, one of the two residual heat exchangers and the safety injection pump.[3]
	2. As 1 above.	a. From containment sump to discharge header of the residual heat exchangers via the residual heat removal pumps. b. If flow not established in low-head injection lines, as (a), except path is from discharge of one residual heat exchanger to the high-head injection header via the safety injection pumps.

TABLE 6.2-11 (Cont.)
Single Passive Failure Analysis
(Loss Of Recirculation Flow Path) (5)

Flow Path	Indication Of Loss Of Flow Path	Alternative Flow Path1
High-head recirculation		
From recirculation sump to high-head injection header via the recirculation pumps, one of the two residual exchangers and the high-head injection pumps	1. No flow in high-head injection header (four flow monitors, one in each cold leg injection line and one pressure monitor)	a. From containment sump to high head injection header via the residual heat removal pumps, one of the residual heat exchangers and the high-head injection pumps.
		b. If flow is not established in high-head injection header – as (a), except path is from discharge of the residual heat removal pumps to the high-head injection pumps via the middle safety injection pump (by-passing the residual heat exchangers 4).
	2. Flow in only one of the two high-head injection branch headers (two flow monitors per branch header)	a. As 1(b), except that flow from the middle safety injection pump is only supplied to the unbroken branch header.

Notes:
1. As shown in Plant Drawings 9321-2735 & 235296 [Formerly UFSAR Figure 6.2-1], there are valves at all locations where alternative flow paths are provided.
2. If minimum flow requirements have been established, the supply of recirculated water using low-head recirculation will maintain the core flooded even in the event of a low-head spilling line and one failed flow meter or other single failure.
3. Manual start.

TABLE 6.2-11 (Cont.)
Single Passive Failure Analysis
(Loss Of Recirculation Flow Path) [5]

4. In this recirculation mode, water is returned to the core without being cooled by the residual heat exchangers. Heat is removed from the core by boiloff of the water to the containment; heat is then removed from the containment by either the containment fan coolers and/or the containment spray system (using cooled water from the recirculation sump via the recirculation pumps and one residual heat exchanger).
5. Loss of the recirculation flow path due to a passive failure is not postulated until 24 hours into the accident (Reference Technical Specification Amendment #257).

TABLE 6.2-12
Shared Functions Evaluation

Component	Normal Operating Function	Normal Operating Arrangement	Accident Function	Accident Arrangement
Refueling water storage tank	Storage tank for refueling operations	Lined up to suction of safety injection, residual heat removal, and spray pumps	Source of borated water for core and spray nozzles	Lined up to suction of safety injection, residual heat removal, and spray pumps
Accumulators (4)	None	Lined up to cold legs of reactor coolant piping	Supply borated water to core promptly	Lined up to cold legs of reactor coolant piping
Safety injection pumps (3)	None	Lined up to hot and cold legs of reactor coolant piping	Supply borated water to core	Lined up to hot and cold legs of reactor coolant piping
Residual heat removal pumps (2)	Supply water to core to remove residual heat during shutdowns	Lined up to cold legs of reactor coolant piping	Supply borated water to core	Lined up to cold legs of reactor coolant piping
Recirculation pumps (2)	None	Lined up to cold legs of reactor coolant piping, spray headers and suction of safety injection pumps	Supply borated water to core and spray nozzles from recirculation sump	Lined up to cold legs of reactor coolant piping, spray headers and suction of safety injection pumps
Service water pumps (non-essential header) (3)	Supply river cooling water to component cooling heat exchangers and non-nuclear components	One or two pumps in service	Supply river cooling water to component cooling heat exchangers	Lined up to non-essential service water header [1]
Component cooling pumps (3)	Supply cooling water to station nuclear components	Up to three pumps in service	Supply cooling water to residual heat exchangers, S.I. pump bearings and recirculation pump motor coolers	Lined up to CCW header [1]
Residual heat exchangers (2)	Remove residual heat from core during shutdown	Lined up for residual heat removal pump	Cool recirculated water in containment for	Lined up to discharge of recirculation

		operation	core cooling and containment spray	pumps or RHR pumps
Component cooling heat exchangers (2)	Remove heat from component cooling water	One or two heat exchangers in service	Cool water for residual heat exchangers and other services	Both heat exchangers in service
Auxiliary component cooling pumps (2)	None	Lined up for pump operation	Provide component cooling water to recirculation pump motor coolers[2]	Lined up for pump operation[2]
Service water pumps (essential header) (3)	Supply river water to station safeguards	Up to three pumps in service	Supply river cooling water to safeguards components	Lined up to essential service water header

Notes:
1. Recirculation Phase.
2. These pumps start on a Safety Injection Signal and operate in the injection and recirculation phases of the accident. However, their function is not required during the injection phase. The supply of adequate cooling water to the recirculation pump motor coolers is required only during the recirculation phase.

TABLE 6.2-13
Accumulator Inleakage$_1$

Time Period Between Level Adjustments	Observed Leak Rate (cm^3/hr)	Observed Leak Rate Maximum Allowed Design
1 month	1955	99.8
3 months	665	33.3
6 months	333	16.7
9 months	221	11.1
1 year	167	8
10 years	16.7	0.8

Notes:
1. A total of 83.3-ft^3, added to the initial amount, can be accepted in each accumulator before an alarm is sounded.

TABLE 6.2-14
Residual Heat Removal System
Design, Operation, And Preoperational Test Conditions

	Pumps	Heat Exchangers	Valves	Pipes And Fittings
Design conditions				
Pressure, psig	600	600	665	700
Temperature, °F	400	400	400	400
Operating conditions (max.)- During recirculation[2]				
Pressure, psig	232	232	232	232
Temperature, °F	277	277	277	277
Preoperational Hydrostatic Test pressure, psig	1200	900	1100	900

Notes:
1. Located inside containment.
2. These maximum values have been conservatively calculated assuming saturated conditions at the containment design pressure of 47 psig to demonstrate that significant margin exists between design, normal operating and hypothetical accident conditions. These values are used to support the conservatism of Table 6.2-9 and are not used in the Chapter 14 Accident Analysis or for any other purpose.

6.2 FIGURES

Figure 6.2-9	Deleted

6.3 CONTAINMENT SPRAY SYSTEM

6.3.1 Design Bases

6.3.1.1 Containment Heat Removal Systems

Criterion: Where an active heat removal system is needed under accident conditions to prevent exceeding containment design pressure, this system shall perform its required function, assuming failure of any single active component. (GDC 52)

Adequate containment heat removal capability for the containment is provided by two separate, full capacity, engineered safety feature systems. The containment spray system, whose components operate in the sequential modes described in Section 6.3.2, and the containment air recirculation cooling system, which is discussed in Section 6.4.

The primary purpose of the containment spray system is to spray cool water into the containment atmosphere when appropriate in the event of a loss-of-coolant accident and thereby ensure that containment pressure does not exceed its design value, which is 47 psig at 271°F. (100-percent relative humidity) This protection is afforded for all pipe break sizes up to and including the hypothetical instantaneous circumferential rupture of a reactor coolant loop as discussed in UFSAR Section 14.3.5.1.1. Pressure and temperature transients for a loss-of-coolant accident are presented in Section 14.3. Although the water in the core after a loss-of-coolant accident is quickly subcooled by the safety injection system, the containment spray system design is based on the conservative assumption that the core residual heat is released to the containment as steam.

Any of the following combinations of equipment will provide sufficient heat removal capability to maintain the postaccident containment pressure below the design value, assuming that the core residual heat is released to the containment as steam.

1. Containment Spray alone as follows:
   - Both containment spray pumps operating up to the time the transfer to core recirculation flow begins (during injection phase).
   - One spray pump continuing to take suction from the RWST until the level in the RWST decreases to 2 feet.
   - Both recirculation pumps [Deleted], both residual heat exchangers and both containment recirculation spray headers in operation when the level in the RWST decreases below 2 feet.

2. All five containment cooling fans (to be discussed in Section 6.4).

3. One containment spray pump and three of the five containment cooling fans (the minimum containment safeguards case discussed in Section 14.3.5).

6.3.1.2 Inspection of Containment Pressure-Reducing Systems

Criterion: Design provisions shall be made to the extent practical to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems, such as pumps, valves, spray nozzles and sumps. (GDC 58).

Where practicable, all active components and passive components of the containment spray system are inspected periodically to demonstrate system readiness. The pressure-containing components are inspected for leaks from pump seals, valve packing, flanged joints, and safety valves. During operational testing of the containment spray pumps, the portions of the system subjected to pump pressure can be inspected for leaks. Design provisions for inspection of the safety injection system, which also function as part of the containment spray system, are described in Section 6.2.5.

6.3.1.3 Testing of the Containment Pressure-Reducing Systems Components

Criterion: The containment pressure-reducing systems shall be designed, to the extent practical so that active components, such as pumps and valves, can be tested periodically for operability and required functional performance. (GDC 59)

All active components in the containment spray system are adequately tested both in preoperational performance tests in the manufacturer's shop and inplace testing after installation. Thereafter, periodic tests are also performed as required after any component maintenance. Testing of the components of the safety injection system that are used for containment spray purposes is described in Section 6.2.5.

The component cooling water pumps and the non-essential service water pumps that supply cooling water to the residual heat exchangers are in operation on a relatively continuous schedule during plant operation. Those pumps not running during normal operation may be tested by changing the operating pump(s).

6.3.1.4 Testing of Containment Spray Systems

Criterion: A capability shall be provided to the extent practical to test periodically the operability of the containment spray system at a position as close to the spray nozzles as is practical. (GDC 60)

Permanent test lines for the containment spray loops are located so that all components up to the containment isolation valves upstream of the spray nozzles may be tested. These isolation valves and spray nozzles are tested separately.

Each spray pump is provided with a recirculation line from the pump discharge line back to the pump suction line with a globe valve to allow greater flow through the pump during surveillance testing of the pump. An ultrasonic flow instrument is installed on each recirculation line during testing to allow resetting of the globe valve, after testing to the original flow value of the eductor (112 gpm). Note the globe valve replaced the eductor.

Temporary test connections, downstream of the isolation valves, are provided to verify that spray nozzles are not obstructed. Air flow through the nozzles will be monitored by means of the helium-filled balloon method, or by using an infrared scanning technique.

6.3.1.5 Testing of Operational Sequence of Containment Pressure-Reducing Systems

Criterion: A capability shall be provided to test initially under conditions as close as practical to the design and the full operational sequence that would bring the containment pressure reducing systems into action, including the transfer to alternate power sources. (GDC 61)

Capability is provided to test initially to the extent practical the operational startup sequence of the containment spray system including the transfer to alternative power sources.

6.3.1.6 Performance Objectives

The containment spray system is designed to spray at least 5000 gpm of borated water into the containment whenever the coincidence of two sets of two out of three (Hi-Hi) containment pressure (approximately 50-percent of design value) signals occur or when a manual signal is initiated. Either of two subsystems containing a pump and associated valving and spray header is independently capable of delivering one-half of the designed flow, or 2500 gpm, which exceeds the minimum containment spray flow of 2180 gpm assumed in the Containment Analysis as described in Table 14.3-40.

The design basis for the containment spray system is, full capacity flow will provide sufficient heat removal capability to maintain the post accident containment pressure below 47 psig, assuming that the core residual heat is released to the containment as steam.

A second purpose served by the containment spray system is to remove elemental iodine and particulates from the containment atmosphere should they be released in the event of a loss-of-coolant accident. The analysis, indicating the system's ability to limit the offsite dose to within applicable limits after a hypothetical loss-of-coolant accident is presented in Section 14.3.6.

To meet the above bases, the following design requirements were established:

1. All components of the system have to meet Class I seismic criteria.
2. The system's initial response has to be fully automatic.
3. Total redundancy of equipment, flow paths, and power supply.
4. Provisions for periodic testing have to be provided.
5. Equipment is to be arranged to provide maximum protection from missiles.

The spray system, including recirculation spray, is designed to operate over an extended time period following a reactor coolant system failure, as required to restore and maintain containment conditions at or near atmospheric pressure. It has the capability of reducing the containment postaccident pressure and subsequent containment leakage. A tertiary function of the system is to provide an alternative means of filling the reactor refueling cavity during reactor vessel head removal.

Portions of other systems that share functions and become part of the containment spray system, when required, are designed to meet the criteria of the containment cooling function. Neither a single active component failure in such systems during the injection phase nor an active/passive failure during the recirculation phase will degrade the design heat removal capability of containment cooling (See section 6.2.3.3).

System piping located within the containment is redundant and separable in arrangement unless fully protected from damage that may follow any reactor coolant system loop failure.

System isolation valves relied upon to operate for containment cooling are redundant with automatic actuation.

6.3.1.7 Service Life

All portions of the system located within containment are designed to withstand, without loss of functional performance, the postaccident containment environment and to operate without benefit of maintenance for the duration of time required to restore and maintain containment conditions at near atmospheric pressure. The recirculation pumps are designed to be operable for 1 yr following a loss-of-coolant accident. Per the 12/06/04 NRC generic SER on NEI 04-07 (Reference 7), evaluations of PWR post accident emergency recirculation performance to address the potential impact of debris blockage per GSI-191 (Reference 6) and Generic Letter 2004-02 (Reference 5) will use a mission time of 30 days.

6.3.1.8 Codes and Classifications

Table 6.3-1 tabulates the codes and standards to which the containment spray system components are designed.

6.3.2 System Design And Operation

6.3.2.1 System Description

Adequate containment cooling and iodine removal by the containment spray system are provided by system components operating in sequential modes. These modes are:

1. Spray a portion of the contents of the refueling water storage tank into the entire containment atmosphere using the containment spray pumps.

2. Recirculation of water from the containment sump by the diversion of a portion of the recirculation flow from the safety injection system to the spray headers inside the containment after injection from the refueling water storage tank has been terminated.

The bases for the selection of the various conditions requiring system actuation are presented in Section 14.3.

The system diagram for the containment spray system is shown in Plant Drawings 9321-2735 & 235296 [Formerly UFSAR Figure 6.2-1].

The principal components of the containment spray system that provide containment cooling and iodine removal following a loss-of-coolant accident consist of two spray pumps, Sodium Tetraborate baskets located in containment, spray ring headers and nozzles, and the necessary piping and valves. The containment spray pumps are located in the primary auxiliary building and the spray pumps take suction directly from the refueling water storage tank.

The containment spray system also uses the two 100-percent capacity recirculation pumps, two residual heat removal heat exchangers and associated valves and piping of the safety injection

system for the long-term recirculation phase of containment cooling and iodine removal after the refueling water storage tank has been exhausted.

The Containment Spray System suction piping and the Containment Spray pumps up to the first closed discharge line isolation valve will be maintained sufficiently full of water to ensure the system remains operable and performs properly.

The spray water is injected into the containment through spray nozzles connected to four 360 degree ring headers located in the containment dome area. Each of the spray pumps supplies two of the ring headers.

6.3.2.1.1 Injection Phase

Containment spray will be actuated by two sets of coincidence logic circuits each requiring two-out-of-three, high-high containment pressure signals. This starting signal will start the pumps and open the discharge valves to the spray header.

6.3.2.1.2 Recirculation Phase

When the refueling water storage tank level drops below 2 feet and its contents have been added to the containment floor, recirculation spray flow will be initiated since the NPSH requirements for the recirculation pumps are met for the additional flow needed for combined core injection and recirculation spray. The operator can remotely open the stop valves on either of the two spray recirculation lines. With this split flow, decay heat can be removed and containment airborne activity reduced. This mode of operation will be continued for a period of at least 3.4 hr following the accident in order to continue removal of airborne activity from the containment atmosphere.

After the 3.4 hr containment scrubbing operation it is expected that spray flow would be discontinued while maintaining containment pressure with the containment fan-cooler units, and returning all of the recirculated water to the core. In this mode, the bulk of the core residual heat is transferred directly to the sump by the spilled coolant to be eventually dissipated through the residual heat removal heat exchanger once the sump water becomes heated. The heat removal capacity of three of the five fan coolers is sufficient to remove the corresponding energy addition to the vapor space as a result of steam boiloff from the core, assuming flow into the core from one recirculation pump at the termination of injection spray without exceeding containment design pressure; hence, it is not expected that continued spray operation for containment heat removal would be required. Spray flow termination is also assumed in the chemical generation analyses for GL 2004-02 compliance. Longer spray times increase exposure time of Aluminum components in the containment to the spray solution and may result in additional chemicals (precipitants) being generated than accounted for in sump strainer head loss calculations. [Deleted]

Sodium Tetraborate is stored at elevation 46’ inside the containment building. During the injection phase the level of the boric acid solution from the containment spray and the coolant lost from the reactor coolant system will rise above the Sodium Tetraborate bins. The Sodium Tetraborate will dissolve into the solution, providing a solution with pH in the range of 7 to 7.6 to enhance long-term iodine retention in the solution and to minimize corrosion.

6.3.2.1.3 Cooling Water

The cooling water for the residual heat exchangers has been described in Section 6.2.

6.3.2.1.4 Changeover

The sequence for the changeover from injection to recirculation has been described in Section 6.2.

Remote-operated valves of the containment spray system that are under manual control (that is, valves that normally are in their ready position and do not receive a containment spray signal) have their positions indicated on a common portion of the control board. At any time during operation when one of these valves is not in the ready position for injection, an audible and visual annunciation is provided on the board to alert the operator of this condition.

6.3.2.2 Components

All associated components, piping, structures, and power supplies of the containment spray system are designed to Class I seismic criteria.

All components inside containment are capable of withstanding or are protected from differential pressures that may occur during the rapid pressure rise to 47 psig in 10 sec. Section 14.3.5.1.1 discusses the analyses that show that the calculated postaccident containment pressures are less severe than this. The lines of the system are protected from missile damage by the concrete crane wall and operating floor.

Parts of the system in contact with the spray solution are stainless steel or an equivalent corrosion-resistant material.

The containment spray system shares the refueling water storage tank capacity with the safety injection system. For a detailed description of this tank, see Section 6.2.

6.3.2.2.1 Pumps

The two containment spray pumps are of the horizontal centrifugal type driven by electric motors.

The design head of the pump is sufficient to continue at rated capacity, with a minimum level in the refueling water storage tank, against a head equivalent to the sum of the design pressure of the containment, the head to the uppermost nozzles, and the line and nozzle pressure losses. Pump motors are direct-coupled and large enough for maximum power requirement of the pump. The materials of construction, which are suitable for use with the spray solutions, are stainless steel or equivalent corrosion-resistant material. Design parameters are presented in Table 6.3-2 and the containment spray pump characteristics are shown on Figure 6.3-1.

The containment spray pumps are designed in accordance to the specifications discussed for the pumps in the safety injection system, Section 6.2.

The recirculation pumps of the safety injection system, which provide flow to the containment spray system during the recirculation phase, are described in Section 6.2.

Details of the component cooling pumps and service water pumps, which serve the safety injection system, are presented in Sections 9.3 and 9.6.

6.3.2.2.2 Heat Exchangers

The two residual heat removal heat exchangers of the safety injection system, which are used during the recirculation phase, are described in Section 6.2.

6.3.2.2.3 Spray Nozzles

The spray nozzles, which are of the hollow cone, ramp bottom design, are not subject to clogging by particles 0.25-in. in maximum dimension, and are capable of producing a surface area averaged drop diameter of approximately 1000 microns at 15 gpm and 40 psi differential pressure. With the spray pump operating at design conditions and the containment at design pressure the pressure drop across the nozzles will exceed 40 psi.

During recirculation spray operation, the water is screened through the 3/32" diameter holes of the perforated plate strainer modules before leaving the recirculation or containment sump. The spray nozzles are stainless steel and have a 0.375-in. diameter orifice. The nozzles are connected to four 360 degree ring headers (alternating headers connected) of radii 7-ft 1.75-in. (El. 228.5-ft), 25-ft 3.438-in. (El. 223.5-ft), 42-ft 3-in. (El. 218.5-ft) and 59-ft 6-in. (El. 213.5-ft). There are 315 nozzles distributed on the four headers. This nozzle and header arrangement results in maximum area coverage with either branch of the system operating alone, while ensuring minimum overlap of spray trajectories in the minimum flow case (Section 14.3).

6.3.2.2.4 Spray Additive Tank

The spray additive tank was removed based on the use of Trisodium Phosphate baskets stored in the Containment building. In response to NRC Generic Letter 2004-02 (Generic Safety Issue 191), the pH buffer material was changed from Trisodium Phosphate to Sodium Tetraborate to minimize the potential for sump screen blockage due to the formation of chemical products. The Sodium Tetraborate baskets are described in Section 6.3.2.2.12.

6.3.2.2.5 Spray Pump Recirculation Line

Each spray pump is provided with a recirculation line from the pump discharge line back to the pump suction line. A globe valve is installed in this line to allow setting of the desired flow rate for both on line and testing configurations.

6.3.2.2.6 Valves

The valves for the containment spray system are designed in accordance with the specifications discussed for the valves in the safety injection system (Section 6.2).

6.3.2.2.7 Piping

The piping for the containment spray system is designed in accordance to the specifications discussed for the piping in the safety injection system (Section 6.2).

The system is designed for 150 psig at 300^{o}F on the suction side and 300 psig at 300^{o}F on the discharge side of the spray pumps.

6.3.2.2.8 Motors for Pumps and Valves

The motors inside and outside containment for the containment spray system are designed in accordance with the specifications discussed for motors in the safety injection system (see Section 6.2).

6.3.2.2.9 Electrical Supply

Details of the normal and emergency power sources are presented in the discussion of the electrical system, Chapter 8.

6.3.2.2.10 Missile Protection

The spray headers are located outside and above the reactor and steam generator concrete shields. A shield, which is removable for refueling also provides missile protection for the area immediately above the reactor vessel. The spray headers are therefore protected from missiles originating within the reactor coolant system.

6.3.2.2.11 Material Compatibility

Parts of the system in contact with the spray solutions are stainless steel or an equivalent corrosion-resistant material. An analysis of materials compatibility with the long-term storage conditions of concentrated sodium hydroxide is presented in Appendix 6D. Appendix 6D is being retained for historical purposes.

All exposed surfaces within the containment have coatings that will not be affected by short term exposure to low pH containment spray solution or to long term exposure to high pH solution. An analysis of the materials exposed to the Post-accident Containment Environment using the original containment spray additive (NaOH) solution is presented in Appendix 6C.

Post-accident chemistry changes due to the elimination of the spray additive tank and the installation of Trisodium Phosphate Baskets were evaluated and it was determined that this change has little effect on the compatibility of materials located in containment, which will come in contact with the initial spray and recirculation spray solution. This evaluation is documented in Reference 2. This evaluation supersedes the information contained in Appendix 6C, with the exception of sections 6C.4.1, and 6C.7. Therefore, with these exceptions, Appendix 6C is being retained for historical purposes.

To improve post-accident ECCS performance, specifically in order to meet the requirements of Generic Letter 2004-02 (Generic Safety Issue 191), the Trisodium Phosphate pH buffer was replaced with Sodium Tetraborate (Reference 3). This buffer material replacement has also been evaluated with respect to post-accident chemistry and material interaction. The evaluation is documented in Reference 4 and concluded that the pH buffer replacement is acceptable and does not detrimentally affect material compatibility. Appendix 6C has been updated where appropriate to include post accident buffer change to Sodium Tetraborate.

Maintaining the long-term pH of the recirculated ECC solution no less than 7.0, prevents chloride-induced stress corrosion cracking of austenitic stainless steel components, and minimizes hydrogen produced by the corrosion of galvanized surfaces and zinc-based paints as discussed in Reference 1. These chemistry changes using Sodium Tetraborate also do not

affect the environmental qualification of equipment located within the containment required to mitigate the consequences of design basis accidents as discussed in Section 7.1.4.

6.3.2.2.12 Sodium Tetraborate Baskets

Sodium Tetraborate (STB) is stored in four baskets at elevation 46' in the containment building. During the injection phase the baskets will be flooded, allowing the STB to dissolve into the fluid for pH control. The four baskets are constructed of stainless steel and are seismically qualified and mounted.

6.3.3 Design Evaluation

6.3.3.1 Range of Containment Protection

For the first 15 to 20 min following the maximum loss-of-coolant accident (i.e., during the time that the containment spray pumps take their suction from the refueling water storage tank), this system provides the design heat removal capacity for the containment. After the injection phase, one spray pump continues to take suction from the RWST and spray into the containment until RWST level drops below 2 feet. This continued spray injection is sufficient to maintain the containment pressure below the design value even if no containment fans were operating.

With the completion of containment spray injection the operator sets up recirculation to one spray header and to the core; the systems are aligned so that sufficient cooled recirculated water is delivered to keep the core flooded as well as to provide flow to one spray header. Flow is maintained to the spray header at this stage primarily to continue scrubbing of airborne activity, i.e., for at least 3.4 hr after the accident; the flow, however, is also sufficient to maintain the containment pressure below the design value. Spray flow termination is also assumed in the chemical generation analyses for GL 2004-02 compliance. Longer spray times increase exposure time of Aluminum components in the containment to the spray solution and may result in additional chemicals (precipitants) being generated than accounted for in sump strainer head loss calculations.

Any of the following combinations of equipment will provide sufficient heat removal capability to maintain the post-accident containment pressure below the design value, assuming that the core residual heat is released to the containment as steam.

1. Containment Spray alone as follows:
   - Both containment spray pumps operating up to the time the transfer to core recirculation flow begins (during injection phase).
   - One spray pump continuing to take suction from the RWST until the level in the RWST decreases to 2 feet.
   - Both recirculation pumps [Deleted], both residual heat exchangers and both containment recirculation spray headers in operation when the level in the RWST decreases below 2 feet.

2. All five containment cooling fans (discussed in Section 6.4)

3. One containment spray pump and three of the five containment cooling fans (the minimum containment safeguards case discussed in Section 14.3.5).

During the injection and recirculation phases the spray water is raised to the temperature of the containment in falling through the steam-air mixture. The minimum fall path of the droplets is approximately 118-ft from the lowest spray ring headers to the operating deck. The actual fall path is longer due to the trajectory of the droplets sprayed out from the ring header. Drops of approximately 1000 micron average size will reach temperature equilibrium with the steam-air containment atmosphere after falling through less than half the available spray fall height as discussed in UFSAR Section 14.3.5.2.1.

At containment design temperature, 271^{o}F, the total design heat absorption capability of one spray pump is 218 x 10^{6} Btu/hr based on the assumption of 100^{o}F refueling water and design flow of 2500 gpm.

When the refueling water storage tank level drops below 2 feet, injection spray is terminated and the recirculation pumps supply the flow to the containment recirculation spray headers. Recirculation spray can be established at a flow rate that will maintain containment pressure below the design pressure of 47 psig even if no containment fan coolers are operating.

Elemental iodine and aerosols are removed by the containment spray system. Removal coefficients and the limitations on removal are discussed in Appendix 6A. A discussion of the effectiveness of containment spray as a fission product trapping process is contained in Reference 1.

A single train of containment spray will provide sufficient iodine removal capability to ensure postaccident fission product leakage that would not result in exceeding the applicable dose limits. This is evaluated in Section 14.3.6.

6.3.3.2 System Response

The starting sequence of the containment spray pumps and their related emergency power equipment is discussed in sections 7.2 and 8.2.3.4 and their analyzed performance is discussed in the various Chapter 14 safety analyses.

6.3.3.3 Single Failure Analysis

A failure analysis has been made on all active components of the system to show that the failure of any single active component will not prevent fulfilling the design function. This analysis is summarized in Table 6.3-4.

In addition, each spray header is supplied from the discharge from one of the two residual heat removal heat exchangers. As described in Section 6.2.2.1.2, these two heat exchangers are redundant and can be supplied with recirculated water via separate and redundant flow paths. The analysis of the loss-of-coolant accident presented in Section 14.3 reflects the single failure analysis.

6.3.3.4 Reliance on Interconnected Systems

For the injection phase, the containment spray system operates independently of other engineered safety features following a loss-of-coolant accident except that it shares the source of water in the refueling water storage tank with the safety injection system. The system acts as a backup for the cooling function of the containment air recirculation cooling system. For extended operation in the recirculation mode, water is supplied through recirculation pumps.

During the recirculation phase, some of the flow leaving the residual heat removal heat exchangers may be diverted to the containment spray headers or the high-head safety injection pumps. Minimum flow requirements are established for the flow being sent to the core and for the flow being sent to the containment spray. Sufficient flow instrumentation is provided so that the operator can monitor each flow path as shown in Plant Drawings 9321-2735 & 235296 [Formerly UFSAR Figure 6.2-1].

Normal and emergency power supply requirements are discussed in Chapter 8.

6.3.3.5 Shared Functions Evaluation

Table 6.3-5 contains an evaluation of the main components that have been discussed previously and a brief description of how each component functions during normal operation and during the accident.

6.3.3.6 Containment Spray Pump Net Positive Suction Head Requirements

The net positive suction head for the containment spray pumps is evaluated for injection operation. The end of the injection phase gives the limiting net positive suction head requirement. The net positive suction head available is determined from the elevation head and vapor pressure of the water in the refueling water storage tank, and the pressure drop in the piping to the pump. At the end of the injection phase, the net positive suction head available exceeds the net positive suction head required.

6.3.4 Minimum Operating Conditions

The Technical Specifications establish limiting conditions regarding the operability of the system.

6.3.5 Inspections And Tests

6.3.5.1 Inspections

All components of the containment spray system may be inspected periodically to demonstrate system readiness.

The pressure-containing systems can be inspected for leaks from pump seals, valve packing, flanged joints, and safety valves during system testing. During the operational testing of the containment spray pumps, the portions of the system subjected to pump pressure can be inspected for leaks.

6.3.5.2 Preoperational Testing

The principal components of the containment spray system are two pumps, spray ring headers and nozzles, and the necessary piping and valves.

In discussing preoperational testing and generally proving that the system meets the design specification, it is necessary to consider both individual component testing and onsite testing.

6.3.5.2.1 Offsite Test Work

Three components in the system were subjected to offsite test work:

1. Spray pumps - The spray pumps were subjected to conventional acceptance tests and the performance characteristic was plotted to illustrate the pumps met the design specification.

2. Spray nozzles – As part of the development work in support of Westinghouse Plants, a nozzle of the type used in the spray system was subjected to a performance test to demonstrate and prove the nozzle characteristic, e.g., flow/pressure drop, droplet size, spread of spray, etc.

   As part of the quality assurance program, a random 25-percent of the nozzles installed at the Indian Point Unit 2 site were given a general performance test.

6.3.5.2.2 Onsite Test Work

The aim of onsite preoperational testing was to:

1. Demonstrate and prove that the system is adequate to meet the design pressure conditions. Outside the containment, this involved partial radiographic inspection and partial hydro-testing; inside the containment, the spray headers were subjected to 100-percent radiographic inspection.

2. Demonstrate that the spray nozzles in the containment spray header are clear of obstructions by passing air through the test connections.

3. Verify that the proper sequencing of valves and pumps occurs on initiation of the containment spray signal and demonstrate the proper operation of all remotely operated valves

4. Verify the operation of the spray pumps; each pump was run at shutoff and the mini-flow directed through the normal path back to the refueling water storage tank. During this time, the mini-flow was adjusted to that required for routine testing.

5. Demonstrate the operation of the spray eductors. The eductor and spray additive system was checked by running, in turn, each spray pump on mini-flow with the spray additive tank filled with water and open to the spray eductor suction. During draindown of the spray additive tank, the tank level and corresponding eductor suction flow was recorded via the system instrumentation. Finally, the system performance with water was extrapolated to that with sodium hydroxide, and the adequacy of the system thus verified.

   In order to establish a reference eductor suction test flow for routine testing of the system, the above test was made with the spray additive tank isolated and the eductor drawing water through the refueling water storage tank/eductor suction test line.

6.3.5.2.3 System Testing

The functional test of the injection system described in Section 6.2.5 demonstrates proper transfer to the emergency diesel generator power source in the event of a loss of power. A test signal simulating the containment spray signal is used to demonstrate the operation of the spray system up to the isolation valves on the pump discharge. The isolation valves are blocked closed for the test. These isolation valves are checked separately.

6.3.5.3 Postoperational Testing

6.3.5.3.1 Component Testing

Routine periodic testing of the containment spray system components and all necessary support systems at power is performed. When testing indicates a need for corrective maintenance, the redundancy of equipment in these systems permits such maintenance to be performed without shutting down or reducing load under conditions defined in the Technical Specifications. These conditions would include such matters as the period within which the component should be restored to service and the capability of the remaining equipment to meet safety limits within such a period.

6.3.5.3.2 Routine Inservice Testing

The aim of the periodic test is:

1. To verify that the proper sequencing of valves and pumps occurs on initiation of the containment spray signal and demonstrate the proper operation of all remotely operated valves.

2. To verify the operation of the spray pumps; each pump will be run at shutoff and the mini-flow directed through the normal path back to the refueling water storage tank.

During these tests the equipment can be visually inspected for leaks, leaking seals, or packing; or flanges are tightened to eliminate the leak, and valves and pumps operated and inspected after any maintenance to ensure proper operation.

6.3.5.3.3 System Testing

The post operational testing of the safety injection system is described in Section 6.2.5. Section 8.5 describes the testing required to demonstrate proper transfer to the emergency diesel-generator power source in the event of a loss of power.

REFERENCES FOR SECTION 6.3

1. Letter from Jefferey F. Harold, NRC, to Stephen E. Quinn, Con Edison, Subject: Issuance of Amendment No. 191 for Indian Point Nuclear Generating Unit No.2, dated April 23,1997

2. “Elimination of the Emergency Containment Filtration and Spray Additive Systems from Indian Point Nuclear Generating Station Unit No.2” WCAP-14542 (Westinghouse Non-Proprietary Class 3)

3. Letter from John P. Boska, NRC, to Michael A. Balduzzi, Entergy Nuclear Operations, Subject: Indian Point Nuclear Generating Unit No. 2 – Issuance of Amendment 253, dated February 7, 2008

4. “Evaluation of Alternative Emergency Core Cooling System Buffering Agents,” WCAP-16596-NP Revision 0, July 2006

5. NRC Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors”, dated September 13, 2004.

6. NRC Generic Safety Issue (GSI) 191, “Assessment of Debris Accumulation on PWR Sump Performance.”

7. NRC Generic SER dated 12/06/04 on the NEI 04-07 Guidance Report entitled, “Pressurized Water Reactor Sump Performance Evaluation Methodology.”

TABLE 6.3-1
Containment Spray System - Code Requirements

Component	Code
Valves	USAS B16-5
Piping (including headers and spray nozzles)	USAS B31.1

TABLE 6.3-2
Containment Spray System Design Parameters

Pumps	
Quantity	2
Design pressure, discharge, psig	300
Design pressure, suction, psig	150
Design temperature, ^{o}F	150
Design flow rate, gpm	2600
Design head, ft	450
Maximum pump flow rate, gpm	3450
Shutoff head, ft	490
Motor, hp	400
Type	Horizontal-centrifugal

TABLE 6.3-3
DELETED

TABLE 6.3-4 (Sheet 1 of 2)
Single Failure Analysis - Containment Spray System

Component	Malfunction	Comments and Consequences
A. Spray nozzles	Clogged	Large number of nozzles (315) renders clogging of a significant number of nozzles incredible.
B. Pumps 1) Containment spray pump	Fails to start	Two provided. Evaluation based on operation of one pump in addition to three out of five containment cooling fans operating during injection phase.
2) Recirculation pump	Fails to start	Two provided. Evaluation based on operation of one pump in addition to three out of five containment cooling fans operating during recirculation phase.
3) Non-essential service water	Fails to start	Three provided. Operation of one pump during recirculation required.
4) Component Cooling	Fails to start	Three provided. Operation of one pump during recirculation required.
5) Auxiliary Component Cooling Pump	Fails to start	Two provided. Operation of one pump during recirculation required.

TABLE 6.3-4 (Sheet 2 of 2)
Single Failure Analysis - Containment Spray System

Component	Malfunction	Comments and Consequences
C. Automatically Operated Valves: (Open on coincidence of two -2/3 high containment pressure signals)		
1) Containment spray pump discharge isolation valve	Fails to open	Two provided. Operation of one required.
2) Isolation valve on component cooling water lines from residual heat exchangers	Fails to open	Two parallel lines, one valve in either line is required to open.
D. Valves operated from control room for recirculation		
1) Containment sump recirculation isolation	Fails to open	Two lines in parallel, one valve in either line is required to open.
2) Containment spray header isolation valve from residual heat exchangers	Fails to open	Two valves provided. Operation of one required.
3) Residual heat removal pump recirculation line	Fails to close	Two valves in series, one required to close.
4) Residual heat removal pump discharge line	Fails to close	Two valves in series, one required to close (one valve is a check valve).
E. Automatically operated valves (Close from control room on injection to recirculation changeover)		
1) Isolation valves at spray pump discharge	Fails to close	Check valve in series with two parallel valves provided. Operation of one of the two valve arrangement series required.

TABLE 6.3-5
Shared Functions Evaluation

Component	Normal Operating Function	Normal Operating Arrangement	Accident Function	Accident Arrangement
Containment Spray Pumps (2)	None	Lined up to spray headers	Supply spray water to containment atmosphere	Lined up to spray headers

NOTE: Refer to Section 6.2 for a brief description of the refueling water storage tank, recirculation pumps, non-essential service water pumps, component cooling pumps, residual heat exchangers, component cooling heat exchangers and the auxiliary component cooling pumps, which are also associated either directly or indirectly with the containment spray system.

6.3 FIGURES

Figure No.	**Title**
Figure 6.3-1	Containment Spray Pump Performance Characteristics

6.4 CONTAINMENT AIR RECIRCULATION COOLING SYSTEM

6.4.1 Design Basis

6.4.1.1 Containment Heat Removal Systems

Criterion: Where an active heat removal system is needed under accident conditions to prevent exceeding containment design pressure, this system shall perform its required function, assuming failure of any single active component. (GDC 52)

Adequate heat removal capability for the containment is provided by two separate, full capacity, engineered safety features systems. These are the containment spray system, whose components are described in Sections 6.2 and 6.3 and the containment air recirculation cooling and filtration system, whose components operate as described in Section 6.4.2. These systems are of different engineering principles and serve as independent backups for each other. Together these two systems provide the single failure protection for the containment cooling function as analyzed in Chapter 14.

The containment air recirculation cooling system is designed to recirculate and cool the containment atmosphere in the event of a loss-of-coolant accident and thereby ensure that the containment pressure will not exceed its design value of 47 psig at 271°F (100-percent relative humidity). Although the water in the core after a loss-of-coolant accident is quickly subcooled by the safety injection system, the containment air recirculation cooling system is designed on a conservative assumption that the core residual heat is released to the containment as steam.

Any of the following combinations of equipment will provide sufficient heat removal capability to maintain the postaccident containment pressure below the design value, assuming that the core residual heat is released to the containment as steam.

1. All five containment cooling fans.

2. Containment Spray alone as follows:
   - Both containment spray pumps operating up to the time the transfer to core recirculation flow begins (during injection phase).
   - One spray pump continuing to take suction from the RWST until the level in the RWST decreases to 2 feet.
   - Both recirculation pumps [Deleted], both residual heat exchangers and both containment recirculation spray headers in operation when the level in the RWST decreases below 2 feet.

3. One containment spray pump and three of the five containment cooling fans (the minimum containment safeguards case discussed in Section 14.3.5).

6.4.1.2 <u>Inspection of Containment Pressure-Reducing Systems</u>

Criterion: Design provisions shall be made to extent practical to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems, such as pumps, valves, spray nozzles, torus, and sumps. (GDC 58)

Design provisions are made to the extent practical to facilitate access for periodic visual inspection of all important components of the containment air recirculation cooling system.

6.4.1.3 <u>Testing of Containment Pressure-Reducing Systems Components</u>

Criterion: The containment pressure-reducing systems shall be designed to the extent practical so that components, such as pumps and valves, can be tested periodically for operability and required functional performance. (GDC 59)

The containment air recirculation cooling system is designed to the extent practical so that the components can be tested periodically, and after any component maintenance, for operability and functional performance.

A number of air recirculation and cooling units are normally in operation and no additional periodic tests are required. The service water pumps that supply the cooling units can be part flow-tested during plant operation via the installed bypass test loop.

6.4.1.4 <u>Testing of Operational Sequence of Containment Pressure-Reducing Systems</u>

Criterion: A capability shall be provided to test initially under conditions as close as practical to the design and the full operational sequence that would bring the containment pressure-reducing systems into action, including the transfer to alternate power sources. (GDC 61)

Means are provided to test initially to the extent practical the full operational sequence of the air recirculation system including transfer to alternative power sources.

6.4.1.5 Inspection of Air Cleanup Systems

Criterion: Design provisions shall be made to the extent practical to facilitate physical inspection of all critical parts of containment air cleanup system, such as, ducts, filters, fans, and dampers. (GDC 62)

Access is available for periodic visual inspection of the containment of recirculation cooling system components.

6.4.1.6 Testing of Air Cleanup Systems Components

Criterion: Design provisions shall be made to the extent practical so that active components of the air cleanup systems, such as fans and dampers, can be tested periodically for operability and required functional performances. (GDC 63)

The valves in a nonoperating unit can be periodically tested by actuating the controls and verifying deflection by instruments in the control room. A number of fans are normally in operation; no additional periodic fan tests are necessary.

6.4.1.7 Deleted

6.4.1.8 Deleted

6.4.1.9 Performance Objectives

The containment ventilation system, discussed in Section 5.3, of which all of the components of the containment air recirculation cooling system are a part, is designed to remove the normal heat loss from equipment and piping in the reactor containment during plant operation and to remove sufficient heat from the reactor containment, following the initial loss-of-coolant accident containment pressure transient, to keep the containment pressure from exceeding the design pressure as discussed in Section 14.3.5. The fans and cooling units continue to remove heat after the loss-of-coolant accident and reduce the containment pressure close to atmospheric within the first 24 hr as discussed in Section 14.3.5.1.3. The fan-cooler units could operate continuously after the loss-of-coolant accident and are designed to be operable for 1 year.

In addition to the design bases specified above, the following objectives are met to provide the engineered safety features functions:

1. The heat transfer rate that is assigned to the currently installed fan-cooler units under accident conditions is shown in Figures 14.3-104B and 14.3-104D. The establishment of basic heat transfer design parameters for the cooling coils of fan cooler units are discussed in Section 14.3.5.2.2. Among the topics covered are selection of the tube-side fouling factor, effect of air-side pressure drop, effect of moisture entrainment in the air steam mixture entering the fan coolers, and calculation of the various air-side to water-side heat transfer resistances.

2. In removing heat at the design basis rate, the coils are capable of discharging the resulting condensate without impairing the flow capacity of the unit and without raising the exit temperature of the service water to the boiling point. Since condensation of water from the air-steam mixture is the principal mechanism for removal of heat from the postaccident containment atmosphere by the cooling coils, the coil fins will operate as wetted surfaces under these conditions. Entrained water droplets added to the air-steam mixture, such as by operation of the containment spray system, will therefore have essentially no effect on the heat removal capability of the coils.

3. Each of the five air-handling units is equipped with moisture separators rated for full unit flow.

In addition to the above design bases, the equipment was originally specified to be capable of withstanding, without impairing operability, a pressure of 70.5 psig and 298°F for a period of one hour. The motors were further specified to be capable of running for 48 hours at required fan load in an atmosphere consisting of an air water vapor mixture initially at 47 psig and 271°F, and of continuous operation at 10 psig and 175°F. These ambient conditions and operating times have been updated and are maintained by the ongoing Environmental Qualification Program discussed in Section 7.1.4. As part of this program, the fan motors are qualified to withstand containment environment conditions following the loss of coolant accident so that the fans can perform their required function during the recovery period (1 year).

All components are capable of withstanding or are protected from differential pressures that may occur during the rapid pressure rise to 47 psig in 10 sec. Section 14.3.5.1.1 discusses the analyses that show that the calculated postaccident containment pressures are less severe than this.

Portions of other systems that share functions and become part of this containment cooling system when required are designed to meet the criteria of the containment cooling function. Neither a single active component failure in such systems during the injection phase nor an active/passive failure during the recirculation phase will degrade the heat removal capability of containment cooling (See Section 6.2.3.3).

Where portions of these systems are located outside of containment, the following features are incorporated in the design for operation under postaccident conditions:

1. Means for isolation of any section.
2. Means to detect and control radioactivity leakage into the environs.

6.4.2 System Design And Operation

The flow diagram of the containment air recirculation cooling system is shown in Plant Drawing 9321-4022 [Formerly UFSAR Figure 5.3-1].

Individual system components and their supports meet the requirement for seismic Class I structures and each component is mounted to isolate it from fan vibration.

6.4.2.1 Containment Cooling System Characteristics

The air recirculation system consists of five 20-percent capacity air-handling units, each including motor, fan, cooling coils, moisture separators, roughing filters, duct distribution system, instrumentation, and controls. The units are located on the intermediate floor between the containment wall and the primary compartment shield walls. The air recirculation system has a total heat removal capability of at least 308.5 MBtu/hr under conditions following a loss-of-coolant accident and at a service water temperature of 95°F.

Each fan is designed to supply 65,000 cfm at approximately 22.8-in. static pressure, 271°F, 0.175 lb/ft^3 density. The fans are direct-driven, centrifugal type, and the coils are plate fin-tube type.

Air-operated, tight-closing, 125 psi USAS butterfly valves isolate any inactive air-handling unit from the duct distribution system. Ductwork distributes the cooled air to the various containment compartments and areas. During normal and accident operation, the flow sequence through each air-handling unit is as follows: cooling coils, moisture separators, fan, discharge header.

Roughing filters are installed up-stream of the cooling coils during plant cleanup and any time the reactor is down. These roughing filters are not in place during power operation.

Plant Drawing 9321-4026 [Formerly UFSAR Figure 6.4-3] is an engineering layout drawing of an air-handling unit showing the arrangement of the above components in the unit. Plant Drawing 9321-2502 [Formerly UFSAR Figure 5.1-3] shows the location of the five units on the intermediate floor (elevation 68-ft-0-in.).

6.4.2.1.1 Actuation Provisions

The butterfly valves have only two positions, full open and full closed. These valves are air operated and spring loaded. Upon loss of control signal or control air, the spring actuates the valve to the accident position (fail-safe operation).

Upon either manual or automatic actuation of the safety injection safe-guards sequence, the butterfly valves are tripped to the accident position. Accident position is also the fail-safe position.

Redundant, electrically operated, three-way solenoid valves are used with each butterfly valve to control the instrument air supply (control air). These valves are arranged so that failure of a single solenoid valve to respond to the accident signal will not prevent actuation of the butterfly valve to the accident position (fail-safe operation).

The containment pressure is sensed by eight separate pressure transmitters (six of which are used to generate automatic actuation signals) located outside the containment. Containment pressure is communicated to the transmitters through three ¾-in. stainless steel lines penetrating the containment vessel. A high containment pressure signal automatically actuates the safety injection safeguard sequence (reference is made to Section 6.2.2), which trips the valves to the accident position.

Two high-range containment pressure transmitters have been installed in response to NUREG-0737. These transmitters allow for the indication and recording of pressure up to 150 psig. These signals are recorded on two independent recorders located on the accident assessment

panel in the common Unit 1/Unit 2 central control room. The fans are part of the engineered safety features and either all five, or at least three of five fans will be started after an accident, depending on the availability of emergency power. (Reference is made to Section 8.2.)

Overload protection for the fan motors is provided at the switchgear by overcurrent trip devices in the motor feeder breakers. The breakers can be operated from the control room and can be reclosed from the control room following a motor overload trip.

Flow switches in the ductwork system, operating both normally and after an accident, indicate whether air is circulating in accordance with the design arrangement. Abnormal flow alarms are provided in the control room.

6.4.2.1.2 Flow Distribution and Flow Characteristics

The location of the distribution ductwork outlets, with reference to the location of the air-handling unit return inlets, ensures that the air will be directed to all areas requiring ventilation before returning to the units. The arrangement is shown in Plant Drawing 9321-4022 [Formerly UFSAR Figure 5.3-1].

In addition to ventilating areas inside the periphery of the shield wall, the distribution system also includes two branch ducts located at opposite extremes of the containment wall for ventilating the upper portion of the containment. These ducts are provided with nozzles and extend upward along the containment wall as required to permit the throw of air from nozzles to reach the dome area and ensure that the discharge air will mix with the atmosphere.

The air discharge inside the periphery of the shield wall will circulate and rise above the operating floor through openings around the steam generators where it will mix with air displaced from the dome area. This mixture will return to the air-handling units through floor gratings located at the operating floor directly above each air-handling unit inlet. The temperature of this air will be essentially the ambient existing in the containment vessel. The steam-air mixture from the containment entering the cooling coils during the accident will be at approximately 271^{o}F and have a density of 0.175 lb/ft^3. Part of the water vapor condenses on the cooling coil and is collected by the condensate trays. The condensate from the trays is directed below to the floor tray by means of an individual piping system. The condensate collection and drain system is important to the proper functioning of the cooling coils. The air leaving the unit thus will be saturated at a temperature slightly below approximately 265^{o}F. The fluid will leave the cooling coils and enter the moisture separators at approximately 265^{o}F and saturated (100-percent relative humidity) condition. The purpose of the moisture separators is to remove the entrained moisture.

The fluid will remain in this condition as it flows through the fan, but will pick up some sensible heat from the fan and fan motor before flowing into the distribution header. This sensible heat will increase the dry-bulb temperature slightly above 265^{o}F and will decrease the relative humidity slightly below 100-percent.

With a flow rate of approximately 65,000 cfm from each fan under accident conditions and the containment free volume of 2,610,000-ft^3, the recirculation rate with five fans operating is approximately 7.5 containment volumes per hour.

6.4.2.1.3 [Deleted]

6.4.2.1.4 Cooling Water for the Fan-Cooler Units

The cooling water requirements for all five fan-cooling units during a major loss-of-primary-coolant accident and recovery are supplied by two of the three nuclear service water pumps. The service water system is described in Section 9.6.

The cooling water discharge from the fan and motor cooling coils flows to the discharge canal. As a protective measure a sample of the effluent from these coils is monitored for radioactivity. The sample from the coils is monitored by two redundant radiation monitors. Upon indication of radioactivity in the effluent, each cooler discharge line is monitored individually to locate the defective cooling coil, which when identified would be isolated and operation would continue with the remaining units. The service water system pressure at locations inside the containment in the incident mode system alignment (which the system automatically assumes following a safety injection signal) could be below the containment design pressure of 47 psig. However, since the cooling coils and service water lines are a completely closed system inside the containment, no contaminated leakage is expected into these units.

Local indication of service water discharge temperature from each fan-motor heat exchanger, as well as a fan cooler unit combined outlet header temperature indicator, are provided in the pipe pen outside containment. A fan-motor heat exchanger combined discharge header flow indicator is also located in the pipe pen outside containment. Flow for each fan cooler unit is indicated in the control room. Abnormal flow alarms are provided in the control room. A permanent differential pressure indicator has been installed across the 10-in. inlet and outlet water headers of the fan cooling units for pressure drop measurements.

During normal plant operation, flow through the cooling units can be throttled for containment temperature control purposes by a valve on the common discharge header from the cooling units. Two independent, full flow, isolation valves open automatically to bypass the control valve in the event of a safety injection signal. Both valves fail in the open position upon loss of air pressure and either valve is capable of passing the full flow required for all five fan cooling units.

6.4.2.1.5 Environmental Protection

All system control and instrumentation devices required for containment accident conditions are located to minimize the danger of control loss due to missile damage.

All fan parts, valve shaft and disk seating surfaces, and ducts in contact with the containment fluid are protected against corrosion. The fan motor enclosures, electrical insulation, and bearings are designed for operation during accident conditions.

All of the air-handling units are located on the intermediate floor between the containment building and the primary containment shield wall. The distribution header and service water cooling piping are also located outside the shield wall. This arrangement provides missile protection for all components.

6.4.2.2 Components

6.4.2.2.1 Moisture Separators

The moisture separators are designed to remove a minimum of 99.9-percent of the entrained water in the air-steam mixture entering the air-handling units following a loss-of-coolant accident. With an air entrained moisture content of 0.35 lb H_2O/1000-ft^3, the water flow rate entering the moisture separator section is approximately 23 lb/min and the moisture separator effluent has essentially zero moisture content.

Each bank is designed for horizontal air flow and is composed of 40 elements. Each element or separator is 24-in. x 25-in. x 2-in. (minimum) thick and is mounted in a steel support frame.

A steel drain trough is incorporated for each horizontal tier of separators to collect and remove the water that is recovered from the air steam. Further, the design enables the separators to be removed from the upstream side of the support frame.

In order to prevent the bypass of air around the bank, airtight seals are provided between the floor, walls, plenum, and around the perimeter of each moisture separator. The tight seal is accomplished by gaskets, adhesive, and pressure-sealing tape, all of which can withstand a temperature of 300^o. The thickness of the gaskets is 0.25-in. for the separator elements and 0.375-in. for the perimeter sealing of the support frame; they do not extend into the media area when installed.

The moisture separator elements are of fire-resistant construction and consist of mats of fiberglass pads reinforced with stainless steel wire mesh. Nonstainless steel parts used in the construction are protected against corrosion by painting with one 3-mil shop coat of Carbo Zinc No. 11 or the equivalent. The separator frames are fabricated of type 304L stainless steel with welded joints.

6.4.2.2.2 Roughing Filters

The roughing filters remove the large particles from the air stream before contact is made with the cooling coils. The roughing filters are in operation during plant cleanup and any time the reactor is down. These are efficient for removing large particles and under normal conditions they offer a resistance to air flow of 0.2-in. of water.

As in the case for all components of the air-handling recirculating system, the bank is designed for horizontal air flow. The bank contains 40 filters, each of which has dimensions of 22.875-in. wide x 23.5-in. high x 2-in. thick.

All other details of the mounting frame, sealing and materials of construction, other than the filters themselves, are the same as described for the moisture separators.

The filter is of fire-resistant construction and the medium is fiber glassmat.

6.4.2.2.3 Humidity Detectors

Located just upstream of each fan cooling unit is a humidity detector used to determine the dewpoint in containment. See Section 6.7 for further details.

6.4.2.2.4 [Deleted]

6.4.2.2.5 Fan Motor Units

The five containment cooling fans are of the centrifugal, nonoverloading, direct-drive type.

Each fan can provide a minimum flow rate of 65,000 cfm when operating against the system resistance of approximately 22.8-in. static pressure existing during the accident condition (0.175 lb/ft^3 density, containment pressure of 47 psig, and temperature of 271°F).

The reactor containment fan cooler motors are Westinghouse or Schulz Supplied Re-wound, totally enclosed water-cooled, 350 hp, induction type, three-phase, 60 cycles, 1200 rpm, 440-V with ample insulation margin. Significant motor details are as follows:

1. Insulation

   Class F (NEMA rated total temperature 155°C) Westinghouse Thermalastic or Class H (NEMA rated total temperature 180°C) Schulz Epoxilite. It is impregnated and coated to give a homogeneous insulation system that is highly impervious to moisture. Internal leads and the terminal box-motor interconnection are given special design consideration to ensure that the level of insulation matches or exceeds that of the basic motor system. The Fan Cooler Motors and their lubrication are environmentally qualified for use inside the containment building as documented in their respective EQ files.

2. Heat Exchanger

   An air-to-water heat exchanger is connected to the motor to form an entirely enclosed cooling system. Air movement is through the heat exchanger and is returned to the motor. A flapper type vent relief valve permits incident ambient (increasing containment pressure) to enter the motor air system so the bearings will not be subjected to differential pressure. The cooling coil condensate drain line will enable pressure equalization by the motor heat exchanger as the containment pressure is reduced. Water connections are welded throughout and supply and discharge are common with the containment cooler water system, i.e., supplied from the nuclear service water header. The drain is piped to the containment cooler drain system.

3. Bearings

   The motors are equipped with high-temperature, grease-lubricated ball bearings as would be required if the bearings were subjected to incident ambient temperatures. Continuous bearing monitoring that will alarm in the control room is provided.

4. Conduit (Connection) Box

   The motor leads are brought out of the frame through a seal and into a sealed conduit box.

5. [Deleted]

6.4.2.2.6 Charcoal Filter Housing Pressure Equalization

The charcoal filter housing includes a hole on the external wall of the fan cooler unit. The hole remained after the spring-loaded damper and the charcoal filters were removed and the charcoal filter inlet and outlet dampers were blocked closed during the 2002 refueling outage. When containment pressure rapidly rises as a result of an accident condition, the hole will relieve air into the charcoal filter compartment so as to minimize the negative pressure differential across the walls of the charcoal filter unit. The hole will help minimize positive differential pressure (between containment and the charcoal filter compartment) that can occur as a result of relieving containment pressure during normal plant operation.

6.4.2.2.7 [Deleted]

6.4.2.2.8 Cooling Coils - Original Plant Design

This section describes the cooling coils provided as part of the original plant operation.

The heat removal capability of the cooling coils was 76.32 x 10^6 Btu/hr per air-handling unit at saturation conditions (271°F, 47 psig). The design internal pressure of the coil was 150 psig at 300°F and the coils could withstand an external pressure of 70.5 psig at a temperature of 300°F without damage.

Each recirculating unit consisted of 10 coil units mounted in two banks of five coils high. These banks were located one behind the other for horizontal series air flow and the tubes of the coil were horizontal.

Each coil assembly consisted of the first bank having six rows of coils and the second bank having four rows of coils. Each bank contained four Westinghouse Sturtevant designation WC-36208 (36-in. high by 108-in. long) coil, and one Westinghouse Sturtevant WC-30108 (30-in. high by 108-in. long) coil. This latter coil was at the top and had 16.7-percent fewer tubes. The coils were stacked five high to a bank. The total coil assembly (two banks of coils) was 42-in. wide. There were 10 rows of tubes in the horizontal flow direction and a total of 116 rows of tubes in the vertical direction. Cooling water flow was 1/3 velocity through the first coil bank (six rows of tubes in the horizontal). Tube supports were provided on 15-in. center lines to permit free expansion and contraction of the tubes. Supply and return manifolds at each coil were 4-in. and 3-in. in diameter 90-10 copper-nickel schedule 40 pipe for each bank, respectively. Tubes were 0.625-in. in diameter with 0.035-in. wall thickness. Each "U" tube contained six brazed joints and was expanded to copper plate fins, each .008-in. thick along its straight lengths. The original coils were fabricated of copper plate fins vertically oriented on cupro-nickel tubes. For normal operation, 8 fins/in. were required to remove 2,000,000 Btu/hr using 108 tubes.

The original fan-motor heat exchangers had similar construction. Each tube had 8 passes and 18 brazed joints, was 0.625-in. in diameter and 0.049-in. wall thickness, and had manifold headers 2-in. in diameter.

In 1981, however, these coils were replaced by those of another manufacturer (CVI Corporation). These replacement coils are described in Section 6.4.2.2.9.

Local indication of service water discharge temperature from each fan-motor heat exchanger, as well as a fan cooler unit combined outlet header temperature indicator, are provided in the pipe

pen outside containment. A fan-motor heat exchanger combined discharge header flow indicator is also located in the pipe pen outside containment. Flow for each fan cooler unit is indicated in the control room. Alarms indicating abnormal service water flow and radioactivity are provided in the control room.

Pressure taps to which a ΔP meter can be attached or provided to allow measurements of service water pressure drops through the individual fan cooler units. This instrumentation is intended to indicate fouling in the river water side of the cooling coil.

The coils are provided with drain pans and drain piping to prevent flooding during accident conditions. This condensate is drained to the containment sump. Reference is made to Section 6.7.

Drain flow is measured by a level transmitter located in a standpipe containing a slotted weir. A level alarm and indication is provided in the control room. Actual discharge flow rate is determined by referring the level to a calibration curve for the weir or by use of the weir meter.

If the drainage rate for all five units is nearly the same, it may be concluded that this water is condensate from the containment atmosphere. A particular unit with a high drainage rate with respect to the other units could be an indication of a leak in one of the cooling coils.

6.4.2.2.9 Cooling Coils – Modified

Fan-cooler cooling coils and fan-motor heat exchangers were replaced during the 1980/1981 refueling outage. Two of these coils and two exchangers were replaced during the 1986 refueling outage, and the remaining three were replaced during the 1987 refueling outage.

The modified (1980/1981) design for all cooling coils (fan cooler coils and motor heat exchangers) was changed to 90-10 copper-nickel (CuNi) water box headers with removable cover plates to allow for inservice inspection and maintenance. Water box headers were of bolted construction and consist of tubesheets, spacer sections, coverplates, and flanged elbows with gaskets for all mating surfaces. Perforated baffle plates, also of 90-10 coppernickel (CuNi) were installed at the water box inlet of each fan-cooler/motor cooler unit to provide a more even flow distribution through the cooling coils. All "U" tubes are of 0.049-in. wall thickness, "hair-pin" construction, and are rolled into a tubesheet. All brazed joints (approximately 1800) are eliminated in this modified design. The modified fan cooler cooling coil assemblies have two banks each with five coils. However, both banks now have six rows of tubes for each coil. The upper coils are still proportionally shorter than the rest.

The coils and exchangers installed in 1986-87 are similar, but utilize larger titanium water boxes, AL6X tubes, and fully-captured "O" ring gaskets.

The gasket material provided with the new cooling coils is ethylene propylene diene monomer. Evaluations performed on this type of material have included exposure to radiation levels of 2 x 10^8 rads over one year (postaccident). The results of these tests indicated that this gasket material retained its functional capability (no leaks under hydrostatic test pressure of 225 psig). Evaluation of the integrated radiation dose rate to the cooling coils is estimated to be 7.4 x 10^6 rads based on the guidelines set forth in NUREG-0588 at a power level of 2758 MWt.

6.4.2.2.10 Ducting

The ducts are designed to withstand the sudden release of reactor coolant system energy and energy from associated chemical reactions without failure due to shock or pressure waves by incorporation of dampers along the ducts, which open at slight overpressure of 5 psi or less. The ducts are designed and supported to withstand thermal expansion during an accident.

Where flanged joints are used, joints are provided with gaskets suitable for temperatures to 300°F.

Ducts are constructed of corrosion-resistant material.

6.4.2.2.11 [Deleted]

6.4.2.2.12 Electrical Supply

Details of the normal and emergency power sources are presented in Chapter 8.

Further information on the components of the containment air recirculation cooling system is given in Section 5.3.

6.4.3 Design Evaluation

6.4.3.1 Range of Containment Protection

The containment air recirculation cooling system provides the design heat removal capacity for the containment following a loss-of-coolant accident assuming that the core residual heat is released to the containment as steam. The system accomplishes this by continuously recirculating the air-steam mixture through cooling coils to transfer heat from containment to service water.

The performance of the containment recirculation cooling system for pressure reduction is discussed in Section 14.3.

Any of the following combinations of equipment will provide sufficient heat removal capability to maintain the postaccident containment pressure below the design value assuming that the core residual heat is released to the containment as steam.

1. All five containment cooling fans.

2. Containment Spray alone as follows:
   - Both containment spray pumps operating up to the time the transfer to core recirculation flow begins (during injection phase).
   - One spray pump continuing to take suction from the RWST until the level in the RWST decreases to 2 feet.
   - Both recirculation pumps [Deleted], both residual heat exchangers and both containment recirculation spray headers in operation when the level in the RWST decreases below 2 feet.

3. One containment spray pump and three of the five containment cooling fans (the minimum containment safeguards case discussed in Section 14.3.5).

Following a loss-of-coolant accident both the containment spray system and reactor containment fan cooler system are placed in operation for heat removal, and containment air recirculation. The containment spray system also provides fission product reduction.

During the injection phase of the accident, a minimum of one spray pump and three of five fan coolers are in operation.

The heat removal requirement for the design basis accident are met with these minimum requirements during both the injection and recirculation phase. Section 14.3.5 discusses the pressure transient and heat removal capability of using minimum safeguards.

Since the spray is effective in removal of inorganic iodine during the first 3.4 hr period following the accident, the spray flow could be terminated (subsequent to 3.4 hr) after the containment pressure is reduced and stabilized.
The fan cooling units would continue in operation alone during the long-term recirculation phase during which the containment pressure is continually reduced. In addition, effective recirculation is provided to all parts of the containment. Suction to the fan cooling units is taken from the upper portion of the containment and discharged from the fan coolers through a ring header to various compartments below the operating deck.

6.4.3.2 System Response

The starting sequence of the five containment cooling fans and the related emergency power equipment is discussed in sections 7.2 and 8.2.3.4 and their analyzed performance is discussed in the various Chapter 14 safety analyses.

6.4.3.3 Single-Failure Analysis

A failure analysis has been made on all active components of the system to show that the failure of any single active component will not prevent fulfilling the design function. This analysis is summarized in Table 6.4-1.

The analysis of the loss-of-coolant accident presented in Section 14.3 is consistent with the single-failure analysis.

6.4.3.4 Reliance on Interconnected Systems

The containment air recirculation cooling system is dependent on the operation of the electrical and service water systems. Cooling water to the coils is supplied from the service water system. Three nuclear service water pumps are provided, only two of which are required to operate during the postaccident period.

6.4.3.5 Shared Functions Evaluation

Table 6.4-2 is an evaluation of the main components, which have been discussed previously and a brief description of how each component functions during normal operation and during the accident.

6.4.3.6 Reliability Evaluation of the Fan Cooler Motor

The basic design of the motor and heat exchanger as described herein is such that the incident environment is prevented, in any major sense, from entering the motor winding or when entering in a very limited amount (equalizing motor interior pressure) the incoming atmosphere is directed to the heat exchanger coils where moisture is condensed out. If some quantity of moisture should pass through the coil, the changed motor interior environment would "cleanup" because the interior air continually recirculates through the heat exchanger.

The Fan cooler Motor is an Environmentally qualified motor designed to operate during accidents in the accident environment that it is exposed to. The increase in service water temperature to 95°F will not impact the life expectancy of the motors.

During the lifetime of the plant, these motors perform the normal heat removal service and as such are only loaded to approximately 120-150 hp.

The bearings are designed to perform in the incident ambient temperature conditions. However, it will be noted that the interior bearing housing details are cooled by the heat exchanger. It is expected that bearing temperatures would be 125°C to 140°C under incident conditions.

The insulation has high resistance to moisture and tests performed indicate the insulation system would survive the incident ambient moisture condition without failure. The heat exchanger function of preventing moisture from reaching the winding keeps the winding in much more favorable conditions. In addition, it will be noted that at the time of the postulated incident, the load on the fan motor would increase, internal motor temperature would increase, and would therefore tend to drive any moisture, if present, out of the winding. [Deleted]

Following the incident rise in pressure, it is not expected that there will be significant mixing of the motor (closed system) environment and the containment ambient.

The heat exchanger has been designed using a very conservative fouling factor.

To prove the effectiveness of the heat exchanger in inhibiting large quantities of the steam-air mixture from impinging on the winding and bearings, a full-scale motor of the exact same type as described was subjected to prolonged exposure of accident conditions. The test exposed the motor to a steam-air mixture as well as boric acid and alkaline spray at 80 psig and saturated temperature conditions. Insulation resistance, winding and bearing temperature, relative humidity, voltage and current, as well as heat exchanger water temperature and flow were recorded periodically during the test.

Following the test the motor was disassembled and inspected to ensure that the unit performed as designed. The posttesting inspection showed no degradation of the motor components (Details are reported in Reference 5).

6.4.4 Minimum Operating Conditions

The Technical Specifications establish limiting conditions regarding the operability of the air recirculation units.

6.4.5 Inspections And Testing

6.4.5.1 Inspection

Access is available for visual inspection of the containment fan-cooler and recirculation components including fans, cooling coils, butterfly valves, and ductwork.

6.4.5.2 Component Testing

The butterfly valves on each air-handling unit can be operated periodically to ensure continued operability. The degree of leaktightness of the valves was established by test at the time of installation.

6.4.5.3 System Testing

Each fan cooling unit was tested after installation for proper flow and distribution through the duct distribution system. Four of the fan cooling units are expected to be used during normal operation. Five will only be required for normal operation when the service water inlet temperature is 85°F or higher. The fan not in use can be started from the control room to verify readiness. The associated butterfly valves will be tested only when the fan is running.

6.4.5.4 Operational Sequence Testing

The test described in Section 6.2.5 demonstrates proper transfer and sequencing of the fan motor supplies to the diesel generators in the event of loss of power. A test signal is used to demonstrate proper valve motion and fan starting. This test verifies proper functioning of the vane-switch flow indicators.

REFERENCES FOR SECTION 6.4

1. Deleted

2. Deleted

3. Nuclear Safety Quarterly Report - August, September, October, 1969, Engineered Safety System Studies, BNWL-1266.

4. Deleted

5. C. V. Fields, Fan Cooler Motor Unit Development and Test, WCAP-9003 (Proprietary), Westinghouse Electric Corporation.

6. Schulz Electric Report No. N4446EQFWCD, “Environmental Qualification Report Number N4446EQFWCD for Schulz Electric Company’s Form Wound, Continuous Duty Insulation”.

7. Schulz electric Report No. 45925-1, “Schulz Electric Company’s Environmentally Qualified Insulation System Supplement 1.”

TABLE 6.4-1
Single Failure Analysis – Containment Air Recirculation Cooling System

Component	Malfunction	Comments And Consequences
A. Containment cooling fan	Fails to start	Five provided. Evaluation based on three fans in operation and one containment spray pump operating during the injection phase.
B. Nuclear service water pumps	Fails to start	Three provided. Two required for operation.
C. Automatically operated valves: (Open on automatic safeguards sequence)		
1. Charcoal filter compartment butterfly valves	Fails to open	None, charcoal filters are no longer credited and have been removed.
2. Nuclear service water discharge line isolation valve	Fails to open	Two provided. Operation of one required.

TABLE 6.4-2
Shared Functions Evaluation

Component	Normal Operation Function	Normal Operating Arrangement	Accident Function	Accident Arrangement
Containment cooling fan units (5)	Circulate and cool containment atmosphere	Up to five fan units in service	Circulate and cool containment atmosphere	Five fan units in service
Nuclear service water pump (3)	Supply river cooling water to fan units	Up to three pumps in service	Supply river cooling water to fan units	Three pumps in service

6.4 FIGURES

Figure 6.4-3	Containment Building Air Recirculation Fan Cooler Filter Unit - Plan and Section, Replaced with Plant Drawing 9321-4026
Figure 6.4-4	Deleted

6.5 ISOLATION VALVE SEAL-WATER SYSTEM

6.5.1 Design Bases

The isolation valve seal-water system ensures the effectiveness of those containment isolation valves that are located in lines connected to the reactor coolant system or that could be exposed to the containment atmosphere during any condition, which requires containment isolation, by providing a water seal (and in a few cases a gas seal) at the valves. The system provides a simple and reliable means for injecting seal water between the seats and stem packing of the globe and double-disk types of isolation valves, and into the piping between closed-diaphragm type isolation valves. This system operates to limit the fission product release from the containment. Although the isolation valve seal-water system is designed to automatically initiate during any accident condition requiring containment isolation, the primary function of the system is to limit the fission product release associated with a large break LOCA.

Although no credit is taken for the operation of this system in the calculation of offsite accident doses as discussed in Section 14.3.6, it does provide assurance that the containment leak rate is lower than that assumed in the accident analysis should an accident occur.

Design provisions for inspection and testing of the isolation valve sealwater system are discussed in Section 6.5.5.

See Section 5.2, containment isolation system, for containment isolation diagrams (Figures 5.2-1 through 5.2-29), the tabulation of isolation valve parameters (Table 5.2-1), and a description of the derivation of "phase A" and "phase B" containment isolation signals. Section 5.2.2 discusses the containment isolation valves that are sealed, postaccident, by air from the penetration and weld channel pressurization system.

6.5.2 System Design And Operation

6.5.2.1 System Description

The isolation valve seal-water system flow diagram is shown in Plant Drawing 9321-2746 [Formerly UFSAR Figure 6.5-1]. System operation is initiated either manually or by a "phase A" containment isolation signal. When actuated, the isolation valve seal water system interposes water inside the penetrating line between two isolation points located outside the containment. The resulting water seal blocks the leak-age of the containment through valve seats and stem packing. The water is introduced at a pressure slightly higher (52 psig, minimum) than the containment design pressure of 47 psig. The high-pressure nitrogen supply used to maintain pressure in the seal-water tank does not require an external power source to maintain the required driving pressure. The possibility of leakage from the containment or reactor coolant system past the first isolation point is thus prevented by ensuring that if leakage does exist, it will be from the seal-water system into the containment.

The following lines would be subject to pressures greater than the operating pressure of the seal water portion of the isolation valve seal-water system. These lines are supplied with 250 psig nitrogen for the high pressure portion of the system which exceeds worst case internal post-

post-accident process pressure:

1. Residual heat removal loop inlet line.
2. Residual heat removal loop outlet line.
3. Bypass line from residual heat exchanger outlet to safety injection pumps suction.
4. Residual heat removal pumps mini-flow line.
5. Residual heat removal loop sample line.
6. Recirculation pump discharge sample line.

The isolation valves for those lines can be sealed by nitrogen gas from the high-pressure nitrogen supply of the isolation valve seal-water system. A self-contained pressure regulator operates to maintain the nitrogen injection pressure slightly higher than the maximum expected line pressure. The nitrogen gas injection is manually initiated. The system includes one seal-water tank capable of supplying the total requirements of the system. The tank is normally pressurized from the Nitrogen System through pressure control valves. As a backup, the tank may be pressurized from the system's own supply of high-pressure nitrogen cylinders through pressure control valves. Design pressure of the tank and injection piping [*Note - The injection piping runs and nitrogen supply piping are fabricated using 3/8-in.-OD tubing, which is capable of 2500-psig service.*] is 150 psig, and relief valves are provided to prevent overpressurization of the system if a pressure control valve fails, or if a seal-water injection line communicates with a high-pressure line due to a valve failure in the seal-water line. The design parameters of the seal-water tank are presented in Table 6.5-1.

In lines approximately 3-in. and larger, double disk gate valves are generally used for isolation. A drawing of this valve is presented in Figure 6.5-2. Redundant isolation barriers are provided when the valve is closed. The upstream and downstream disks are forced against their respective seats by the closing action of the valve. Seal-water is injected through the valve bonnet and pressurizes the space between the two valve disks. The seal-water pressure in excess of the potential accident pressure eliminates any outleakage past the first isolation point.

For smaller lines, isolation is generally provided by two globe valves in series with the seal-water injected into the pipe between the valves. The valves are oriented such that the seal water wets the stem packing. When the valves are closed for containment isolation, the first isolation point is the valve plug in the valve closest to containment and the water seal is applied between the valve plug and stem packing. In a number of the smaller lines, isolation is provided by two diaphragm (Saunders Patent) valves in series, with the seal water injected into the pipe between the valves.

The maximum acceptable leakage across both the seat and stem packing of any gate or globe valve is nominally 10 cm^3/hr-in. of nominal pipe diameter. Tests on these valves have indicated that much lower leakage rates can be expected. However, the design of the isolation valve seal-water system is based on the conservative assumption that all isolation valves are leaking at five times the acceptable value, or 50 cm^3/hr-in. of nominal pipe diameter. In addition, a screening criterion of 25 cm^3/min for an individual isolation valve leakage has been established, beyond which point a determination will have to be made whether the valve is still functionally acceptable. Should one of the isolation valves close, but fail to seal, it is conservatively assumed that flow through the failed valve will be limited to approximately 100 times the maximum acceptable leakage valve, or 1000 cm3/hr-in. of nominal pipe diameter, by the resistance of seal-water injection path through the valve.

If a containment isolation valve fails to close, a water seal at the failed valve is ensure by proper slope of the potential line, or a loop seal, or by additional valves on the side of the isolation

valves away from the containment. Excessive seal-water flow to those motor operated isolation valves that could potentially fail to close in response to a containment isolation signal is limited by flow restrictive orifices installed in the seal-water injection lines.

The seal-water tank is sized to provide at least 24-hr supply of seal-water with all of the isolation valves leaking at the design rate of 50 cm3/hr-in., plus the failure of the largest containment isolation valve to seat and leaking at the maximum rate of 1000 m3/hr-in. The seal-water volume required to satisfy these conditions is approximately 144 gal. The 176-gal seal-water tank is provided with low level and low-low level alarms to signal the need for makeup during normal operation and accident conditions, respectively. If all of the isolation valves seat properly, as expected, the tank volume is sufficient for more than 2.5 days of operation at design seal-water flow rates before makeup is required. Two separate sources of makeup water are provided to ensure that an adequate supply of seal-water is available for long-term operation.

For an event resulting in a "phase A" containment isolation signal, but not a "phase B" containment isolation signal, the isolation valve seal-water system is automatically initiated. Flow to the four isolation valves associated with a "phase B" containment isolation signal only will be automatically isolated by solenoid operated valves, and remain isolated unless the containment isolation valves close. This design will prevent seal-water flow to the opened containment isolation valves. In the event that the solenoid operated valves fail to close, excessive flow would be limited to flow restrictive orifices installed in the seal-water injection lines. This design ensures sufficient time for operator action to provide make-up water if long-term system operation is required to limit fission product releases. If long-term isolation valve seal-water system operation is not required, the system may be isolated by operator action.

There are two separate seal-water lines supplying the potentially radioactive and nonradioactive systems, respectively. This prevents the contamination of nonradioactive systems by way of isolation valve seal-water manifolds.

6.5.2.2 <u>Seal-Water Actuation Criteria</u>

Containment isolation (Section 5.2) and seal-water injection are accomplished automatically on phase A isolation actuation for certain penetrating lines requiring early isolation, and manually for others, depending on the status of the system being isolated and the potential for leakage in each case. Generally, the following criteria determine whether the isolation and seal-water injection are automatic or manual.

Automatic containment isolation and automatic seal-water injection are required for lines that could communicate with the containment atmosphere and be void of water following a loss-of-coolant accident. For example, these lines include:

1. Reactor coolant pump cooling water supply and return lines (phase B isolation).

2. Reactor coolant pump seal-water return line (phase B isolation).

3. Excess letdown heat exchanger cooling water supply and return lines (phase A isolation).

4. Chemical and volume control system letdown line (phase A isolation).

5. Reactor coolant system sample lines and sample return line (phase A isolation).

6. Containment vent header (phase A isolation).

7. Reactor coolant drain tank gas analyzer line (phase A isolation).

8. Auxiliary steam supply and condensate return lines (Manual valves qualifying as automatic isolation valves per Section 5.2.2).

9. Service air and city water lines (Manual valves qualifying as automatic isolation valves per Section 5.2.2).

Automatic containment isolation and automatic seal-water injection are also provided for the following lines, which are not connected directly to the reactor coolant system, but terminate inside the containment at certain components. These components can be exposed to the reactor coolant or containment atmosphere as the result of leakage or failure of a related line or component. The isolated lines are not required for postaccident service. For example, these lines include:

1. Pressurizer relief tank gas analyzer line.
2. Pressurizer relief tank makeup line.
3. Safety injection system test line.
4. Reactor coolant drain tank pump discharge line.
5. Steam-generator blowdown/sample lines.
6. Accumulator sample line.
7. Containment sump pump discharge.

Remote manual containment isolation and remote manual seal-water injection are provided for lines that are normally sufficiently filled with water and will remain sufficiently filled following the loss-of-coolant accident and for lines that must remain in service for a time following the accident. The remote manual seal-water injection ensures a long-term seal. For example, these lines include:

1. Reactor coolant pump seal-water supply lines.
2. Chemical and volume control system charging line.
3. Safety injection headers.
4. Containment spray headers.

Manual containment isolation and remote manual nitrogen seal injection are provided for lines that are sufficiently filled with water during the accident, but which are at a pressure higher than that provided by the isolation valve seal-water system. These lines must remain in service for a period of time following the accident or may be placed in service on an intermittent basis following the accident. For example, these lines include:

1. Residual heat removal loop inlet line.
2. Bypass line from residual heat exchanger outlet to safety injection pumps suction.
3. Residual heat removal loop sample line.
4. Recirculation pump discharge sample line.

Seal-water injection is not necessary to ensure the integrity of isolated lines in the following categories:

1. Lines that are connected to non-radioactive systems outside the containment and in which a pressure gradient exists, which opposes leakage from the containment. These include nitrogen supply lines to the pressurizer relief tank, accumulators,

accumulators, and reactor coolant drain tank, the instrument air header and the weld channel pressurization air lines.

2. Lines that do not communicate with the containment or reactor coolant system and are missile protected throughout their length inside containment. These lines are not postulated to be severed or otherwise opened to the containment atmosphere as a result of a loss-of-coolant accident. These include the steam and feedwater headers and the containment ventilation system cooling water supply and return lines.

3. Lines that are designed for postaccident service as part of the engineered safety features, such as the containment sump recirculation line. This line is connected to a closed system outside containment.

4. Special lines, such as the fuel transfer tube, containment purge ducts, and the containment pressure relief line, which are pressurized by the containment penetration and weld channel pressurization system (see Section 6.6). The zone between the two gaskets sealing the blind flange to the inner end of the fuel transfer tube is pressurized to prevent leakage from the containment in the event of an accident. The zone between the two butterfly valves in each containment purge duct is pressurized above incident pressure, while the valves are closed during power operation, as are the two spaces between the three butterfly valves in the containment pressure relief line.

6.5.2.3 Components

All associated components, piping, and structures of the isolation valve seal-water system are designed to seismic Class I criteria. There are no components of this system located inside containment.

The piping and valves for the system including the air-operated valves are designed in accordance with the USAS Code for Pressure Piping (Power Piping Systems), B31.1.

6.5.3 Design Evaluation

The isolation valve seal-water system provides an extremely prompt and reliable method of limiting the fission product release from the containment isolation valves in the event of a loss-of-coolant accident.

The employment of the system during a loss-of-coolant accident, while not considered for the analysis of the consequences of the accident as discussed in Section 14.3.6, provides an additional means of conservatism in ensuring that leakage is minimized. No detrimental effect on any other safeguards system will occur should the seal-water system fail to operate.

6.5.3.1 System Response

Automatic containment isolation will be completed within approximately 10 sec following the generation of the phase A containment isolation signal. This is the estimated closing time of the non-essential containment isolation valves (Section 5.2). Closing times of greater than 10 seconds are permitted on a case by case basis if properly justified by a safety evaluation. Since the isolation valve seal-water system is also actuated by this signal, automatic seal-water injection will be in

will be in effect within this time period, which is less than the 1 min credited in Section 14.3.6.1.

Subsequent generation of the phase B isolation signal on containment high-high pressure (spray actuation signal) will close the essential containment isolation valves with an estimated closing time of 10 sec. Closing times of greater than 10 seconds are permitted on a case by case basis if properly justified by a safety evaluation. Automatic seal-water injection flow will have been initiated in advance of this signal by the phase A signal.

6.5.3.2 Single-Failure Analysis

A single-failure analysis is presented in Table 6.5-2. The analysis shows that the failure of any single active component will not prevent fulfilling the design function of the system.

6.5.3.3 Reliance on Interconnected Systems

Normally the high-pressure nitrogen supply used to maintain pressure in the seal water tank is from the Nitrogen System. However, in the backup mode, when the tank is pressurized from the system's own supply of high-pressure nitrogen cylinders, the isolation valve seal-water system can operate and meet its design function without reliance on any other system. Electric power is not required for system operation, although instrument power is required to provide indication on the Waste Disposal Panel of seal-water tank pressure and level.

6.5.3.4 Shared-Function Evaluation

Table 6.5-3 is an evaluation of the main components discussed previously and a brief description of how each component functions during normal operation and during an accident.

6.5.4 Minimum Operating Conditions

The Technical Specifications establish limiting conditions regarding the operability of the system.

6.5.5 Inspections And Tests

6.5.5.1 Inspections

The system components are all located outside the containment and can be visually inspected at any time.

6.5.5.2 Component Testing

Each automatic isolation valve can be tested for operability at times when the penetrating line is not required for normal service. Lines supplying automatic seal-water injection can be similarly tested.

6.5.5.3 System Testing

Containment isolation valves and the isolation valve seal-water system can be tested periodically to verify capability for reliable operation. The seal-water tank pressure and water level can be observed locally on the Waste Disposal Panel, and these parameters are also monitored continuously via local alarms on the Waste Disposal Panel and a category alarm in the control room.

The system will not be in service during the containment leak rate test.

6.5.5.4 Operational Sequence Testing

The capacity of the system to deliver water at the required rate was verified initially during the pre-operational test period of plant construction and startup. Prior to plant operation, a containment isolation test signal was used to ensure proper sequence of isolation valve closure and seal-water addition.

TABLE 6.5-1
Isolation Valve Seal-Water Tank

Number	1
Total volume, ft^3	23.6
Minimum volume, gal	144
Material	ASTM A-240
Design pressure, psig	150
Design temperature, ^oF	200
Operating pressure, psig	52-62
Operating temperature, ^oF	Ambient
Code	ASME UPV (Section VIII)

TABLE 6.5-2
Single Failure Analysis – Isolation
Valve Seal-Water System

Component	Malfunction	Comments
A. Automatically operated valves (open on phase A containment isolation signal)		
1. Isolation valve for automatic injection headers	Fails to open	Two provided. Operation of one required.
B. Instrumentation		
1. Level transmitter	Fails	Local level indicator at tank also provided.
2. Pressure transmitter	Fails	Local pressure indicator at tank also provided

TABLE 6.5-3
Shared Functions Evaluation

Component	Normal Operating Function	Normal Operating Arrangement	Accident Function	Accident Arrangement
Isolation valve seal-water storage tank (1)	None	Lined up to seal-water injection piping	Source of water for sealing isolation valves	Lined up to seal-water injection piping
N_2 supply bottles N_2 injection piping	None	Lined up to seal-water tank and water tank pressure and N_2 to those valves sealed with isolation valve seal-water system	Source of N_2 to maintain seal injection piping	Lined up to seal-water tank and N_2

6.5 FIGURES

Figure No.	Title
Figure 6.5-1	Isolation Valve Seal - Water System - Flow Diagram, Replaced with Plant Drawing 9321-2746
Figure 6.5-2	Double Disk Isolation Valve With Seal-Water Injection

6.6 CONTAINMENT PENETRATION AND WELD CHANNEL PRESSURIZATION SYSTEM

6.6.1 Design Bases

The containment penetration and weld channel pressurization system provides a means for continuously pressurizing the positive pressure zones incorporated into the containment penetrations and the channels over the welds in the steel inner liner and certain containment isolation valves in the event of a loss-of-coolant accident. Although no credit is taken for system operation in the calculation of offsite accident doses as discussed in Section 14.3.6, it is designed as an engineered safety feature and does provide assurance that the containment leak-rate in the event of an accident is lower than that assumed in the accident analysis.

The system is designed to provide a means for determining the leaktightness of the containment during power operation, thereby reducing the frequency for performing postoperational integrated leakage rate tests.

6.6.2 System Design And Operation

The containment penetration and weld channel pressurization system is shown in Plant Drawing 9321-2726 [Formerly UFSAR Figure 6.6-1]. A regulated supply of clean and dry compressed air

from either of the plant's 100-psig compressed air systems located outside the containment is supplied to all containment penetrations and inner liner weld channels. The system maintains a pressure in excess of containment design pressure continuously during all reactor operations, thereby ensuring that there will be no outleakage of the containment atmosphere through the penetrations and liner welds during an accident. Typical piping and electrical penetrations are described in Sections 5.1 and 5.2.

The primary source of air for this system is the instrument air system (Section 9.6). Two instrument and control air compressors are used, although only one is required to maintain pressurization at the maximum allowable leakage rate of the pressurization system. The station air compressor acts as a backup to the instrument and control air compressors (Section 9.6) for added reliability.

A standby source of gas pressure for the system is provided by the bank of nitrogen cylinders. The associated nitrogen backup system will actuate a low weld channel pressure of approximately 49 psig and deliver nitrogen to the main weld channel pressure regulator. This regulator controls downstream weld channel pressure to containment at approximately 52 psig. Thus, in the event of failure of the normal and backup air supply systems during periods when the system is in operation, the penetration and weld channel pressure requirements will be automatically maintained by the nitrogen supply. This ensures reliable pressurization under both normal and accident conditions.

Containment penetrations and liner weld channels are grouped into four independent zones to simplify the process of locating leaks during operation. Each such zone is served by its own air receiver. In the event that all normal and backup air supplies are lost, each of the four pressurization system zones continues to be supplied with air from its respective air receiver. Each of the air receivers, (see Table 6.6-1), is sized to supply air to its pressurized zone for a period of at least 4 hr, based on a leakage rate of 0.2-percent of the containment free volume per day (0.1-percent leakage into the containment and 0.1-percent leakage to the environment).

If the receivers become exhausted before normal and backup air supplies can be restored, nitrogen from the bank of pressurized cylinders can be supplied to the affected zones. The nitrogen bank is sized to provide a 24-hr supply of gas to the system, again based on a total leakage rate from the pressurization system of 0.2-percent of the containment free volume in 24 hr. There are three nitrogen cylinders in the bank, each approximately 24-in. OD by 20-ft 6½-in long, providing a total volume of 153 cu-ft (51 cu-ft/cylinder). The nitrogen supply will also automatically assume the pressurization gas load in the event an air receiver fails.

A pressure relief valve set at 150 psig (sized for 167 scfm at 10-percent accumulation) protects the system from failure of the pressure-reducing valve in the line to each zone from the bank of nitrogen cylinders. Each zone of piping is also protected by a rupture disk, designed to open at 175 psig. In addition, the electric penetration assemblies (Zone 1) are protected by a pressure relief valve set at 70 psig. Pressure control valves, isolation valves, and check valves are located outside of the containment for ease of inspection and maintenance. The failure of any of these components does not lead to a loss of pressure in the system since backup systems automatically augment the normal air supply.

The line to each of the four pressurized zones is equipped with a critical pressure drop orifice (installed in the pressure control valve body) to ensure that air consumption will be within the capacity of the system. High air consumption in one zone cannot affect the operation of the other zones under any circumstances.

Means for ensuring that all the weld channels and penetrations are pressurized is provided by flow-through test lines connected to the pressurized weld channel zones and penetrations at points as far away from the supply points as possible. The pressurization of the zone can be verified by closing off the air supply line and opening the flow-through test line valve to observe the escape of the pressurizing medium.

Certain portions of the Weld Channel Pressurization System may be disconnected if that portion has become inoperable and repairs to that portion of the system have been determined not to be practicable. Currently, sections W-10, W-11, D-2, B-2, B-5, B-6, and connection to penetration MP-"O-O" are disconnected. The method of disconnection is selected so as not to interfere with the ability to pressurize those portions of weld channel to containment atmospheric conditions during an integrated leak rate test (ILRT) in accordance with 10 CFR 50 Appendix J.

6.6.2.1 Pressure Indication

The following instrumentation is provided as described below to ensure that the station operators are aware at all times that all penetrations and liner weld seam channels are pressurized.

The following pressurized zones are equipped with local pressure gauges, mounted outside the containment for ready accessibility and available for regular reading.

1. Each piping penetration.
2. Each electrical penetration.
3. The spaces between the two isolation (butterfly) valves in the purge supply and exhaust ducts.
4. The two spaces between the three isolation (butterfly) valves in the containment pressure relief line.
5. The double-gasketed space on the outside hatch of each of the two personnel air locks.

The pressurized zones located entirely inside the containment and those zones located in inaccessible areas are equipped to actuate pressure switches to provide remote low-pressure alarms in the central control room. Examples of the zones so equipped are as follows:

1. Each liner seam weld channel, except for the disconnected zones listed in Section 6.6.2.
2. The double-gasketed space on each inside hatch of the personnel airlocks.
3. The double-gasketed space on the equipment door flange.
4. The pressurized zones in the spent fuel transfer tube.
5. Shroud rings over penetration-to-containment liner weld piping and electrical penetrations.

The control room low pressure alarm switches and the nitrogen backup actuation switches are set above incident pressure and below the pressure setting of the main weld channel pressure regulators. Should pressure in any of these zones fall below the alarm pressure switch setpoint, a light and an alarm in the control room will be activated. Each penetration and each section of liner weld joint channel so alarmed will be represented by a separate light and identified.

6.6.2.2 Flow Indication

The flow to each zone is measured by two meters mounted in series. In addition to indication, low and high flow alarms are provided in the control room.

6.6.2.3 Personnel Air Lock Interlock

Continuous pressurization of air lock door double-gasketed barriers and the protection of the pressurization header against air loss are ensured by a set of interlocks. One interlock on each airlock door prevents the opening of the door until the pressurization line is isolated and pressure in the double-gasketed closure is relieved to atmosphere. This prevents excessive leakage from the pressurization system. The pressurization line to this zone is also equipped with a restricting orifice to ensure that air consumption, even upon failure of the interlock, will be within the capacity of the pressurization system, and will not result in a loss of pressure in other zones connected to the same pressurization header. Another set of interlocks prevents opening of one air lock door until the double-gasketed zone on the other door is repressurized.

6.6.2.4 Containment Purge Line Interlock

The containment ventilation purge penetration butterfly valves inside containment are interlocked to prevent their opening until the pressurization line to each purge duct pressurization zone has been isolated and the space between been depressurized. The isolation of the pressurization line to each purge duct-pressurized zone is accomplished from the fan room. Alarm lights, prominently displayed on a panel indicating the isolation status of the containment, remain lit identifying an open purge duct isolation valve or a low pressurization zone pressure. Restricting orifices are installed in each pressurization line to the ventilation purge ducts to ensure that air consumption, even on the failure of an interlock, will not result in a loss of pressure to the other zones connected to the same pressurization header.

The containment pressure relief line isolation valve inside containment (PCV-1190), and the pressurized space formed between it and the next butterfly valve in series, are provided with an interlock to prevent the opening of PCV-1190 until the adjacent intervalve space has been depressurized. By procedure the outside containment pressure relief line isolation valves PCV-1191 and PCV-1192 are opened from the central control room after isolation valve PCV-1190 is verified open. A time delay allows the pressurized space between the two outside isolation valves to vent air through the associated solenoid valve to atmosphere before these valves will open. The pressurization lines to these spaces are also equipped with flow restricting orifices; alarm lights in the control room identify open valves or low intervalve space pressure.

6.6.2.5 Containment Inleakage

With a continuous inleakage to the containment from the penetration and liner weld joint channel pressurization system of 0.1-percent of the containment volume per day, the calculated time for the containment pressure to rise by 1 psi is approximately 14 days and therefore is not considered to be an operating or safety problem. From the standpoint of allowable pressure, a much greater inleakage would be permitted. With the ability to limit the activity of the air in the containment during normal operation with the use of the two containment auxiliary charcoal filter units, each complete with roughing filter, HEPA filters, and charcoal filters (Section 5.3), containment overpressure can be relieved as required through the pressure relief duct and

exhaust fan, passing up the discharge duct along with the exhaust air from the primary auxiliary building.

6.6.2.6 Components

All associated components, piping, and structures of the containment penetration and weld channel pressurization system are designed to seismic Class I criteria.

The piping and valves for the system are designed in accordance with the USAS Code for Pressure Piping (Power Piping Systems), B31.1.

For a description of the instrument and control air compressors and the plant air compressors, refer to the discussion on the service air system, Section 9.6.

The three nitrogen cylinders used are designed in accordance with Section VIII (Unfired Pressure Vessels) of the ASME Boiler and Pressure Vessel Code for 2200-psig maximum pressure and contain a total of 22,000 scf of nitrogen.

6.6.3 Design Evaluation

The employment of this system following a loss-of-coolant accident, while not considered in the analysis of the consequences of the accident as discussed in Section 14.3.6, provides an additional means for ensuring that leakage is minimized if not altogether eliminated. No detrimental effect on any other safety features system will be felt should the pressurization system fail to operate.

6.6.3.1 System Response

Since the containment penetration and weld channel pressurization system is continuously pressurized above the containment design pressure during all reactor operations, there is no response time required for the system to operate.

6.6.3.2 Single Failure Analysis

A single failure analysis is presented in Table 6.6-2. The analysis shows that the failure of any single active component will not prevent fulfilling the design function of the system.

6.6.3.3 Reliance on Interconnected Systems

The containment penetration and weld channel pressurization system can operate and meet its design function without reliance on any other system, except as limited by air compressor availability following the depletion of all reserves in the system's air receivers and backup nitrogen cylinders. Electric power is not necessary for the operation of the system, although instrument power is required in order to provide indications in the control room of system operation.

6.6.3.4 Shared Functions Evaluation

Table 6.6-3 is an evaluation of the main components discussed previously and a brief description of how each component functions during normal operation and during an accident.

6.6.4 Minimum Operating Conditions

The Technical Specifications establish limiting conditions regarding the operability of the system.

6.6.5 Inspections And Tests

6.6.5.1 Inspections

The system components located outside the containment can be visually inspected at any time. Components inside the containment can be inspected during shutdown. All pressurized zones have provisions for either local pressure indication outside the containment or remote low-pressure alarms in the control room, except for low pressure alarm lights associated with the disconnected weld channel zones listed in Section 6.6.2.

6.6.5.2 Testing

Since the system is in operation continuously during all reactor operations to maintain the penetrations and liner weld channels pressurized above containment design pressure, no special testing of system operation or components is necessary.

Should one zone indicate a leak during operation, the specific penetration or weld channel containing the leak can be identified by isolating the individual air supply line to each component in the zone and injecting leak test gas through a capped tube connection installed in each line.

Total leakage from penetrations and weld channels is measured by summing the recorded flows in each of the four pressurization zones. A leak would be expected to build up slowly and would therefore be noted before design leakage limits are exceeded. Thus, remedial action can be taken before the limit is reached.

In order to provide facility for containment testing in accordance with Technical Specification 4.4 and 10 CFR 50 Appendix J, test connections are provided in each of the zones.

Flow Instrumentation is installed in the piping to the personnel air locks during testing to provide measurement of the airlock leakage rate independent of the other components served by the same zone.

The makeup air flow to the penetrations and liner weld joint channels during normal operation is recognized to be only an indication of the potential leakage from the containment. However, it does indicate the leakage from the pressurization system, and the degree of accuracy will be increased when correlated with the results of the full-scale containment leak rate tests. The criteria for the selection of operating limits for air consumption of the pressurization system are based upon the design integrated containment leak rate and upon the maintenance of suitable reserve air supplies in the static reserves consisting of the air receivers and nitrogen cylinders. A summary of these operating limits is as follows:

1. A baseline air consumption rate was established for each of the four pressurization headers at the time of successful completion of the pre-operational integrated containment leakage rate tests. Unexplained increases from this consumption rate shall be considered as reason for concern and normal practice will require routine investigation and location of the point of leakage.

2. The upper limit for long-term uncorrected air consumption for the pressurization system shall be 0.2-percent of the containment volume per day (sum of four zones) at the system operating pressure, contingent on the following:

   a. Pressure in all pressurization zones is maintained above incident pressure.
   b. Air supply is maintained from the compressed air systems with compressors running.
   c. The full complement of standby nitrogen cylinders (three) is charged. This is consistent with maintenance of a 24-hr supply.

A variable area flow sensing device is located in each of the headers supplying makeup air to the four pressurization zones. The flow sensing device for each zone has two flow transmitters (high and low). Signal output from each of the two flow transmitters is applied to an integrator which has an output to a flow indicator for each zone located in the control room. Output from each of the four integrators is also applied to a single summing amplifier that drives a single total-flow recorder which is also located in the control room. Two high-flow alarms (one short term and one long term) are also derived in the recording channel to alert the operator in the control room.

TABLE 6.6-1
Containment Penetration And Weld Channel
Pressurization Air Receivers

Number	4
Volume (each), ft^3	360
Material	ASTM A-285-C
Design pressure, psig	140
Design temperature, $^\circ$F	200
Operating pressure, psig	100
Operating temperature, $^\circ$F	100
Code	ASME UPV (Section VIII)

TABLE 6.6-2
Single Failure Analysis Containment Penetration
And Weld Channel Pressurization System

Component	Malfunction	Comments
Instrument and control air compressor	Fails to maintain pressure	One of two instrument and control air compressors required to operate.
Pressure-reducing valve for each zone	Fails to maintain pressure	On valve failure, flow is limited to acceptable value (75 scfm) by the critical pressure drop orifice. Under

Under low-flow conditions, overpressurization of system downstream of valve is prevented by a rupture disk.

TABLE 6.6-3
Shared Functions Evaluation

Component	Normal Operating Function	Normal Operating Arrangement	Accident Function	Accident Arrangement
Instrument and control air compressors (2)	Supply air to plant's instruments and controls and to penetrations and weld channels	2 air compressors in operation	Supply air to penetrations and weld channels	1 air compressor$_1$ in operation
Station air compressors (1)	Supply air to station air headers	1 air compressor in operation	Supply air to penetrations and weld channels	1 air compressor$_1$ in operation
N_2 cylinders (3)	Backup source of N_2 to maintain penetration and channel pressure	Lined up to penetration and weld channel pressurization system	Backup source of N_2 to maintain penetration and weld channel pressure	Lined up to penetration and weld channel pressurization system
Air receivers (1) and dryers (3)	Primary source of air for penetrations and weld channels	Lined up to penetrations and weld channel pressurization	Primary source of air for penetrations and weld channels	Lined up to penetration and weld channel pressurization system

system

Notes:
1. Assuming offsite power available.

6.6 FIGURES

Figure No.	Title
Figure 6.6-1	Weld Channel and Penetration Pressurization System - Flow Diagram, Replaced with Plant Drawing 9321-2726

6.7 LEAKAGE DETECTION AND PROVISIONS FOR THE PRIMARY AND AUXILIARY COOLANT LOOPS

6.7.1 Leakage Detection Systems

The leakage detection systems reveal the presence of significant leakage from the primary and auxiliary coolant loops.

6.7.1.1 Design Bases

6.7.1.1.1 Monitoring Reactor Coolant Leakage

Criterion: Means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary. (GDC 16)

Positive indications in the control room of the leakage of coolant from the reactor coolant system to the containment are provided by equipment that permits continuous monitoring of containment air activity and humidity and of runoff from the condensate collecting pans under the cooling coils of the containment air recirculation units. This equipment provides an indication of normal background, which is indicative of a basic level of leakage from primary systems and components. Any increase in the observed parameters is an indication of change within the containment, and the equipment provided is capable of monitoring this change. The basic design criterion is the detection of deviations from normal containment environmental conditions, including air particulate activity, radiogas activity, humidity, condensate runoff, and in addition, in the case of gross leakage, the liquid inventory in the process systems and containment sump.

6.7.1.1.2 Monitoring Radioactivity Releases

Criterion: Means shall be provided for monitoring the containment atmosphere and the facility effluent discharge paths for radioactivity released from normal operations, from anticipated transients, and from accident conditions. An environmental monitoring program shall be maintained to confirm that radioactivity releases to the environs of the plant have not been excessive. (GDC 17)

The containment atmosphere, the ventilation exhaust from the residual heat removal pump compartments, the containment fan cooler service water discharge, the component cooling loop liquid, the liquid phase of the secondary side of the steam generator, and the condenser air

ejector exhaust are monitored for radioactivity concentration during normal operation, anticipated transients, and accident conditions.

6.7.1.1.3 Principles of Design

The original design of the RHR and HHSI Pump seals incorporated a disaster bushing that would limit the flow to 50 GPM if the seal faces were severely damaged. For Generic Letter 2004-02 compliance, an analysis determined the wear of these disaster bushings if debris laden fluid passed through a failed seal. The potentially abrasive nature of the fluid can wear non-metallic disaster bushings over time, whereby the flow out past the damaged seal could eventually exceed 50 GPM. However, this effect is not immediate and as before, actions would be taken to isolate the pump before the 50 GPM flow rate is reached. The Chesterton seal, an alternate type to the original seal, was tested to demonstrate that severely damaged seal faces would result in a flow rate of less than 50 GPM past the seal. Both the original seal designs and later Chesterton model seals are acceptable and may be used in the HHSI and RHR pumps.

The principles for the design of the leakage detection systems can be summarized as follows:

1. Increased leakage could occur as the result of a failure of pump seals, valve packing glands, flange gaskets, or instrument connections. The maximum leakage rate calculated for these types of failures is 50 gpm, which would be the anticipated flow rate of water through the pump seal if the entire seal were wiped out and the area between the shaft and housing were completely open.

2. The leakage detection systems should not produce spurious annunciation from normal expected leakage rates, but should reliably annunciate increasing leakage.

3. Increasing leakage rate is to be annunciated in the control room. Operator action is required to isolate the leak in the offending system.

6.7.1.2 <u>Systems Design and Operation</u>

For Class 1 systems located outside the containment, leakage is determined by one or more of the following methods:

1. For systems containing radioactive fluids, leakage to the atmosphere would result in an increase in local atmospheric activity levels and would be detected by either the plant vent monitors or by one of the area radiation monitors. Similarly, leakage to other systems that do not normally contain radioactive fluids would result in an increase in the activity level in that system.

2. For closed systems such as the component cooling system, leakage would result in a reduction in fluid inventory.

3. All leakage would collect in specific areas of the building for subsequent handling by the building drainage systems, e.g., leakage in the vicinity of the residual heat removal pumps would collect in the sumps provided, and would result in the operation, or increased operation, of the associated sump pumps and increased inventory in the liquid waste processing system.

Details of how these methods are used to detect leakage from Class 1 systems other than the reactor coolant system are given in the following sections and summarized in Table 6.7-1.

The Class 1 fluid systems for which no special leak detection outside containment is provided include the following:

1. Residual heat removal.
2. Component cooling.
3. Service water.
4. Auxiliary feedwater.
5. Waste disposal.

Various methods are used to detect leakage from either the primary loop or the auxiliary loops. Although described to some extent under each system description, all methods are included here for completeness.

6.7.1.2.1 Reactor Coolant System

In considering potential leakage from the reactor coolant system containing primary coolant at high pressure, four categories should be considered:

1. Leakage to the reactor coolant drain tank.
2. Leakage to the pressurizer relief tank.
3. Leakage to the containment environment.
4. Leakage to the interconnecting systems.

For clarity, each of these paths are discussed in turn.

6.7.1.2.1.1 Paths Directed to the Reactor Coolant Drain Tank

The routes directed to the reactor coolant drain tank may be summarized as follow:

1. Reactor coolant system loop drains.
2. Accumulator drains.
3. Auxiliary system equipment drains.
4. Excess letdown.
5. Valve leakoffs.
6. Reactor coolant pump seal leakage.
7. Reactor flange leakoff.

Of these paths, (1) through (4) do not present a leakage load on the reactor coolant drain tank during normal operation; leakage from the high-pressure systems is not expected because of the use of double isolation valves. Leakage paths (5) through (7), above, are evaluated as follows:

Valve Leakoffs

<u>Source</u> - There are 19 valves in the containment provided with leakoff connections. Of these valves, only four valves in the safety injection system will normally have their valve stem packing subjected to pressure. These are:

894A,B,C,D Accumulator isolation valves that are normally open are provided with backseats. Leakage would only be of borated radioactive water.

Estimated Leakage – Total leakage of reactor coolant fluid during normal power operation is conservatively estimated to be 8 cm^3/hr per the following:

For valves 894A, B, C, and D, leakages are assumed to be 2 cm^3/hr per valve, a total leakage of 8 cm^3/hr is assumed.

Indication to Operator - The operator is alerted to abnormal conditions by an increase of the drain tank water temperature and eventually the change in tank level. Drain tank temperature, pressure, and level are continuously indicated on the "waste disposal/boron recycle" panel in the auxiliary building; high pressure, high temperature, high level, and low level are annunciated on this panel. Any alarm on this panel causes the annunciation of a single window on the main control board in the central control room.

Reactor Coolant Pump Seals

Source - Charging flow is directed to the reactor coolant pumps via a seal-water injection filter and enters each pump at a point between the labyrinth seals and the No. 1 face seal. Here the flow splits and a portion (normally about 5 gpm) enters the reactor coolant system via the labyrinth seals and thermal barrier cooler cavity. The remainder of the flow (normally about 3 gpm) flows up the pump shaft (cooling the lower bearing) and leaves the pump via the No. 1 seal where its pressure is reduced to about 25 to 30 psig and its temperature is increased from 130^{o}F to about 136^{o}F. The labyrinth flows (20 gpm total for four reactor coolant pumps) are removed from the system as a portion of the letdown flow. The No. 1 seal discharges (12 gpm total for four reactor coolant pumps) flow to a common manifold and then via a filter (seal-water filter) through the seal-water heat exchanger (where the temperature is reduced to about 130^{o}F) to the volume control tank.

The leakoff system between the No. 2 and No. 3 seals is considered to be part of the reactor coolant system. The leakoff system collects leakage passed by the No. 2 seal, and provides a constant backpressure on the No. 2 seal and constant pressure on the No. 3 seal. A standpipe is provided to give a constant backpressure during normal operation. The first outlet from the standpipe is orificed to permit normal No. 2 seal leakage to flow to the reactor coolant drain tank; excessive No. 2 seal leakage will result in a rise in the standpipe level and eventual overflow to the reactor coolant drain tank via a second overflow connection.

Leakage – The normal No. 2 seal leakage is anticipated to be approximately 3 gph per pump. This is the value specified in the reactor coolant pump equipment specification.

Indication to Operator - Level instrumentation on the standpipes is provided to alert the operator to abnormal conditions. The standpipe consists of a pipe with an orificed overflow above the midpoint, a normally closed drain (for service) at the bottom, and a free-flowing overflow at the top. Normal No. 2 seal leakage will flow freely out the orificed overflow. Excessive leakage will "back up" in the standpipe until it overflows at the top. A level switch in the upper standpipe actuates an annunciator indicating excessive flow. A level switch in the lower standpipe causes the annunciation of the opposite condition, which could result in undesirable dry operation of the No. 3 seal.

Reactor Vessel Flange Leakoff

Source - The reactor vessel flange and head are sealed by two metallic O-rings. To facilitate leakage detection, a leakoff connection is placed between the two O-rings and a leakoff connection is placed beyond the outer O-ring. Piping and associated valving is provided to direct any leakage to the reactor coolant drain tank.

Leakage - During normal operation, it is anticipated that the leakage will be negligible since it is specified in the reactor vessel equipment specification that there is to be zero leakage past the outer O-ring under normal operating and transient conditions.

Indication to Operator - A temperature detector will indicate leakage by a high temperature alarm. The operator is further alerted by the associated increase in drain tank water temperature and eventually the change in tank level.

6.7.1.2.1.2 Paths Directed to the Pressurizer Relief Tank

These leakage routes are evaluated below.

Source - The pressurizer relief tank condenses and cools the discharge from the pressurizer safety and relief valves. Discharge from small relief valves located inside the containment is also piped to the relief tank. During normal operation, leakage could possibly occur from either the pressurizer safety valves, pressurizer relief valves, or the chemical and volume control system letdown station relief valve.

Leakage - During normal operation, the leakage to the pressurizer relief tank is expected to be negligible since the valves are designed for essentially zero leakage at the normal system operating pressure as specified in the respective valve equipment specifications.

Indication to Operator - For these valves, temperature detectors are provided as an indication of possible leakage. In addition, each pressurizer safety valve is provided with an acoustic monitor in the discharge piping to alert the operator to possible leakage.

The rate of increase of the water temperature in the pressurizer relief tank and the level change will indicate to the operator the magnitude of the leakage. In the event of excessive leakage into an interconnecting system causing lifting of the local relief valves, the operator would again be alerted to the situation by a rising tank water temperature (refer to Section 6.7.1.2.1.4). The acoustic monitors provide a gross indication of safety valve leakage via an alarm in the central control room.

6.7.1.2.1.3 Releases to the Containment Environment

Leakage to the containment environment is discussed as follows.

Source - The main contributors of leakage to the containment environment may be listed as follows:

1. Valve stem leakages

   As previously discussed, the modulating valves within the containment are provided with leakoff connections, which in turn are either piped to the reactor coolant drain tank or are capped. Of the remaining valves that serve lines and

components containing reactor coolant, only two are not normally fully open or fully closed; i.e., the continuous spray bypass needle valves around the main spray valves. The remaining valves are of the backseated type, which prevent the valve stem packings from being subjected to high pressures when in the open position.

2. Reactor coolant pump No. 3 seal leakage

   A small continuous leakage is anticipated past the No. 3 seal to the containment environment; this fluid will be seal injection water. The No.3 seal leakoff is diverted to the local open drains and is thus released to the containment environment.

3. Weld flanges

   The welded joints throughout the system are subjected to extensive nondestructive testing; leakage through metal surfaces and welded joints is very unlikely.

4. Flange joints

   There are a number of flanged joints in the system, all of which are subjected to leak testing before power operation. Experience has shown that hydrostatic testing is successful in locating leaks in a pressure-containing system.

   Methods of leak location that can be used during plant shutdown include visual observation for escaping steam or water or for the presence of boric acid crystals near the leak. The boric acid crystals would be present near the leak as a result of the evaporation process of the leaking fluid.

Leakages - The main contributors to leakage to the containment environment are considered to be items 1 and 2 above; experience with operating reactors has shown that following the normal preoperational testing, leakage from these sources are negligible.

1. Valves stems

   Normally open valves have backseats that limit leakage to less than 1 cm^3/hr-in. of stem diameter assuming no credit for packing in the valve. Normally closed globe valves are installed with recirculation flow under the seat to prevent stem leakage from the more radioactive fluid side of the seat. On the basis of these pessimistic assumptions, the leakage from valves is estimated to be approximately 50 cm^3/hr.

2. Reactor coolant Pump No. 3 Seal Leakage

   The fluid will be seal injection water and is anticipated to be approximately 100 cm^3/hr per pump. This is the value specified in the reactor coolant pump equipment specification.

Conclusion - On the basis of the above, the analysis of the situation indicates a total leak rate to the containment environment of about 450 cm^3/hr. For design purposes, 50 lb/day (i.e., 1000 cm^3/hr) is assumed. Allowable reactor coolant leakage rate and leakage rate to the containment free volume are specified in the Technical Specifications.

6.7.1.2.1.4 Leakage to Interconnecting Systems

Leakage paths to interconnecting systems are evaluated below.

System	Discussion
Chemical and volume control system	This is a normally operating interconnecting system with redundancy for isolating purposes if required.
Sampling system	In the event of sample valves failing to close or seat, adequate redundancy is provided by containment isolation valves; the piping between the sets of valves is designed for reactor coolant system pressure.
Residual heat removal system hot leg connection	Two isolation valves are provided; in the unlikely event of leakage past the two valves, interconnecting piping is provided to enable pressure relief via the residual heat removal system loop relief valve to the pressurizer relief tank.
Residual heat removal system cold leg	In the unlikely event of leakage past the accumulator check valves, residual heat removal system loop check valves and the motorized isolation valves, pressure relief will take place via the residual heat removal system loop relief valves to the pressurizer relief tank.
Safety injection system high-head pump injection lines	In the event of leakage past the check valves and motorized gate valve in any one of the four cold leg injection lines or check valves and motorized gate valve in either of the two hot-leg injection lines, pressure relief will take place to the pressurizer relief tank via the relief valve in the safety injection system test line.
Safety injection system accumulator connections	Provisions have been made to check the leak-tightness of the accumulator check valves. Leakage past these valves is discussed in Section 6.2.

Leakage of primary fluid to the secondary system via the steam-generator primary/secondary boundary would result in an increase of activity level in the secondary system and would be detected by the condenser air ejector gas monitor or by the steam-generator liquid sample monitor. (Refer to Section 11.2.)

During normal operation and anticipated reactor transients, the following methods are employed to detect leakage from the reactor coolant system.

6.7.1.2.2 Containment Air Particulate Monitor

This channel takes continuous air samples from the containment atmosphere and measures the air particulate beta radioactivity. The samples, drawn outside the containment, are in a closed, sealed system and are monitored by a scintillation detector assembly. This assembly collects particulate matter greater than 1 micron in size on its moving filter paper surface, which is viewed

viewed by a scintillation photomultiplier combination. After passing through the gas monitor, the samples are returned to the containment.

The filter paper has a 25-day minimum supply at normal speed. The filter paper mechanism, an electromagnetic assembly, which controls the filter paper movement, is provided as an integral part of the detector unit.

The detector assembly is in a completely closed housing. The detector output is amplified by a preamplifier, processed and transmitted to the radiation monitoring system console, safety related display console and a recorder in the control room. Lead shielding is provided to reduce interference with the detector's sensitivity caused by background radiation.

The activity is indicated on digital displays and recorded by a multipoint recorder. High-activity alarm indications are displayed on the control board annunciator and the safety related display console. Local alarms provide operational status of supporting equipment such as pumps, motors, and flow and pressure controllers.

The containment air particulate monitor is sensitive to low rates. The rates of reactor coolant leakage to which the instrument is sensitive are 0.1 gpm to greater than 10 gpm, assuming corrosion product activity and no fuel cladding leakage. Under these conditions, an increase in reactor coolant system leakage of 1 gpm is detectable within 1 hour after it occurs.

The sensitivity of the air particulate monitors to an increase in reactor coolant leak rate is dependent upon the magnitude of the normal baseline leakage into the containment. The sensitivity is greatest where baseline leakage is low as has been demonstrated by the experience of Indian Point Unit 1 (Appendix 6B), Yankee Rowe, and Dresden Unit 1. Assuming a low background of containment air particulate radioactivity, if we assume a reactor coolant corrosion product radioactivity (Fe, Mn, Co, Cr) of approximately 0.4 $\mu Ci/cm^3$ (a value consistent with little or no fuel cladding leakage), and complete dispersion of the leaking radioactive solids into the containment air, the air particulate monitors would be capable of detecting an increase in coolant leakage rate as small as approximately 0.1 gpm (400 cm^3/min) within 20 min after it occurs. If only 10-percent of the particulate activity were a assumed to be dispersed in the air, leakage rate increases of about 1.0 gpm (4000 cm^3/min) would be detectable within the same time period.

For cases where baseline reactor coolant leakage falls within the detectable limits of the air particulate monitor, the instruments can be adjusted to alarm on leakage increases of from 2 to 5 times the baseline volume.

The containment air particulate monitor together with the other radiation monitors mentioned in this section are further described in Section 11.2.

6.7.1.2.3 Containment Radioactive Gas Monitor

This channel measures the gaseous gamma radioactivity in the containment by taking the continuous air samples from the containment atmosphere, after they pass through the air particulate monitor, and drawing the samples through a closed, sealed system to the gas monitor assembly.

The samples are constantly drawn through the fixed, shielded volumes and are viewed by a scintillation detector. The samples are then returned to the containment.

The detector is in a completely enclosed housing containing a gamma-sensitive scintillation detector mounted in a constant gas volume container. Lead shielding is provided to reduce interference with the detector's sensitivity caused by background radiation.

The detector output is amplified by a preamplifier, processed and transmitted to the radiation monitoring system console, the safety related display console and a recorder in the control room and indicated on digital displays. High-activity alarm indications are displayed on the control board annunciator and the safety related display console. Local alarms annunciate the supporting equipment's operational status.

The containment radioactive gas monitor is inherently less sensitive (threshold at 10^{-6} $\mu Ci/cm^3$) than the containment air particulate monitor and would function in the event that significant reactor coolant gaseous activity exists from fuel cladding defects. The measuring range is 10^{-6} to 10^{-3} $\mu Ci/cm^3$.

The containment air particulate and radioactive gas monitors have assemblies that are common to both channels. They are described as follows:

1. Each flow assembly includes a pump unit and selector valves that provide a representative sample (or a "clean" sample) to the detector.

2. The pump unit consists of:

   a. A pump to obtain the air sample.
   b. A flowmeter to indicate the flow rate.
   c. A flow control valve to provide flow adjustment.
   d. A flow alarm assembly to provide a low flow alarm signal.

3. Selector valves are used to direct the desired sample to the detector for monitoring and to vent flow when the channel is in maintenance or "purging" condition.

4. A pressure sensor is used to protect the system from high pressure. This unit automatically closes an inlet and outlet valve upon a high-pressure condition.

5. Purging is accomplished with a valve control arrangement whereby the normal sample flow is blocked and the detector purged with a "clean" sample.

6. The safety related display console in the control room permits remote operation of the local radiation monitor skid assembly. By operating a pushbutton on the console to start the sample pump, the containment sample can be monitored.

7. A sample flowmeter is calibrated linearly (from 0 to 56.6 standard liters per minute).

In addition to a common CCR High Rad/Trouble Annunciator, the following alarm lights are provided locally:

1. NO CPM - No detector signal in the counting circuit.
2. LOW FLOW - Flowrate drops below low flow setpoint.

3. T. TEAR - Particulate tape filter tears or the supply reel is run to empty.
4. LO TEMP - Monitor sample temperature is less than 10°F above containment air temperature. This alarm has been disabled.
5. WARN - Particulate or gas warn setpoint exceeded.
6. ALARM - Particulate or gas alarm setpoint exceeded.

6.7.1.2.4 Humidity Detectors

The humidity detection instrumentation offers a means for the detection of leakage into the containment. The instrumentation is sensitive to vapor originating from all sources within containment, including the reactor coolant and steam and feedwater systems. Plots of changes in dewpoint of the containment atmosphere will be sensitive to incremental increases of water leakage to the containment atmosphere approximately 1.0 gpm/°F of dewpoint temperature increase (this sensitivity will vary with cooling water temperature, containment air temperature, and air recirculation rate). These detectors are located just upstream of each fan cooler unit.

The information provided by this element and the temperature detector is used to determine the dewpoint in containment. This calculation is done automatically, and the resulting dewpoint information is recorded in the control room. The containment building high humidity alarm on the supervisory panel is initiated by the information being received by the recorder.

6.7.1.2.5 Condensate Measuring System

This method of leak detection is based on the principle that under equilibrium conditions the condensate flow draining from the cooling coils of the containment air-handling units will equal the amount of water (and/or steam) evaporated from the leaking system. A reasonably accurate measurement of leakage from the reactor coolant system by this method is possible because containment air temperature and humidity promote complete evaporation of any leakage from hot systems. The ventilation system is designed to promote good mixing within the containment. During normal operation, the containment air conditions will be maintained below 120°F dry bulb and 92°F wet bulb (approximately 36-percent relative humidity) by the fan coolers.

When the water from a leaking system evaporates into this atmosphere, the humidity of the fan cooler intake air will begin to rise. The resulting increase in the condensate drainage rate is given by the equation

$$D = L\left[1 - \exp\left(-\frac{Q}{V}t\right)\right]$$

Where:

D = condensate drainage rate, gpm
L = evaporated leakage, gpm
Q = containment ventilation rate, cfm
V = containment free volume, ft^3
t = time after start of leak, min

Therefore, if four fan cooler units are operating (Q = 280,000 cfm), the condensation rate would be within 5-percent of a new equilibrium value in approximately 200 min after the start of the leak. The detection of the increasing condensation rate, however, would be possible within 5 to 10 min. The condensate measuring device consists essentially of a vertical 6-in. diameter standpipe with a notched weir cut into the upper portion of the pipe to serve as an overflow.

Each fan cooler is provided with a standpipe, which is installed in the drain line from the fan cooler unit. A differential-pressure transmitter near the bottom of the standpipe is used to measure the water level. Each unit can be drained by a remote-operated valve.

A wide range of flow rates can be measured with this device. Flows less than 1 gpm are measured by draining the standpipe and observing the water level rise as a function of time. Condensate flows from 1 gpm to 15 gpm are indicated in the central control room and can also be measured by observing the height of the water level above the notch of the weir. This water head can be converted to a proportional flow rate by means of a calibration curve. Flow rates greater than 15 gpm can also be determined using the calibration curves. A high-level alarm, set above the established normal (baseline) flow, is provided for each unit to alert the operator.

All indicators, alarms, and controls are located in the control room.

During the period of plant hot functional testing, a reactor coolant leak of known magnitude was simulated inside the containment vessel, and the performance of the humidity detector/condensate measuring system was observed. The leak was simulated by introducing steam into one of the loop compartments during a period when containment atmospheric conditions were stable and the fan cooler units were operating. The increase in containment atmosphere moisture content, as indicated by the humidity detectors, was recorded as a function of time following the initiation of the simulated leak. As a check, the same information was determined independently using different instrumentation. Elapsed time until condensation on the fan cooler unit cooling coils begins, as indicated by the condensate measuring devices, was recorded and compared with the calculated value on the basis of the initial containment humidity. Steam flow continued, and the performance of the condensate measuring devices in indicating the magnitude of steady cooling coil runoff was observed.

6.7.1.2.6 Component Cooling Liquid Monitor

This channel continuously monitors the component cooling loop of the auxiliary coolant system for activity indicative of a leak of reactor coolant from either the reactor coolant system, the recirculation loop, or the residual heat removal loop of the auxiliary coolant system. A scintillation detector is installed in the local radiation monitor skid assembly. This assembly is located in the primary auxiliary building and receives sample flow from the component cooling pump discharge downstream of the component cooling heat exchangers. The detector assembly output is amplified by a preamplifier, processed and transmitted to the radiation monitoring system console, the safety related display console and a recorder in the control room. The activity is indicated on digital displays. High-activity alarm indications are displayed on the control board annunciator and the safety related display console.

The measuring range of this monitor is 10^{-5} to 10^{-2} µCi/cm^3.

6.7.1.2.7 Condenser Air Ejector Gas Monitor

This channel monitors the discharge from the air ejector exhaust header of the condensers for gaseous radiation that is indicative of a primary to secondary system leak. The gas discharge is routed to the turbine roof vent. On high-radiation-level alarm, the condenser exhaust gases are diverted from the normal turbine building vent to the containment.

The processed detector output is transmitted to the radiation monitoring system console, the safety related display console and a recorder in the control room. The activity is indicated on

digital displays. High-activity alarm indications are displayed on the control board annunciator and the safety related display console.

This monitor is composed of an in-line spool piece, containing a normal range and an accident range detector. Each detector uses a beta sensitive scintillation detector to monitor the gaseous radiation level. Each detector includes adequate shielding to reduce interference with the detector's sensitivity caused by background radiation. The normal maximum channel output for this monitor is presented in Table 11.2-7.

6.7.1.2.8 Steam Generator Blowdown Liquid Sample Monitor

This channel monitors the liquid phase of the secondary side of the steam generator for radiation, which would indicate a primary-to-secondary system leak, providing backup information to that of the condenser air ejector gas monitor. Samples from the bottom of each of the four steam generators are piped to a common header, and the mixed sample is continuously monitored by a scintillation counter and holdup tank assembly. Upon indication of a high-radiation level, each steam generator is individually sampled to determine the source. This sampling sequence is achieved by manually selecting the desired unit to be monitored and allotting sufficient time for sample equilibrium to be established (approximately 1 min).

A remote indicator panel, mounted at the detector location, indicates the radiation level and high-radiation alarm.

The measuring range of this monitor is 10^{-5} to 10^{-2} $\mu Ci/cm^3$.

A photomultiplier tube-scintillation crystal (NaI) combination, mounted in a hermetically sealed unit, is used to monitor liquid effluent activity. Lead shielding is provided to reduce the background to a level so it does not interfere with the maximum sensitivity of the detector. The in-line, fixed-volume container is an integral part of the detector unit.

Personnel can enter the containment and make a visual inspection for leaks. The location of any leak in the reactor coolant system would be determined by the presence of boric acid crystals near the leak. The leaking fluid transfers the boric acid outside the reactor coolant system, and the process of evaporation deposits the crystals.

If an accident involving gross leakage from the reactor coolant system occurred, it could be detected by charging pump operation, system and sump inventories, containment sump pump operation, containment radiation monitors, humidity detectors, and the condensate monitoring system.

During normal operation, only one charging pump is operating. If a gross loss of reactor coolant to another closed system occurred, which was not detected by the methods previously described, the speed of the charging pump would indicate the leakage.

The leakage from the reactor coolant system will cause a decrease in the pressurizer liquid level that is within the sensitivity range of the pressurizer level indicator. The speed of the charging pump will automatically increase to try to maintain the equivalence between the letdown flow and the combined charging line flow and flow across the reactor coolant pump seals. If the charging pump at maximum speed is unable to maintain the required charging flow rate, then a pressurizer low level alarm actuates and a second charging pump may be started manually to maintain pressurizer level.

A break in the primary system would result in reactor coolant flowing into the containment and/or recirculation sumps. Gross leakage to these sumps would be indicated by the frequency of operation of the containment sump pump(s) and containment water level indicators. Since the building floor drains preferentially to the containment sump, the activity of the containment sump pumps and/or containment sump level would be more likely to indicate the leak first.

Gross leaks might be detected by an unscheduled increase in the amount of reactor coolant makeup water that is required to maintain the normal level in the pressurizer.

A large tube-side to shell-side leak in the nonregenerative (letdown) heat exchanger would result in reactor coolant flowing into the component cooling water and a rise in the liquid level in the component cooling water surge tank. The operator would be alerted by a high-water alarm for the surge tank and a high-temperature alarm actuated by a monitor at the component cooling water pump suction header and a high-radiation alarm actuated by a monitor sampling the component cooling water pump discharge.

A high-level alarm for the component cooling water surge tank and high-radiation and high-temperature alarms could also indicate a thermal barrier cooling coil rupture in a reactor coolant pump. However, in addition to these alarms, high temperature and low flow on the component cooling outlet line from the pump would activate alarms. Low thermal barrier component cooling water header return flow may be due to closure of FCV-625 on high flow or excessive usage by other loads.

Gross leakage might also be indicated by a rise in the normal containment and/or recirculation sump levels. Level transmitters with control room indication are provided for each sump.

6.7.1.2.9 Residual Heat Removal Loop

The residual heat removal loop removes residual and sensible heat from the core and reduces the temperature of the reactor coolant system during the second phase of plant shutdown.

During normal operation, the containment air particulate and radioactive gas monitors, the condensate measuring system, and the containment sump inventory monitoring capability provide means for detecting leakage from the section of the residual heat removal loop inside the reactor containment. These systems have been described previously in this section (see the description of leak detection from the reactor coolant system). Leakage from the residual heat removal loop into the component cooling water loop during normal operation would be detected outside the containment by the component cooling loop radiation monitor (see the analysis of detection of leakage from the reactor coolant system in this section).

The physical layout of the two residual heat removal pumps is within separate shielded and isolated rooms outside of the containment. This will permit the detection of a leaking residual heat removal pump by means of the plant vent gas monitoring system. Alarms in the control room will alert the operator when the activity exceeds a preset level. Small leaks to the environment could be detected with these systems within a short time after they occurred.

When the plant is shut down, personnel can enter the containment to check visually for leaks. The detection of the location of significant leaks would be aided by the presence of boric acid crystals near the leak. In case of an accident that involves gross leakage from the part of the residual heat removal loop inside the containment, this leakage would be indicated by a rise in

the containment and/or recirculation sump levels. Both of these sumps have level transmitters that provide level indication in the control room.

Should a large tube-side to shell-side leak develop in a residual heat exchanger or the RHR pump seal heat exchanger, the water level in the component cooling surge tank would rise, and the operator would be alerted by a high-water alarm. A radiation monitor for the component cooling loop provides a high activity alarm. A temperature monitor at the component cooling water pump suction header will also signal an alarm.

Leakage from both of the residual heat removal pumps, including leakage resulting from a residual heat removal pump seal failure, is drained to a common sump equipped with a sump pump. In addition, a level monitor in this sump will actuate an alarm when the level exceeds a preset level.

6.7.1.2.10 Recirculation Loop

If a break occurs in the reactor coolant system, the recirculation loop provides long-term protection by recirculating spilled reactor coolant and injected refueling water.

The containment air particulate and radioactive gas monitors, the humidity detectors, the condensate measuring system, and the containment sump inventory monitoring capability (see the section discussing leak detection for the reactor coolant system) provide means of detecting small leaks in the part of the recirculation loop inside the reactor containment.

Leakage from the residual heat exchanger would be detected by a radiation monitor (discussed in the section on leak detection from the reactor coolant system) that receives sample flow from the component cooling water pump discharge downstream of the component cooling heat exchangers.

During a containment entry personnel could check for leaks evidenced by the presence of boric acid crystals.

Gross leakage from the recirculation loop inside the containment might be indicated by a rise in the level of the containment and/or recirculation sumps. Both of these sumps have level transmitters that provide level indication in the control room.

A rise in the liquid level in the component cooling surge tank would result if a large tube-side to shell-side leak developed in a residual heat exchanger. The operator would be alerted by a high-level alarm in the component cooling water surge tank and a high-temperature alarm actuated by a monitor at the component cooling water pump suction header and a high-radiation alarm actuated by a monitor sampling the component cooling water pump discharge.

Leakage outside containment depending on location will be diverted by floor drains to either the PAB sump tank or PAB sump where it will then be transferred to the Waste Holdup Tank.

6.7.1.2.11 Component Cooling Loop

Leakage from the component cooling loop inside the reactor containment could be detected by the humidity detectors and/or condensate measuring system and the containment water level indicators (see the section on reactor coolant system leak detection for a description of these systems).

Visual inspection inside the containment is possible for some locations during containment entries.

Gross leakage from the component cooling loop would be indicated inside the containment by a rise in the liquid level of the containment and/or recirculation sumps. Both of these sumps have level transmitters that provide level indication in the control room.

Leakage outside containment depending on location will be diverted by floor drains to either the PAB sump tank or PAB sump where it will then be transferred to the Waste Holdup Tank.

6.7.1.2.12 Service Water System

During a loss-of-coolant accident, the containment fan coolers service water monitors check the containment fan service water discharge line for radiation indicative of a leak from the containment atmosphere into the service water. A small bypass flow from each of the heat exchangers is mixed in a common header and monitored by redundant scintillation detectors mounted in separate holdup tank assemblies. Upon indication of a high-radiation level, each heat exchanger is individually sampled to determine which unit is leaking. This sampling sequence is achieved by manually selecting the desired unit to be monitored and allotting sufficient time for sample equilibrium to be established (approximately 1 min). The discharge line from the fan coolers motor coolers is also monitored for radioactivity and isolated in a like manner following the detection of a leak.

The measuring range of this monitor is 10^{-5} to 10^{-2} $\mu Ci/cm^3$.

Gross leakage from the service water system due to a faulty cooling coil in the containment air recirculation cooling system can be detected by weir level transmitters. Any significant cooling water leakage would be seen as flow into a fan cooler unit weir.

Leakage outside containment depending on location will be diverted by floor drains to either the PAB sump tank or PAB sump where it will then be transferred to the Waste Holdup Tank.

6.7.1.2.13 Containment Sump Level and Discharge Flow

The sump flow detection system includes flow metering and totalizing that is indicated in the control room. It is capable of detecting a 1 GPM leak within 4 hours (Reference 1).

[Deleted] (LT-941), other level instrument qualified to Class 1E, IEEE-323-1974, and IEEE-344-1975 is provided for continuous level indication (LT-3300). Additionally, existing level monitor (LT-941) was upgraded to meet environmental qualification requirements to support minimum NPSH requirements for recirculation pump start. LT-941 has since re-evaluated and can no longer be used for leakage detection based on the spacing and wiring of its associated sensors. (Reference 1)

Sump level is maintained by the action of containment sump pumps. The pumps are independently powered and have separate power and control cables to each pump.

6.7.1.2.14 Recirculation Sump Level

Water may also collect in the recirculation sump although under most circumstances the containment sump will be filled before the recirculation sump. Water level monitoring of the recirculation sump is provided by two level instruments, which actuate control room lights at discrete sump/containment water levels and provide an audible alarm for certain discrete levels within the recirculation sump. Level monitor LT-939 is environmentally qualified to support minimum NPSH requirements for recirculation pump start. A continuous level monitor is also provided for wide-range level indication.

6.7.1.2.15 Reactor Cavity Pit Level

The level in the reactor cavity pit is controlled through the action of reactor cavity pit pumps that are used to pump to the containment sump any water that may have leaked into the pit. The system is designed for usage during refueling/maintenance outages. Four alarms are located in the central control room on panel SB-1:

Reactor Cavity Pit Water Level High (LC-7049)
Reactor Cavity Pit Water Level High-High (LC-7049)
Reactor Cavity Sump Pit Pump No. 1 Auto Run (LC-7043)
Reactor Cavity Sump Pit Water Level High (LC-7042)

LC-7042 and LC-7043 control Reactor Cavity Sump Pit Pump No. 1. LC-7049 controls Reactor Cavity Pit Pump No. 2. A continuous level monitor (LT-3302) is also provided for wide-range level indication within the reactor cavity pit.

The reactor cavity pit pumps are independently powered and have separate control wiring.

6.7.2 Leakage Provisions

Provisions are made for the isolation and containment of any leakage.

6.7.2.1 Design Basis

The provisions made for leakage are designed to prevent uncontrolled leaking of reactor coolant or auxiliary cooling water. This is accomplished by (1) isolating the leak by valves, (2) designing relief valves to accept the maximum flow rate of water from the worst possible leak, (3) supplying redundant equipment that allows a standby component to be placed in operation while the leaking component is repaired, and (4) routing the leakage to various sumps and holdup tanks.

6.7.2.2 Design and Operation

Various provisions for leakage avert uncontrolled leakage from the primary and auxiliary coolant loops.

6.7.2.2.1 Reactor Coolant System

When significant leakage from the reactor coolant system is detected, action is taken to prevent the release of radioactivity to the atmosphere outside the plant.

If either the containment air particulate gamma activity or the radioactive gas activity exceeds preset levels, the containment purge supply and exhaust duct valves and pressure relief line valves are closed, if open.

On high-radiation alarm signaled by the condenser air ejector monitor, the condenser exhaust gases are diverted from the turbine roof vent to the containment.

A high-radiation alarm actuated by the steam-generator liquid sample monitor initiates the closure of the isolation valves in the blowdown lines. A sample stop valve and a blowdown tank inlet stop valve located downstream of each pair of isolation valves will automatically close when either of their associated blowdown line isolation valves leaves its fully open position in order to isolate their respective lines.

If the component cooling loop radiation monitor signals a high-radiation alarm, the valve in the component cooling surge tank vent line automatically closes to prevent gaseous activity release.

If a leak from the reactor coolant system to the component cooling loop were a gross leak or if the leak could not be isolated from the component cooling loop before the inflow completely filled the surge tank, the relief valve on the surge tank would open. The discharge from this valve is routed to the waste holdup tank in the primary auxiliary building.

A large leak in the reactor coolant system pressure boundary, which does not flow into another closed loop, would result in reactor coolant flowing into the containment sump and/or the recirculation sump.

Experience with the detection of primary system leakage into the containment vessel of Indian Point Unit 1 is discussed in Appendix 6B.

6.7.2.2.2 Residual Heat Removal Loop

High containment air particulate gamma activity or high radioactive gas activity will result in an alarm being activated by either the containment air particulate or radioactive gas monitors. The containment purge supply and exhaust duct valves and pressure relief line valves are closed automatically, if open. This prevents the release of radioactivity to the atmosphere outside the nuclear plant.

If leakage from the residual heat removal loop into the component cooling loop occurs, the component cooling radiation monitor will actuate an alarm and the valve in the component cooling surge tank vent line is automatically closed to prevent gaseous radioactivity release. If the leaking component (i.e., a residual heat exchanger) could not be isolated from the component cooling loop before the inflow completely filled the surge tank, the relief valve on the surge tank would open and the effluent would be discharged to the primary auxiliary building waste holdup tank.

Gross leakage from the section of the residual heat removal loop inside the containment, which does not flow into another closed loop would result in reactor coolant flowing into the containment sump and/or the recirculation sump.

Other leakage provisions for the residual heat removal loop are discussed in Section 9.3.

6.7.2.2.3 Recirculation Loop

The containment purge supply and exhaust duct valves and pressure relief line valves are automatically closed, if open, when either the containment air particulate or the radioactive gas monitors read above a preset level. This prevents radioactivity from escaping to the outside atmosphere.

Leakage from the recirculation loop into the component cooling loop results in a radiation alarm and the automatic closing of the component cooling surge tank vent line to prevent gaseous radioactivity release. If the leak were gross and filled the surge tank before the leaking component could be isolated from the component cooling loop, the relief valve on the surge tank would open and the effluent would be discharged to the waste holdup tank in the primary auxiliary building.

Gross leakage from the internal recirculation loop, which does not flow into another closed loop will flow into the containment sump and/or the recirculation sump. Leakage outside containment depending on location will be diverted by floor drains to either the PAB sump tank or PAB sump where it will then be transferred to the Waste Holdup Tank.

6.7.2.2.4 Component Cooling Loop

Gross leakage from the section of the component cooling loop inside the containment, which does not flow into another closed loop will flow into the containment sump and/or the recirculation sump. Leakage outside containment depending on location will be diverted by floor drains to either the PAB sump tank or PAB sump where it will then be transferred to the Waste Holdup Tank.

Other provisions made for leakage from the component cooling loop are discussed in Section 9.3.

6.7.2.2.5 Service Water System

Leakage outside containment depending on location will be diverted by floor drains to either the PAB sump tank or PAB sump where it will then be transferred to the Waste Holdup Tank.

6.7.3 Minimum Operating Conditions

The Technical Specifications establish limiting conditions regarding the operability of the leakage detection systems.

REFERENCES FOR SECTION 6.7

1. Attachment C of IP2's May 23, 1988 letter to the NRC requesting elimination of Postulated Primary Loop Pipe Rupture as a Design Basis, "Indian Point Unit 2 – Evaluation of RG-1.45 Compliance."

TABLE 6.7-1
Class 1 Fluid Systems For Which
No Special Leak Detection Is Provided

System	Remarks On Leakage Detection
1. Residual heat removal	Refer to items 1, 2, and 3 (Section 6.7.1.2) and Section 6.7.1.2.9.
2. Component cooling	Refer to items 2 and 3 (Section 6.7.1.2) and Section 6.7.1.2.11
3. Service water	Refer to item 3 (Section 6.7.1.2) and Section 6.7.1.2.12
4. Auxiliary feedwater	Visual
5. Waste disposal	Auxiliary building sump pump operation. Also refer to item 1 (Section 6.7.1.2)

6.8 POSTACCIDENT HYDROGEN CONTROL SYSTEMS

On April 14, 2005, NRC issued IP2 License Amendment 243 which eliminated the requirement for hydrogen recombiners to provide any combustible gas control function. Therefore, the technical specification requirements for the hydrogen recombiners have been eliminated. However, the actual equipment remains in service until such time that an alternate disposition of this equipment is established and implemented.

6.8.1 Design Basis

The function of the hydrogen control system is to control the hydrogen generated within the containment following a loss-of-coolant accident.

A Hydrogen Recombiner System is provided to control the post-accident hydrogen concentration in containment. The Hydrogen Recombiner System uses two redundant safety-related Passive Hydrogen Recombiners (PHR). Each recombiner unit is capable of maintaining the hydrogen concentration at or below 4 volume percent. The flame type recombiners system installed originally has been retired in place within the containment building and the associated containment isolation valves have been de-energized in closed position.

The post-accident containment venting system provides a backup method to containment recombiners for controlling the potential hydrogen accumulation in the containment. This is accomplished by the controlled venting of containment atmosphere to maintain the hydrogen concentration at a safe level. The venting system is designed to limit the hydrogen concentration below 4 volume percent.

A containment air sample is taken from each of the containment fan cooler units at a point located downstream from the fan. Sample analysis will determine the requirement for the post-accident containment venting system operation.

6.8.2 System Design and Operation

6.8.2.1 Passive Hydrogen Recombiners

Two 100% capacity independent hydrogen recombiners are provided. The recombiners are passive devices, which contain no moving parts and do not need electrical power or any other support system. Recombination is accomplished by the attraction of oxygen and hydrogen molecules to the surface of the catalyst, Reference 3.

The PHRs used at IP 2 are Passive Autocatalytic Recombiners (PARs) designed and manufactured by NIS Ingenieurgesellschaft mbH of Germany (NIS). The PHR consists of a stainless steel sheet metal box open at the bottom and at both sides on the top as shown on Figure 6.8-1. The approximate size of the box is 1m x 1m x 1m. There are 88 catalytic cartridges inserted into each box. Each cartridge fabricated from perforated steel plates holds catalyst pellets. The catalyst pellets are made from aluminum oxide spheres and are coated with palladium and hydrophobic polymers. The palladium coating acts as a catalyst and the hydrophobic coating provides water proofing. The catalyst cartridges are installed vertically, spaced 1 cm apart, in each box. The spaces between the cartridges serve as flow channels for the gases. Airflow enters at the bottom and the catalyst combines hydrogen and oxygen in the flow channels to form gaseous water.

PHRs are designed for self-starting and self-sustaining reaction. The exothermic reaction of the combination produces heat, which results in a convective flow that draws more gases from the containment atmosphere into the unit from below.

A single recombiner is capable of maintaining the hydrogen concentration in containment below the 4 volume percent flammability limit. The second recombiner is redundant and is installed to provide margin and increased containment coverage.

PHRs are safety related and are seismic class I, and have undergone environmental qualification testing in accordance with IEEE 627 and IEEE 344.

6.8.2.2 Containment Vent System

The postaccident containment venting system consists of a common penetration line that acts as a supply line through which outside air can be admitted to the containment, and an exhaust line, with parallel valving and piping, through which hydrogen-bearing gases from containment may be vented through a filter. The system is shown in Figure 6.8-3.

The supply mode makes use of instrument air to feed containment. The nominal flow rate from either of the two instrument air compressors is 225 scfm.

In the exhaust mode, the line penetrates the containment and then is divided into the parallel lines. Each parallel line contains a pressure sensor and all the valves necessary for controlling the venting operation. The two lines then rejoin and the exhaust passes through a flow sensor and a temperature sensor before passing through charcoal and HEPA filters. The exhaust is then directed to the plant vent.

The venting system requires a differential pressure between the containment and the outside atmosphere in order to permit venting. This is based on a pressure of 2.14 psig in the containment. If required, the containment is pressurized to 2.14 psig with instrument air when

the hydrogen reaches 3 volume percent after the loss-of-coolant accident. The hydrogen concentration is reduced by this pressurization. Purging is then delayed until the hydrogen concentration in the containment has once again built up to 3 volume percent.

6.8.2.3 Containment Air Analyzer System

Two hydrogen/oxygen analyzers have been installed to continuously monitor the hydrogen and oxygen concentrations in the containment atmosphere. A new system of valves, controlling equipment, and a central control room display have also been installed to replace the old method of manual containment air sampling. This system functions independently of the original vacuum pump (which has been isolated) and provides the required sampling capability under NUREG-0737, Item II.B.3. The containment atmosphere sampling system was evaluated by the NRC against the NUREG-0737 requirements and found acceptable.[1,2] The requirements for the containment atmosphere sampling system were removed from the Technical Specifications by License Amendment No. 222 as discussed in Section 9.4.1.1. The Technical Specification requirements for these instruments were eliminated and replaced by a licensee commitment to maintain the monitors as reliable and functional through a preventive maintenance program. In addition, Regulatory Guide 1.97 categorization for these instruments was changed from Category 1 to Category 3. The system is shown in Plant Drawing 208479 [Formerly UFSAR Figure 6.8-4].

The system has a closed-loop flow path with the sampled air withdrawn from and discharged to the containment. The operation of the system is from remote control panels located in the motor control center room at elevation 98-ft of the primary auxiliary building. The new solenoid-operated valves function to both pass a sample for analyzing and to provide containment isolation when not sampling or during an accident.

The hydrogen/oxygen analyzers are located on elevation 80-ft of the primary auxiliary building. Direct indicator readout is provided in the 98-ft motor control center room in the primary auxiliary building. The hydrogen/oxygen values are indicated and recorded on two recorders located on the control room accident assessment panel. One recorder is used for each channel. Nitrogen purging capability of the sample lines is also provided. The purging reduces line activity and radiation field in the primary auxiliary building after a sample is drawn. The purge is so arranged that it is sent back to the containment.

All equipment is procured and installed Class A, seismic Class I. The recorders, solenoid-operated valves and other electrical equipment are Class 1E. The eight solenoid-operated valves provided with this system are powered from four separate dc power supplies and configured so that the system meets single-failure criteria both for the path opening (i.e., sampling) and for the path closing (i.e., containment isolation) safety functions.

6.8.2.4 System Operation

PHRs are located inside the containment, they are totally passive, and are self-starting self-feeding devices. No operator action is required to initiate recombination of accident hydrogen with containment oxygen.

6.8.3 Postaccident Hydrogen Generation

During the postaccident period, hydrogen is generated in the containment from the following sources:

1. Zirconium-water reaction.
2. Chemical reaction of materials subject to corrosive attack.
3. Radiolytic decomposition of coolant in the core.
4. Radiolytic decomposition of coolant in the sump.

These results are shown in Plant Drawings 9321-2568 & -2569 [Formerly UFSAR Figure 6.8-2] for the first 30 days of the post-accident period and have been obtained on the following bases.

6.8.3.1 Zirconium-Water Reaction

1. Five percent of the core cladding reacts immediately with core cooling solution according to the reaction

$$Zr + H_2O \rightarrow ZrO_2 + H_2 + heat$$

2. There are 44,197 lbm of zirconium cladding in the core.

6.8.3.2 Corrosion of Materials of Construction

1. Corrosion of aluminum and zinc according to the reactions

   $2Al + 3H_2O = Al_2O_3 + 3H_2$

   $Zn + H_2O = H_2 + ZnO$

2. Aluminum and zinc corrosion rates versus time postaccident

Time (days)	Al Corrosion Rate (mil/yr)	Zn Corrosion Rate (mil/yr)
0.0	5,500	180
0.0035	1,700	180
0.0116	600	160
0.0232	200	110
0.0464	200	20
20	200	20

3. Aluminum available for reaction as follows:

Item	Mass (lb)	Area ($ft^{2)}$)	Thickness (inch)
Control rod drive mechanism connectors	25	14	0.207
Reactor vessel insulation foil	269	10,000	0.0019
Area monitors	6	4	0.107
Source, intermediate, and power range detectors	140	40	0.245
Process instrumentation and controls	420	84	0.356
Lighting fixtures and	1061	380	0.199

equipment			
Paint on steam generator, pressurizer, and reactor vessel	140	10,000	0.001
Contingency	2500	850	0.209

4. A contingency for zinc available for reaction was made by assuming a 20,000 square-foot surface, thick enough (0.065 inch) to not corrode all the way through in 30 days.

6.8.3.3 Core Radiolysis

Regulatory Guide 1.7, Rev.2, "*Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident*" describes methods acceptable to the NRC staff for implementing the requirements of 10 CFR 50.44. For Indian Point 2, the core and sump radiolysis analyses have been done in accordance with that Guide.

In summary, the effects of core radiolysis are as follows:

1. 50-percent of the halogens, 100-percent of the noble gases, and 99-percent of all other fission products are retained in the core following the accident.

2. 0.50 molecules of hydrogen are generated per 100 eV of energy absorbed by water in the core.

3. 10.0-percent of the core fission product gamma energy is absorbed by the solution in the core.

4. Beta energy is absorbed by the fuel and cladding and does not contribute to hydrogen generation in the core.

5. Core fission product decay energy is calculated in accordance with Branch Technical Position ASB 9-2, as originally implemented in the COGAP computer code and transferred to STARGAP.

6. The plant is operational for 830 days at 3216 MWt before the accident.

6.8.3.4 Sump Radiolysis

1. 50-percent of the halogens, none of the noble gases, and 1-percent of all other fission products are released from the core to the sump during the accident.

2. 0.50 molecules of hydrogen are generated per 100 eV of energy absorbed by the sump solution.

3. All beta and gamma energy emitted by fission products in the sump solution are absorbed and contribute to hydrogen generation.

4. The plant is operational for 830 days at 3216 MWt before the accident.

6.8.4 Evaluation

To evaluate the potential for hydrogen accumulation in containment following a LOCA, the hydrogen generation as a function of time following the initiation of the accident is calculated. Assumptions recommended by Reference 6 are used to maximize the amount of hydrogen calculated. The hydrogen release to containment is graphically presented in UFSAR Figure 6.8-2. It will take more than 20 days after a LOCA for the hydrogen concentration in the containment to reach 4.0 volume percent, if no recombiner was functioning.

The PHRs are designed such that, a single recombiner is capable of limiting the peak hydrogen concentration in containment to less than 4.0 volume percent.

Two PHR are located on the operating deck at an approximate elevation of 29 m (95 feet) out side the missile shield wall. This location is away from the reactor coolant piping and possible impingement from high-energy line breaks.

Mixing of the hydrogen in containment is ensured through the use of the containment fan coolers. Three to five fan coolers operate in the post-accident environment. The capability of the fan coolers is described in Section 6.4. The flow distribution from fan coolers is directed to various locations including locations high in containment. Both PHRs are mounted on the open grating to allow free airflow through the PHRs. The housing extends above the catalyst elevation to provide a chimney to yield additional lift to enhance the efficiency of the device. One condition that is to be avoided for proper PHR functioning is submersion in water. The PHR are located above the containment flood level and they are designed with a spray hood to minimize the direct contact with the post-accident containment sprays.

Since PHRs are credited for post-LOCA hydrogen control the effects of potential catalyst inhibitors and poisons present during the accident conditions were accounted for. The Electric Power Research Institute (see Reference 5) has evaluated the effects of potential catalyst inhibitors and poisons. Limited decreases in catalyst effectiveness were noted. Any decrease in effectiveness is more than compensated for by the combination of conservative hydrogen generation assumptions and substantial excess installed PHR capacity.

6.8.4.1 Qualification Testing

As a prerequisite for final installation of PHRs at Indian Point 2, environmental qualification tests were performed at Wyle Laboratories in accordance with the applicable requirements of 10CFR21, 10CFR50 Appendix B, and ANSI N45.2. Sample cartridges from equipment device designed and built by NIS Ingenieurgesellschaft mbH (NIS) supplied for Indian Point 2 were subjected to the qualification testing. The results of these tests were submitted to NRC and NRC approved (Reference 4) the use of NIS PARs at Indian Point 2.

The details of these tests, the acceptance criteria and the results are documented in the Wyle test report, which is filed in Indian Point EQ files.

6.8.5 Inspections And Tests

A visual examination and an operating check of the system is performed periodically, as specified in the Technical Specifications. Visual examination and cleaning if necessary is performed at each refueling outage to verify that there is no significant fouling by foreign material. Performance of a test on a sample plate removed from each hydrogen recombiner at

each refueling outage ensures the recombiners are operational. The sample plate removed from each recombiner is inserted into a test device and a fixed flow mixture of gas that is 1% to 1.5% hydrogen in air is supplied to the device. The plate is judged to be degraded if the temperature developed is not within the acceptance criteria. In this case the neighboring plate will be tested. Any plates found to be degraded will be evaluated or replaced with new plates.

6.8.6 Minimum Operating Conditions

Operability requirements for the PHRs were removed from the Technical Specifications by license amendment #243.

REFERENCES FOR SECTION 6.8

1. Letter from S. A. Varga, NRC, to J. D. O'Toole, Con Edison, Subject: Postaccident Sampling at the Indian Point Unit 2, Safety Evaluation Report, dated June 28, 1984.

2. Letter from S. A. Varga, NRC, to J. D. O'Toole, Con Edison, Subject: Postaccident Sampling at the Indian Point Unit 2, Safety Evaluation Report, dated December 12, 1984.

3. S. E. Quinn to Document Control Desk, USNRC, "Proposed Technical Specification for H2 Recombiners", August 21, 1996.

4. Letter from Jeffrey F. Harold, NRC to A. Alan Blind, Con Edison “Issuance of Amendment and Bases Change for Indian Point Nuclear Generation Unit No. 2 (TAC No. M96475)”, dated April 27,1999.

5. EPRI ALWR Report GC-108771 “Effects of Inhibitors and Poisons on the Performance of Passive Autocatalytic Recombiners (PARs) for Combustible Gas Control in ALWRs”, June 28, 1999.

6. Regulatory Guide 1.7, Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident (Rev. 2)

7. Letter from P. Milano, NRC to M. Kansler, Entergy; “Issuance of Amendment (243) Eliminating Requirements for Hydrogen Recombiners and Hydrogen Monitors”, dated April 14, 2005.

6.8 FIGURES

APPENDIX 6A
EFFECTIVENESS OF THE CONTAINMENT SPRAY SYSTEM TO REMOVE AIRBORNE ACTIVITY FOLLOWING A LOCA

In the event of a postulated Loss-of-Coolant Accident (LOCA) with degraded core, there would be a substantial release of core fission product activity to the containment atmosphere. The core degradation source term is assumed to release the following fractions of core activity to the containment atmosphere consistent with the source term described in Regulatory Guide 1.183 (Reference 1):

Noble gases	1.0
Iodines	0.4
Alkali metals	0.3
Tellurium group	0.05
Barium & Strontium	0.02
Noble metals	0.0025
Cerium group	0.0005
Lanthanides	0.0002

With the exception of the noble gases, the activity released to the containment atmosphere is subject to removal by the containment spray system. As defined in Regulatory Guide 1.183, the iodine activity entering the containment atmosphere is assumed to exist primarily as an aerosol (95% of the iodine is assumed to be in the form of cesium iodide) with the remainder existing as elemental iodine vapor (4.85%) or as organic compounds (0.15%). The activity for the remaining nuclide groups is assumed to all be in the aerosol form.

The containment spray system is one of the engineered safety features systems employed following a LOCA to reduce the pressure and temperature in the containment. The spray system also affords an excellent means of removing both elemental iodine vapor and aerosols from the containment atmosphere. Organic iodine compounds are assumed not to be subject to removal by sprays.

During the spray injection phase the spray consists of boric acid solution taken from the refueling water storage tank. This spray solution has a pH of ~4.5 – 5.0. While sprays are an effective means to remove airborne iodine, retention of iodine in the sump solution requires that the solution pH be raised to 7.0 or above. This pH adjustment is provided by the sodium tetraborate stored in baskets in the sump area (See Section 6.3).

The removal of airborne activity by the containment sprays is expressed by the following equation which calculates the amount of activity remaining at a given time:

$$C = C_O \, e^{-(\lambda_s t)} \qquad (6A\text{-}1)$$

where:

C = Current activity, Ci
C_o = Initial activity, Ci
λ_s = Spray removal coefficient, hr^{-1}

t = time, hr

6A.1 SPRAY REMOVAL COEFFICIENT FOR PARTICULATES

The spray removal coefficient for particulates is determined using the model described in SRP Section 6.5.2 (Reference 2):

$$\lambda_p = 3hFE / 2VD \qquad (6A\text{-}2)$$

where:

λ_p = Particulate removal rate constant due to spray removal, hr^{-1}
h = Drop fall height, ft
F = Spray flow rate, ft^3/hr
V = Volume Sprayed, ft^3
E = Single drop collection efficiency
D = Drop diameter, ft

The value for E/D is conservatively defined in SRP Section 6.5.2 to be 10 m^{-1} (3.048 ft^{-1}) initially and is reduced by a factor of 10 after the suspended aerosol mass has been depleted by a factor of 50 (i.e., after 98% of the aerosols have been removed).

Spray Fall Height and Sprayed Volume

The spray system nozzle and header arrangement is designed to cover a maximum area in the upper containment. Four headers, arranged in concentric circles, are located in the containment dome at elevations of 213.5, 218.6, 223.6, and 228.6 feet.

Credit is taken for spray coverage of the total volume above the operating deck. The spray fall height from the lowest of the spray headers is 118.5 feet. It is conservatively assumed that all spray falls through the same distance (ignoring the additional fall-time associated with the higher spray headers). The sprayed volume is 80 percent of the containment free volume (i.e., 2.088×10^6 cubic feet).

Containment Spray Flow Rate

During the spray injection phase it is assumed that only one of the two spray pumps is operating, drawing water from the refueling water storage tank. The minimum spray flow rate is 2135 gpm (17,646 ft^3/hr) for one spray pump under design basis accident conditions.

Once the inventory of the refueling water storage tank is depleted, there is a switch to the recirculation spray phase. The containment spray pump is not used during the spray recirculation phase; instead, the flow to the spray headers is obtained from the recirculation pump which recirculates water from the sump to the reactor vessel. The system resistance provided by the physical configuration of the recirculation piping and components is hydraulically balanced such that sufficient flow is established to the core and to the spray headers. The minimum recirculation spray flow rate of 1100 gpm (8823 ft^3/hr) is half the injection spray flow rate.

Spray Removal Coefficient

Using the above-defined values, the spray removal coefficient for aerosols is determined as follows:

$\lambda_p = 3hFE / 2VD$

Injection Phase: $\lambda_p = 4.5\ hr^{-1}$

Recirculation Phase: $\lambda_p = 2.28\ hr^{-1}$

These values are reduced by a factor of 10 once the aerosol inventory is reduced to two percent of the total aerosol inventory released to the containment atmosphere.

6A.2 SPRAY REMOVAL COEFFICIENT FOR ELEMENTAL IODINE

The original design of the containment spray system included spray additive (sodium hydroxide) to increase the pH of the boric acid solution to approximately 9.5. The purpose of the pH adjustment was to increase spray removal of elemental iodine. However, as discussed in SRP Section 6.5.2 (Reference 2), it has been determined that the boric acid spray without pH adjustment to alkaline conditions is an effective means to remove airborne elemental iodine. The spray removal coefficient for elemental iodine is determined using the model described in SRP Section 6.5.2:

$$\lambda_s = 6K_g tF / VD \qquad (6A\text{-}3)$$

where

λ_s = Elemental iodine removal rate constant due to spray removal, hr^{-1}
K_g = Gas phase mass transfer coefficient, ft/min
t = Average spray droplet fall time, min
F = Spray flow rate, ft^3/hr
V = Volume sprayed, ft^3
D = Average drop diameter, ft

The values for spray flow rate and sprayed volume are defined in Section 6A.1. The gas phase mass transfer coefficient (K_g) is conservatively defined as 3 m/min (9.084 ft/hr) in Reference 3 based on a number of experimental studies.

SRP Section 6.5.2 specifies an upper limit for λ_s of 20 hr^{-1} for fresh solution.

Drop Diameter

There is a spectrum of drop sizes in the containment atmosphere at any time. Measurements of the drop size distribution of the Sprayco 1713 nozzle have shown that the drop size varies between 80 and 3800 μm, with ~80 percent of the generated drops being 500 μm or smaller (Reference 4). The initial distribution of drops after release from the nozzles is affected during the fall through the containment atmosphere by coalescence and by condensation of steam onto the drops. The increase in drop size associated with condensation increases the total surface area of the droplets that would be available for absorbing iodine. While coalescence of drops

drops also increases individual droplet size, it decreases the number of droplets and thus decreases the total surface area of the airborne droplets. Increases in droplet size also decrease the fall time during which the drops are available to remove elemental iodine from the containment atmosphere.

Since the spray droplets enter the containment at temperatures far below that of the initial temperature of the containment atmosphere, condensation of steam from the containment air-steam mixture will increase the initial size of the drops until they are in thermal equilibrium with the ambient.

From an energy balance on the drop:

$$mh + m_c\, h_g = m'h_f \qquad (6A\text{-}4)$$

where:

m = mass of drop before condensation
h = enthalpy of drop at spray inlet temperature
m_c = mass of condensed steam
m' = mass of drop after condensation
h_g = saturation vapor enthalpy at containment condition
h_f = saturation liquid enthalpy at containment condition

From a mass balance: $m_c = m' - m$

Substituting into equation 6A-4: $mh + (m' - m)h_g = m'h_f$

or: $m'/m = (h_g - h)/(h_g - h_f)$

But, $m = 4\pi r^3/3v$ and $m' = 4\pi r'^3/3v_f$

where v is the specific volume at inlet conditions and v_f is the specific volume at containment conditions.

The increase in the drop radius due to condensation then is:

$$\frac{r'}{r} = \left(\frac{v_f (h_g - h)}{v (h_g - h_f)} \right)^{\frac{1}{3}} \qquad (6A\text{-}5)$$

Just as the spray droplets can remove aerosols from the containment atmosphere, the droplets are capable of removing other droplets. This collision and resulting coalescence result in an increase in the average drop size. Coalescence efficiency is the probability that a collision between two drops will result in the formation of a single larger drop. It is conservatively assumed that all droplet collisions result in coalescence. This is a very conservative assumption which results in the prediction of an average droplet size significantly larger than would be expected to occur.

Taking the effects of coalescence and condensation into consideration, the value for the mean droplet diameter is 1200 µm (3.94E-3 ft).

Spray Droplet Fall Time

One of the simplifying assumptions of the model is that the residence time of the drop in the containment atmosphere may be approximated by fall height divided by the terminal velocity of the drop (h/U_t).

However, the actual residence time of the drop is considerable longer, since the drops do not leave the nozzle with only a vertical velocity component, but with an additional horizontal component, which causes the droplets to fall along a trajectory, which increases the residence time of the droplet. (This fact is further amplified by the 45-degree nozzles.) Thus, the use of h/U_t, combined with defining the spray fall height as the fall height for the lowest elevation spray ring header, adds further conservatism to the model. This model gives a minimum residence time of 10.7 seconds for a 1200 μm drop.

Spray Removal Coefficient

Using the above-defined values, the spray removal coefficient for aerosols is determined as follows:

$$\lambda_s = 6K_g tF / VD$$

Injection Phase:	$\lambda_s = 20\ hr^{-1}$	The calculated value of 22.5 hr^{-1} is reduced to 20 hr^{-1} consistent with the upper limit defined in SRP Section 6.5.2.
Recirculation Phase:	$\lambda_s = 5.6\ hr^{-1}$	The calculated value of 11.2 hr^{-1} is reduced by 50% to 5.6 hr^{-1} to address the loading of the recirculating solution with elemental iodine.

6A.3 LIMITS OF REMOVAL

There is no defined limit in SRP Section 6.5.2 on the removal of aerosols from the atmosphere. A conservative approach, which is used for Indian Point Unit 2, is to limit credit for aerosol removal to a DF of 1000 (i.e., 0.1% of the original inventory release to the containment atmosphere remaining airborne). If recirculation spray is terminated prior to reaching this limit, sedimentation will continue to remove aerosols until the DF limit is reached. The removal rate associated with sedimentation is addressed in Section 14.3.6.1.

SRP Section 6.5.2 specifies a limit on elemental iodine removal of a DF of 200 (i.e., 0.5% of the original inventory release to the containment atmosphere remaining airborne). This is dependent on the sump solution pH being adjusted to ≥7.0. As discussed in Section 6.3.2, the mass of Sodium Tetraborate stored in the containment is sufficient to assure that following a LOCA a sump solution pH of ≥7.0 is achieved and maintained.

REFERENCES FOR APPENDIX 6A

1. USNRC Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000.

2. NUREG-0800, USNRC Standard Review Plan, Section 6.5.2, “Containment Spray as a Fission Product Cleanup System,” Revision 2, December 1988.

3. R. E. Davis, H. P. Nourbakhsh, and M. Khatib-Rahbar, “Fission Product Removal Effectiveness of Chemical Additives in PWR Containment Sprays,” Technical Report A-3788, Brookhaven National Laboratory, 8/12/86.

4. WCAP-8659, “CIRCUS Computer Code – Calculation of Vapor Phase Elemental Iodine Removal in the Reactor Containment by Chemical Additive Spray,” June 1975, (Westinghouse Proprietary)

APPENDIX 6B
PRIMARY SYSTEM LEAK DETECTION INTO CONTAINMENT VESSEL, INDIAN POINT UNIT 1

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6B.0 OPERATIONAL EXPERIENCE

During the lifetime of Indian Point 1, primary system leakage was minimal. A combination of all the following instrumentation was used to detect several leaks ranging in size from 0.1 to 3 gal/hr. However, because of the magnitude of these leaks, positive identification resulted only from visual inspection during containment entries made after scheduled plant shutdown.

Small leaks that developed in the primary system pressure boundary could be detected by several continuously recording instruments available to the plant operators. The most sensitive of these detectors was the radioactive air particulate monitor, which continuously sampled the air in the containment cooling system. The purpose of the containment cooling system was to maintain proper ambient temperatures for equipment in the containment vessel. This system took air from the upper elevations of the vessel and recirculated it through cooling coils on the suction side of the supply fan. This air was then discharged at a rate of 40,000 cfm through steam coils. The turnover rate of the containment vessel as a result of this system was approximately once every hour. By sampling air from the discharge of the containment cooling system supply fan, leak rates as small as 0.3 gal/hr (20 cm^3/min) could be detected.

Another detector, the radiogas monitor, sampling air from the same position as the air particulate monitor, continuously analyzed air from the containment cooling system for gaseous radioactivity. This monitor was capable of detecting a leak rate of about 100 gal/hr (6500 cm^3/min).

In addition to measuring changes in the radioactivity of the containment vessel, dewpoint sensors continuously sampled the air from the suction side of the containment cooling system supply fans. These instruments could detect a primary coolant leak rate of approximately 4 gal/hr (250 cm^3/min) by measuring changes in the moisture content of the containment vessel.

By the use of the above instruments, plant operators could continuously monitor the containment vessel for primary system leakage and take any steps necessary to operate the facility safely. Measurements made by the New York University Medical Center, Institute of Environmental Medicine, have shown that the samples analyzed by these instruments are representative of the containment vessel and that samples take manually to backup these detectors were accurate to within a factor of 2.

Other methods for detecting and locating primary system leakage included visual inspection for escaping steam or water, boric acid crystal formation, component and primary relief tank levels, hydrogen concentration and radioactivity, containment sump level, and manual samples for tritium radioactivity in condensed moisture from the containment vessel.

6B.1 ASSUMPTIONS

1. Uniform mixing in containment occurs within 1 hr after a leak, based upon one containment cooling fan in service at 40,000 cfm.

2. The smallest significant changes instrumentation are as follows:
   a. Radiogas monitor on the containment cooling system
      1 count per sec is equivalent to 3×10^{-7} μCi/cm^3
   b. Particulate monitor 8 counts per sec is equivalent to 8×10^{-9} μCi/cm^3

c. Dewpoint 4°F

An 8-hr period was used to evaluate these changes; that provides time for checking instrumentation and determining the cause of the changes. The 8-hr evaluation period was predicted on determination of the magnitude of small leaks. Large leaks would of course be evaluated much sooner.

6B.2 BASIC DATA USED FOR CALCULATIONS

1. Sphere volume.

   1.8×10^6-ft^3 (5.05×10^{10} cm^3)

2. Sphere environment.

   a. Average temperature - 120°F
   b. Dewpoint - 70°F

3. Normal containment cooling radioactivity.

   a. Radiogas 2.5 counts per sec (7.5×10^{-7} μCi/cm^3)
   b. Particulate 16 counts per sec (1.6×10^{-3} μCi/cm^3)

4. Normal primary coolant radioactivity after 1 hr.

   a. Radiogas activity 5×10^{-3} μCi/ml H_2O
   b. Particulate 5×10^{-2} μCi/ml H_2O

6B.3 SAMPLE CALCULATIONS

1. Dewpoint in containment cooling system.

   a. At 120°F and 70°F dewpoint - the water content of the sphere would be 0.016 lb of water per pound of dry air

   b. At a dewpoint of 74°F the water content of the sphere would be 0.018 lb of water per pound of dry air let X = the leak rate into the sphere in gallons per hour

$$X = \frac{(0.018 - 0.016)\,\text{lb } H_2O / \text{lb dry air} \times 1.8 \times 10^6\,\text{ft}^3 \times 0.081 \times \frac{109}{121}\text{lb / ft}^3}{8\text{hrs} \times 8.3\text{lb / gal}}$$

= 3.95 gal/hr or 100 gal/day

2. Radioactivity in containment cooling system.

   a. Radiogas activity

      (1) Increase in activity 1 cps on installed monitor
      The radiogas activity increase = 3×10^{-7} μCi/cm^3

(2) Let Y = leak rate into a sphere in gallons per hour

$$Y = \frac{3.0 \times 10^{-7} \mu Ci / cm^3 air \times 5.05 \times 10^{10} cm^3 air}{8hr \times 5 \times 10^{-3} \mu Ci / ml \ H_2O \times 3.8 \times 10^3 ml / gal}$$

= 99.8 gal/hr or 2400 gal/day

b. Particulate activity in containment cooling system 8 counts per sec on the installed monitor
Radioactivity increase = 8 x 10^{-9} µCi/cm^3 air
Let Z = leak rate into the sphere in gallons per hour

$$Z = \frac{8 \times 10^{-9} \mu Ci / cm^3 air \times 5.05 \times 10^{10} cm^3 air}{8hr \times 5 \times 10^{-2} \mu Ci / ml \ H_2O \times 3.8 \times 10^3 ml / gal}$$

= 0.265 gal/hr or 6 gal/day

APPENDIX 6C
POSTACCIDENT CONTAINMENT ENVIRONMENT, REVISION 1

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Appendix 6C
POST ACCIDENT CONTAINMENT ENVIRONMENT

6C.1 DEFINITION OF POSTACCIDENT CONTAINMENT ENVIRONMENTAL CONDITIONS

As part of the initial license application, an evaluation of the suitability of materials of construction for use in the reactor containment system was performed considering the following:

1. The integrity of the materials of construction of engineered safeguards equipment when exposed to post-design-basis accident conditions.

2. The effects of corrosion and deterioration products from both engineered safeguards (vital equipment) and other (non vital) equipment on the integrity and operability of the engineered safeguards equipment.

Reference post-design-basis accident environment conditions of temperature, pressure, radiation, and chemical composition are described in the following sections. The time-temperature-pressure cycle used in the materials evaluation is most conservative, since it considers only partial safeguards operation during the design-basis-accident. The spray and core-cooling solutions considered here include both the design chemical compositions and the fission products, which may conceivably be transferred to the solution during recirculation through the various containment safeguards systems.

The original chemistry for the Containment Spray System utilized an alkaline adjusted sodium borate containment spray with the pH adjusted by sodium hydroxide. Use of solid Trisodium Phosphate for pH control of the solution has been used at a number of plants and was implemented at Indian Point 2 (IP2) in 1997. Reference 24 (WCAP-14542) discusses the benefits and justification for this change. In response to NRC Generic Letter 2004-02, Trisodium Phosphate (TSP) has been replaced by Sodium Tetraborate (STB) for pH control in order to reduce the risk of sump screen plugging due to the formation of chemical products. Replacement of TSP with STB was evaluated in WCAP-16596-NP (Reference 41) and concluded STB was the most comparable alternative to TSP and NaOH. This section was updated to incorporate this change and much of the updated information was drawn from WCAP-16596-NP. Updated references were also added as appropriate.

6C.1.1 Design-Basis Accident Temperature-Pressure Cycle

Figures 6C-1 and 6C-12 present the temperature-pressure-time relationship following the design-basis accident. These figures represent the containment condition for the following safety feature operation. One of the two spray pumps is considered to inject into the containment. When the refueling water storage tank is empty, the recirculation pumps can supply flow to the spray headers. Recirculation flow through one recirculation pump is cooled in the residual heat exchanger.

Figures 6C-2 and 6C-3 present materials evaluation test conditions for the containment and core environment, respectively.

Materials evaluations were performed, in general, for conditions either simulating the time-temperature conditions of Figure 6C-2 or conservatively considering higher temperatures for longer periods. The basis for each material evaluation is described with the discussion of its particular suitability.

6C.1.2 Design-Basis Accident Radiation Environment

The evaluation of materials for use in containment includes a consideration of the radiation stability requirements for the particular materials application. Figures 6C-4 and 6C-5 present the post-design-basis accident containment atmosphere direct gamma dose rate and the integrated direct gamma dose, respectively. These data were calculated on the basis of a core meltdown and assume the following fission product fractional releases, consistent with the TID-14844 model:

1. Noble gases, 1.0
2. Halogens, 0.5
3. Other isotopes, 0.01

6C.1.3 Design Chemical Composition Of The Emergency Core-Cooling Solution

Nuclear Regulatory Commission Branch Technical Position MTEB 6-1 (ref 31) specifies a minimum pH level of 7.0 of the postaccident emergency coolant water, operation higher in the pH 7.0 to 9.5 range for greater assurance that no stress corrosion cracking will occur, and if pH greater than 7.5 is used consideration should be given to hydrogen generation from aluminum.

The system designs provide for the use of alkaline-adjusted boric acid solution as the spray and core-cooling fluid. Initially the injection solution is not alkaline-adjusted since the RWST contains only boric acid and not STB. It is not until re-circulation from the sump, where the injected water has dissolved STB, that the spray and core-cooling fluid is alkaline adjusted.

Plant designs that use the spray solution for retention of fission product iodine in solution, as well as containment cooling, include provisions for addition of chemical additive to the emergency core cooling system. Originally, that additive was a concentrated sodium hydroxide solution but a number of plants have converted to dry Trisodium Phosphate granules and IP2 has since converted to Sodium Tetraborate granules. Boric acid solution, containing approximately 2600 ppm boron, is pumped from the refueling water storage tank to the containment system by means of the safety injection system pumps, residual heat removal pumps, and the spray pumps.

Granular Sodium Tetraborate is stored in baskets strategically located in the post-accident flooded region of the containment. Initially the containment spray will be boric acid solution from the refueling water storage tank which has a pH of approximately 4.5. As the initial spray solution and subsequently the recirculation solution comes in contact with the Sodium Tetraborate, the STB dissolves raising the pH of the sump solution to an equilibrium value between 7.0 and 7.6 (Reference 42, pg 18).

Based on Reference 42, 8096 pounds of Sodium Tetraborate Decahydrate is sufficient to assure a post-LOCA sump pH of 7.0 at 30 days with a margin of approximately 2 (in terms of strong acid addition) to reach a pH of 7.0 (Reference 42, Figure 2). Titration curves for TSP in Boric acid developed in CN-CDME-00-10 (Ref. 32) are shown in figure 6C-6. Based on these curves, 8000 pounds of Trisodium Phosphate Dodecahydrate ($Na_3PO_4 \cdot 12H_2O$) is sufficient to assure a post-LOCA sump pH of 7.0 (Ref. CN-CRA-96-005, Rev. 2) with a margin of approximately 41% to account for formation of acids over time in the solution. In addition the maximum pH due to Trisodium Phosphate is 7.61.

Reference 24A addresses the time required to submerge and dissolve the TSP and concludes that conservatively the dissolution times would be 1.7 hours for 6000 pounds of TSP and 2 hours for 8000 pounds. Realistically, the dissolution time would be considerably less. Dissolution testing performed in support of the buffer change to STB concluded that the rate of dissolution for STB was much faster than TSP (References 41 & 42).

The solutions are considered aerated through the entire exposure period.

6C.1.4 Trace Composition Of Emergency Core Cooling Solution

During spraying and recirculation, the emergency core cooling solution will wash over virtually all the exposed components and structures in the reactor containment. The emergency core cooling solution is recirculated through a common sump, and hence, any contamination deposited in or leached by the solution from the exposed components and structures will be uniformly mixed in the solution.

The materials compatibility discussion includes consideration of the effects of trace elements that are identified as conceivably being present in the emergency core cooling solution during recirculation.

To identify the trace elements in the containment that may have a deleterious effect on engineered safeguards equipment, one must first establish, which elements are potentially harmful to the materials of construction of the safeguards equipment, and second, ascertain the presence of these elements in forms, which can be released to the emergency core cooling solution following a design-basis accident. Table 6C-1 presents a listing of the major periodic groups of elements. Elements that are known to be harmful to various metals are noted and potential sources of these elements are identified. A discussion of the effects of these elements is presented in later sections.

6C.2 MATERIALS OF CONSTRUCTION IN CONTAINMENT

All materials in the containment are reviewed from the standpoint of ensuring the integrity of the equipment of which they are constructed and to ensure that deterioration products of some materials do not aggravate the accident condition. In essence, therefore, all materials of construction in the containment must exhibit resistance to the postaccident environment or, at worst, contribute only insignificant quantities of the trace contaminants that have been identified as potentially harmful to vital safeguards equipment. This must be true for these materials in both the new condition and for the aged condition at which the postaccident environment might be more likely to be encountered. In addition to the integrity of major components (e.g. piping, supports, vessels, containment structures, etc.), environmental qualification of Class 1E equipment must not be affected. Section 6C.9 addresses requalification of Class 1E equipment.

Table 6C-2 lists typical materials of construction used in the reactor containment system. Examples of equipment containing these materials are included in the table.

Corrosion testing, described in Section 6C.3, showed that of all the metals tested only aluminum alloys were found incompatible with the alkaline sodium borate solutions. Aluminum was observed to corrode at a significant rate with the generation of hydrogen gas. Since hydrogen generation can be hazardous to containment integrity, a detailed survey was conducted to identify all aluminum components in the containment.

Table 6C-3 lists the nuclear steam supply system aluminum inventory that is present in the reactor containment. Included in the table is the mass of metal and exposed surface area of each component. The 1100 and the 6000 series aluminum alloys are generally the major types found in the containment. This inventory reflects the determination to exclude as much as practicable the actual use of aluminum in the containment.

6C.3 CORROSION OF METALS OF CONSTRUCTION IN DESIGN-BASIS EMERGENCY CORE COOLING SOLUTION

Emergency core cooling components are primarily austenitic stainless steel and hence are quite corrosion resistant to the alkaline sodium borate solution as demonstrated by corrosion tests performed at Westinghouse and Oak Ridge National Laboratory.[1] The general corrosion rate, for type 304 and type 316 stainless steels was found to be 0.01 mils per month in pH 10 solution at 200°F. Data on corrosion rates of these materials in the alkaline sodium borate solution have also been reported by Oak Ridge National Laboratory [2,3] to confirm the low values.

Extensive testing was also performed on other metals of construction that are found in the reactor containment. Testing was performed on these materials to ascertain their compatibility with the spray solution at design postaccident conditions and to evaluate the extent of deterioration product formation, if any, from these materials.

Metals tested included Zircaloy, Inconel, aluminum alloys, cupro-nickel alloys, carbon steel, galvanized carbon steel, and copper. The results of the corrosion testing of these materials are reported in detail in Reference 1. Of the materials tested, only aluminum was found to be incompatible with the alkaline sodium borate solution. Aluminum corrosion is discussed in Section 6C.5. The following is a summary of the corrosion data obtained on various materials of construction exposed for several weeks in aerated, alkaline (pH 9.3 to 10.0) sodium borate solution at 200°F. The exposure condition is considered conservative since the test temperature (200°F) is considerably higher than the long-term design-basis accident temperature.

Material	Maximum Observed Corrosion Rate (mil/month)
Carbon Steel	0.003
Zircolay-4	0.004
Inconel 718	0.003
Copper	0.03
90-10 Cu-Ni	0.002
70-30 Cu-Ni	0.051
Galvanized carbon steel	0.031
Brass	-0.01

Tests conducted at Oak Ridge National Laboratory [2,3] have also verified the compatibility of various materials of construction with alkaline sodium borate solution. In tests conducted at 284, 212, and 130°F, stainless steels, Inconel, cupro-nickels, Monel, and Zircaloy-2 experienced negligible changes in appearance and negligible weight loss.

Corrosion tests at both Westinghouse and Oak Ridge National Laboratory have shown copper suffers only slight attack when exposed to the alkaline sodium borate solution at design-basis accident conditions. The corrosion rate of copper, for example, in alkaline sodium borate solution at 200°F is approximately 0.03 mils per month.[1] The corrosion of copper in an alkaline sodium borate environment under spray conditions at 284°F and 212°F have been reported by Oak Ridge National Laboratory. Corrosion penetration of less than 0.02 mil was observed after 24-hr exposure at 284°F (see Reference 3, Table 3.13) and a corrosion rate of less than 0.3 mils per month was observed at 212°C (see Reference 2, Table 3.6).

It can be seen therefore, that the corrosion of copper in the postaccident environment will have a negligible effect on the integrity of the material. Further, the corrosion product formed during exposure to the solution appears tightly bound to the metal surface and hence will not be released to the emergency core cooling solution.

The corrosion rate of galvanized carbon steel in alkaline sodium borate (3000 ppm boron, pH 9.3) is also low. Tests conducted in aerated solutions showed the corrosion rate to be 0.003 mils per month (0.046 mg/dm^2-hr) and 0.002 mils per month (0.036 mg/dm^2-hr) for temperatures of 200 and 150°F, respectively. Therefore, it can be seen that the corrosion of zinc (galvanized) in alkaline borate solution is minimal and will not contribute significantly to the postaccident hydrogen buildup.

Consideration was given to possible caustic corrosion of austenitic steels by the alkaline solution. Data presented by Swandby (Figure 6C-7) show that these steels are not subject to caustic stress cracking at the temperature (285°F and below) and caustic concentrations (less than one weight percent) of interest. It can be seen from Figure 6C-7 that the stress cracking boundary minimum temperature as defined by Swandby coincides with a high free caustic concentration (approximately 40-percent) and is considerably above (approximately 80°F) the long-term postaccident design temperature. Further, from Figure 6C-7 a temperature in excess of 500°F is required to produce stress corrosion cracking at a sodium hydroxide concentration greater than 85-percent.

6C.4 CORROSION OF METALS OF CONSTRUCTION BY TRACE CONTAMINANTS IN EMERGENCY CORE COOLING SOLUTION

Of the various trace elements that could occur in the emergency core cooling solution in significant quantities, only chlorine (as chloride) and mercury are adjudged potentially harmful to the materials of construction of the safeguards equipment.

6C.4.1 MERCURY

The use of mercury or mercury-bearing items, however, is restricted in the containment. This includes mercury vapor lamps, fluorescent lighting, and instruments that employ mercury for pressure and temperature measurements and for electrical equipment. The use of mercury is limited to the refueling cavity lights. Potential sources of mercury therefore, are generally excluded from the containment and hence, no hazard from this element is recognized.

The refueling cavity lights contain a small amount of mercury in the arc tube, enclosed in a quartz enclosure to preclude breakage due to thermal shock. An evaluation has demonstrated that neither the arc tube nor the outer quartz protective tube will break during a seismic event and they can withstand both the pressure and temperature expected during a loss-of-coolant accident.

6C.4.2 Chlorine

The possibility of chloride stress corrosion of austenitic stainless steels has also been considered. It is believed that corrosion by this mechanism will not be significant during the postaccident period for the following reasons.

6C.4.2.1 Low Temperature of Emergency Core Cooling Solution

The temperature of the emergency core cooling solution is reduced after a relatively short period of time (i.e., a few hours) to about 150°F. While the influence of temperature on stress corrosion cracking of stainless steel has not been unequivocally defined, significant laboratory work and field experience indicate that lowering the temperature of the solution decreases the probability of failure. Hoar and Hines[5] observed this trend with austenitic stainless steel in 42 weight percent solutions of $MgCl_2$ with temperature decrease from 310°F to 272°F. Staehle and Latanision[6] present data that also show the decreasing probability of failure with decreasing solution temperature from about 392°F to 302°F. Staehle and Latanision[6] also report the data of Warren[7] that showed the significant change with decrease in temperature from 212°F to 104°F. The work of Warren, while pertinent to the present consideration in that it shows the general relationship of temperature to time to failure, is not directly applicable in that the chloride concentration (1800 ppm Cl) believed to have effected the failure was far in excess of reasonable chloride contamination, which may occur in the emergency core cooling solution. More recent articles by Sedricks (ref 28), Moller (ref 29), and Macdonald & Cragnolino (ref 30) all state the importance of temperature as a variable in determining whether chloride stress corrosion cracking will occur but yet do not provide any definitive guidance. Jiang and Staehle (ref 34) correlate data from 17 references and conclude that for constant stress an Arrhenius form equation is reasonable for presenting the dependencies on t_f.

6C.4.2.2 Low Chloride Concentration of Emergency Core Cooling Solution

It is anticipated that the chloride concentration of the emergency core cooling solution during the post accident period will be low. Throughout plant construction, surveillance is maintained to ensure that the chloride inventory in the containment would be maintained at a minimum. Controls on use of chloride-bearing substances in the containment include the following:

1. Restriction in chloride content of water used in concrete.
2. Prohibition of use of chloride in cleaning agents for stainless steel components and surfaces.
3. Prohibition of use of chloride in concrete etching for surface preparation.
4. Use of non-chloride bearing protective coatings in containment.
5. Restriction in chloride concentration in safety injection solution, 0.15-ppm chloride maximum.

The effect of decreasing chloride concentration on decreasing the probability of failure of stressed austenitic stainless steel has been shown by many experimenters. Staehle and Latanision[6] present data of Staehle that show the decrease in probability of failure with decrease in chloride concentration at 500°F. Edeleanu[8] shows the same trend at chloride concentrations from 40 to 20-percent as $MgCl_2$ and reported no failures in this experiment at less than about 5-percent $MgCl_2$. Westinghouse corrosion tests (ref. 22) intended to simulate design basis accident conditions showed that crack initiation time increases with decreasing chloride concentration in tests.

Instances of chloride cracking at representative emergency core cooling solution temperatures and at low solution chloride concentration have generally been on surfaces on which concentration of the chloride occurred. In the emergency core cooling system, concentration of chlorides is not anticipated since the solution will operate subcooled with respect to the containment pressure and further, the containment atmosphere will be 100-percent relative humidity.

6C.4.2.3 Alkaline Nature of the Emergency Core Cooling Solution

The emergency core cooling solution will have a solution pH of greater than 7.0 after dissolution of the Sodium Tetraborate additive stored in the sumps. Numerous investigators have shown that increasing the solution pH decreases the probability of failure. Thomas et al.,[9] showed that the failure probability decreases with increasing pH of boiling solutions of $MgCl_2$. More directly applicable, Scharfstein and Brindley[10] showed that increasing the solution pH to 8.8 by the addition of NaOH prevented the occurrence of chloride stress corrosion cracking in a 10-ppm Chloride (as NaCl) solution at 185°F. Thirty stressed stainless-steel specimens including 304 as received, 347 as received, and 304 sensitized were tested. No failures were observed.

Other test runs by Schafstein and Brindley showed the influence of solution pH on higher chloride concentrations, up to 550 ppm Chloride; however, in these tests the pH-adjusting agents were either sodium phosphate or potassium chromate. The authors express the opinion, however, that in the case of the chromate solution, chloride cracking inhibition was simply because the hydrolysis yielded a pH of 8.8 and not because of an influence of the chromate anion. A similar hydrolysis will occur in the borate solution.

Studies conducted at Oak Ridge National Laboratory by Griess and Bocarella[11] on type 304 and type 316 stainless-steel U-bend stress specimens exposed to an alkaline borate solution (0.15M NaOH - 0.28M H_3BO_3) containing 100-ppm chloride (as NaCl) showed no evidence of cracking after 1 day at 140°C, 7 days at 100°C, 29 days at 55°C. These extreme test conditions, combined with the fact that some parts of the test specimens were subjected to severe plastic deformation and intergranular attack before exposure, show that the probability of chloride induced stress corrosion cracking in a postaccident environment is very low indeed.

Westinghouse corrosion tests (ref. 22) showed that at pH 7, 100 ppm Cl, sensitized and non-sensitized samples of 304 stainless steel cracked in approximately 7.5 and 10 months, respectively. Based primarily on those results, Branch Technical Position MTEB 6-1 (ref. 31) recommends that the minimum pH of the sump solution should be 7.0 and that the higher the pH, in the range 7 to 9.5, the greater the assurance that stress corrosion cracking will not occur.

As discussed in section 6C.1.3, the initial pH of the solution will be approximately 4 and will increase after the Sodium Tetraborate dissolves. Therefore, for some period of time the spray solution will be below 7.0. This period of time was conservatively estimated at 2.0 hours for Trisodium Phosphate (Reference 24) and would be less than that for Sodium Tetraborate (References 41 & 42). Westinghouse corrosion tests (Ref. 22) indicate that the minimum time to crack (100% crack of a 304 SS welded single U-bend) in a pH 4.5, 100 ppm Cl solution is 3 days and no cracking of any test materials was observed before 8 hours. Thus crack initiation occurred between 8 hours and three days. pH adjustment must occur prior to initiation of cracking. Hence, based on these results, it is necessary that the pH of the sump solution be raised above 7.0 within 8 hours.

Chlorides are not expected to instantaneously appear in the sump solution in concentrations sufficient to initiate cracking. The initial spray and safety injection solution is drawn from the refueling water storage tank where the chloride concentration is limited to 0.15 ppm. The Westinghouse tests indicate that crack initiation in boric acid with 0.4 ppm chloride and pH of approximately 4 requires extended exposure times (12 months in one example). Hence, cracking will not occur during the relatively brief spray and safety injection.

During recirculation, as the solution washes over the containment structures and components, chlorides and other contaminants will be removed from the surfaces and dissolved in the solution. Concrete is potentially a significant chloride source but is painted with a nuclear qualified coating which is expected to greatly impede chloride leaching. An extended time period will be required for chloride concentration to build up to critical concentrations (if they ever do). Since the time required to adjust the sump solution pH to greater than 7.0 by dissolution of Sodium Tetraborate is much less than 8 hours, pH adjustment will occur well before chloride concentrations have built up to a critical level.

In summary, therefore, it is concluded that exposure of the stainless steel engineered safety feature components to the emergency core cooling solution during the postaccident period will not impair its operability from the standpoint of chloride stress corrosion cracking. The environment of low temperature, low chlorides, and high pH that will be experienced during the postaccident period will not be conducive to chloride cracking.

6C.5 CORROSION OF ALUMINUM ALLOYS

Corrosion testing has shown that aluminum alloys are not compatible with alkaline borate solution. The alloys generally corrode fairly rapidly at the postaccident condition temperatures with the liberation of hydrogen gas. A number of corrosion tests were conducted in the Westinghouse laboratories (ref. 1, 23) and at Oak Ridge National Laboratory facilities. A review of applicable corrosion data is given in Table 6C-4 and on Figure 6C-8.

For purposes of the resolution of Generic Letter 2004-02 in regards to chemical effects on sump strainers, the methodology provided in WCAP-16530-NP-A was used for the prediction of the postulated chemical compounds produced in precipitate form from the corrosion of Aluminum and other materials in containment subject to sump and spray fluid. The results from the WCAP were employed, in conjunction with scaled strainer tests, to quantify head losses to be considered for the strainers when calculation NPSH for the Recirculation and RHR Pumps during the recirculation phase of a LOCA.

6C.5.1 Aluminum Corrosion Products In Alkaline Solution

The corrosion of aluminum in alkaline solution expected following a design-basis accident, has been shown to proceed with the formation of aluminum hydroxide[12-14] and the aluminate ion, as well as with the production of hydrogen gas.

The design-basis accident conditions expected for the Indian Point Unit 2 plant include the establishment of an alkaline emergency core cooling solution having a total volume of liquid of 4.47×10^5 gal after actuation of the engineered safety features.

As mentioned above, aluminum is known to corrode in alkaline solution to give a precipitate of $Al(OH)_3$, which in turn can redissolve in an excess of alkali to form a complex aluminate. Van Horn[12] noted that the precipitation of $Al(OH)_3$ begins about pH 4 and is essentially complete at

pH 7. A further increase in pH to about 9 causes dissolution of the hydroxide with the formation of the aluminate.

Therefore, it can be seen that the solubility of aluminum corrosion products is a function of the pH of the environment. Consistent with this, the corrosion of aluminum is also strongly dependent on the solution pH, because when the corrosion products are dissolved from the metal surface, corrosion of the base metal can proceed more freely.

Figure 6C-9 presents a plot of aluminum corrosion rate as a function of solution pH.[1] The corrosion rate of aluminum is seen to decrease by a factor of 21 (1/0.048) as the pH decreases from 9.3 to 8.3 and by a factor of 83 (1/0.012) as the pH decreases from 9.3 to 7.0.

Therefore, one must consider both corrosion and the dissolution of the corrosion products at specific reference conditions, since the two are directly related.

The corrosion reactions that are of interest in the design-basis accident condition here would include the reaction of aluminum in alkaline solution to form aluminum hydroxide, i.e.,

$$2\ Al + 6\ H_2O \Leftrightarrow 2\ Al\ (OH)_3 + 3\ H_2 \qquad (6C\text{-}1)$$

and dissolution of the hydroxide to form the aluminate, i.e.,

$$Al\ (OH)_3 \Leftrightarrow Al0_2^- + H^+ + H_2O \qquad (6C\text{-}2)$$

Knowledge of the solubility product of the aluminum hydroxide in an alkaline solution allows the determination the solubility expected for the hydroxide in the design-basis accident environment.

Deltombe and Pourbaix[15] have determined the solubility product of aluminum hydroxide. Using the value of 2.28×10^{-11} for K_{sp}, as reported by Deltombe and Pourbaix, the following calculation can be made.

The solubility of $Al\ (OH)_3$ is determined from Equation 6A-2

$$Al\ (OH)_3 \Leftrightarrow Al0_2^- + H^+ + H_2O \qquad (6C\text{-}3)$$

$$K_{SP} = (AlO_2^-)\ (H^+)$$

$$2.28 \times 10^{-11} = (AlO_2^-)\ (H^+)$$

at pH = 9.3

$$(AlO_2^-) = \frac{2.28 \times 10^{-11}}{5 \times 10^{-10}} = 4.6 \times 10^{-2} \text{ moles/liter} \qquad (6C\text{-}4)$$

Therefore, the solubility of $Al(OH)_3$ in a pH 9.3 solution at 25°C (77°F) is 4.6×10^{-2} moles per liter or 3.0×10^{-2} lb/gal. Expressed as aluminum, the solubility at these conditions is 1.05×10^{-2} lb/gal.

The solubility of the aluminum corrosion products in the postaccident environment is a function of both solution pH and temperature. Figure 6C-10 presents plots of the corrosion product solubility, expressed in terms of aluminum versus solution pH for temperatures of 77 and 150°F. The change in solubility with temperature is found using the relationship of the free energy of formation, temperature, and the solubility product.

With the data available from Figures 6C-9 and 6C-10 and a knowledge of the reference aluminum corrosion behavior for any specific plant, one can calculate the expected solubility limits for the corrosion reaction.

For Indian Point Unit 2, there are 4.47 x 10^5 gal of emergency core cooling solution after actuation of the safety features. The total amount of aluminum present in the Indian Point Unit 2 containment is given in Table 6C-3. Table 6C-5 shows the corrosion of aluminum with time for the original (NaOH additive) design basis pH 9.3 postaccident environment. CN-CRA-96-005 calculates a maximum pH of 7.61 with Trisodium Phosphate used for solution pH control and IP-CALC-07-00129 calculates a maximum pH of 7.6 with Sodium Tetraborate used for pH control.

Table 6C-6 presents a summary of the applicable solubility and corrosion parameters for various conditions. The table lists the applicable solubility products (K_{sp}) and solubilities at the various temperatures and solution pH together with the soluble aluminum limit for the Indian Point Unit 2 system at the specific conditions. The last values in the table given the aluminum solubility margin after 100 days of corrosion; that is, the soluble aluminum limit divided by the aluminum corroded. It can be seen that in all cases, including the very conservative low-temperature and low-pH conditions; the emergency core cooling solution is not expected to be saturated with aluminum corrosion products. Furthermore, within the expected design conditions for temperature and pH, the aluminum solubility margin ranges from approximately 20 to 106.

The preceding analysis is based on the original design basis with NaOH addition and a pH of 8.5 to 10.0 in the solution. Use of Trisodium Phosphate or Sodium Tetraborate reduces the long term pH to a minimum of 7.0 and a maximum of approximately 7.6 (Ref. 27 & 42). Figures 6C-9 and 6C-10 show significant (orders of magnitude) decreases in the corrosion rate and solubility of aluminum when the pH is reduced from the 9.3 range to the low 7 range. This is also shown in reference 23 (WCAP-8776) for Trisodium Phosphate. Thus the corrosion rate of aluminum and the production of hydrogen due to that corrosion can be expected to significantly decrease in the post accident environment due to replacement of the NaOH pH control additive by Trisodium Phosphate. Although corrosion of aluminum is greater for Sodium Tetraborate it is not excessive (Ref. 41).

It is concluded therefore, that the corrosion products of aluminum will be in the soluble form during the postaccident period considered and hence, there is no potential for deposition on flow orifices, spray nozzles, or other equipment.

6C.5.2 Behavior Of Circulating Aluminum Corrosion Products

The solubility of aluminum corrosion products has shown that for this plant, the entire inventory produced after 100 days of exposure to the post-design-basis accident condition would remain in solution. The review also indicates that the emergency core cooling solution is only approximately 17-percent saturated at 77°F and less than 1-percent saturated at 150°F.

However, it is of interest to review the experience of facilities that have operated with insoluble aluminum corrosion products and to relate their conditions with those expected in the postaccident environment.

The most significant experience available is that of Griess[16] who operated a recirculating test facility to measure the corrosion resistance of a variety of materials in alkaline sodium borate spray solution.

Tests were conducted on 1100, 3003, 5052, and 6061 aluminum alloys exposed at 100°C in pH 9.3 sodium borate solution (0.15M NaOH - 0.28M H_3BO_3). It was reported that even though the solution contained copious amounts of flocculent aluminum hydroxide, it had no effect on flow through the spray nozzle (0.093-in. orifice). The pH of the solution did not change because of the increase in the corrosion products.

Griess[16] in describing his observations regarding aluminum corrosion product deposition potential stated that:

1. No significant deposition was observed on the cooling coil installed in the solution.
2. No significant deposition was observed on the heated surfaces of the facility.
3. No significant deposition was observed on isothermal facility surfaces.

The amounts of aluminum corroded to the solution in the tests conducted by Griess at 55 and 100°C were approximately 4.0 and 18.6 g, respectively. The concentration of aluminum present in the recirculation stream, therefore, was approximately 0.2 and 1 g/liter, respectively. This value is a factor of about 5 above the aluminum concentration expected in the postaccident emergency core cooling solution at Indian Point Unit 2 in a pH 9.3 (NaOH additive) solution after 100 days, and many times that when the lower pH Trisodium Phosphate additive is considered. Although corrosion of aluminum is greater for Sodium Tetraborate it is not excessive (Ref. 41).

Hatcher and Rae[17] describe the appearance of turbidity in the NRU reactor and "purpose" that deposition of aluminum corrosion products may have occurred on heat exchanger surfaces, although they do not report any specific examination results. Moreover, Hatcher and Rae report no operations problems associated with the presence of aluminum corrosion product turbidity in the NRU reactor. The overall heat transfer coefficient for each NRU reactor heat exchanger was measured after 2 yr. of full-power operation on several occasions and within the limit of accuracy of the measurements, reported at approximately 5-percent, no change in the thermal resistance had been observed.

It is concluded, therefore, both from the work of Griess and Hatcher and Rae, that the deposition of aluminum corrosion products on heat exchangers surfaces will not be significant in the postaccident environments even for the circumstances of insoluble product formation.

6C.6 COMPATIBILITY OF PROTECTIVE COATINGS WITH POSTACCIDENT ENVIRONMENT

The investigation of materials compatibility in the postaccident design basis environment also included an evaluation of protective coatings for use in containment.

The results of the protective coatings evaluation presented in WCAP-7198[18], showed that several inorganic zincs, modified phenolics, and epoxy coatings are resistant to an environment of high temperature (320°F maximum test temperature) and alkaline sodium borate. Long-term

tests included exposure to spray solution at 150°F to 175°F for 60 days, after initially being subjected to the conservative design-basis accident cycle shown in Figure 6C-3. The protective coatings, which were found to be resistant to the conditions, that is, exhibited no significant loss of adhesion to the substrate nor formation of deterioration products, comprise virtually all of the protective coatings recommended for use in the containment. Hence, the protective coatings will not add deleterious products to the core-cooling solution.

It should be pointed out that several test panels of the recommended types of protective coatings were exposed for two design-basis accident cycles and showed no deterioration or loss of adhesion with substrate.

In July 1973, Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants", (35) was issued to describe an acceptable method for complying with the NRC's quality assurance requirements with regard to protective coatings applied to ferritic steels, stainless steel, zinc-coated (galvanized) steel, concrete, or masonry surfaces of water-cooled nuclear power plants. Revision 1 of Regulatory Guide 1.54 was issued in 2000. Plants whose design basis that includes a commitment to Regulatory Guide 1.54 require that protective coatings be qualified and capable of surviving a design basis accident without adversely affecting safety related structures, systems, or components needed to mitigate the accident. NRC Generic Letter 98-04 (Ref 36) identified the potential for degradation and failure of "qualified" protective coatings applied to exposed surfaces within nuclear power plant containments. The NRC sponsored work at the Savannah River Technology Center to investigate the performance and potential for failure of Service Level 1 coating systems used in nuclear power plant containments (37). The Nuclear Energy Institute has issued "Condition Assessment Guideline: Debris Sources Inside PWR Containments" (38) that provides guidance to PWR operators in assessment of condition of coatings (among other items).

6C.7 EVALUATION OF THE COMPATIBILITY OF CONCRETE AND EMERGENCY CORE COOLING SOLUTION IN THE POST ACCIDENT ENVIRONMENT

Concrete specimens were tested in boric acid and alkaline sodium borate solutions at conditions conservatively (320°F maximum and 200°F steady-state) simulating the post design-basis accident environment.

The purpose of this study was to establish:

1. The extent of debris formation by solution attack of the concrete surfaces.
2. The extent and rate of boron removal from the emergency core cooling solution through boron-concrete reaction.

Tests were conducted in an atmospheric pressure, reflux apparatus to simulate long-term exposure conditions and in a high-pressure autoclave facility to simulate the design-basis accident short term, high-temperature transient.

For these tests the total surface area of concrete in the design containment, which may be exposed to the emergency core cooling solution following a design-basis accident was estimated at 6.3×10^4-ft^2. This value includes both coated and uncoated surfaces. The emergency core cooling solution volume for a reference plant was considered at approximately 313,000 gal and the surface to volume ratio from these values is approximately 29 in.^{2}/gal. The surface to volume ratios for the concrete-boron tests used were between 28 and 78 in.^{2}/gal of solution. Table 6C-7

solution. Table 6C-7 presents a summary of the data obtained from the concrete-boron test series.

Testing of uncoated concrete specimens in the postaccident environment showed that attack by both boric acid and the alkaline boric solution is negligible and the amount of deterioration product formation is insignificant. Other specimens covered with modified phenolic and epoxy protective coatings showed no deterioration product formation. These observations are in agreement with Orchard[19] who lists the following resistances of portland cement concrete to attack by various compounds:

1. Boric acid, little or no attack.
2. Alkali hydroxide solution under 10-percent, little or no attack.
3. Sodium borate, mild attack.
4. Sodium hydroxide over 10-percent, very little attack.

Exposure of uncoated concrete to spray solution between 320°F and 210°F has shown a tendency to remove boron very slowly, presumably precipitating an insoluble calcium salt. The rate of change of boron in solution was measured at about 130 ppm per month with pH 9 solution at 210°F for an exposed surface of about 36 in.^{2}/gal of solution (much greater than any potential exposure in the containment). The boron loss during the high-temperature transient test (320°F maximum) was about 200 ppm. Figure 6C-11 shows a representation of the boron loss from the emergency core cooling solution versus time by a boron-concrete reaction following a design-basis-accident. The time period from 0 to 6 hr shows the loss during a conservative high-temperature transient test, ambient to 320°F to 285°F. The data from 6 hr to 30 days is base on 210°F data.

A depletion of boron at this rate poses no threat to the safety of the reactor because of the large shutdown margin and the feasibility of adding more boron solution should sample analysis show a need for such action.

Griess and Bacarella (ref. 21) report on tests of concrete cylinders poured from the actual construction mix at Browns Ferry Nuclear Plant in a simulated post accident environment. None of the cylinders underwent any visible changes and in all cases the strength of the concrete after exposure exceeded the design strength. Solutions in these tests picked up about 10 ppm chloride (leached from the concrete) and the boron concentration was reduced by about 10%. However, the ratio of surface area to solution volume was very high (240 in^2/gal), about 10 times higher than the ratio calculated above, and the results are therefore substantially greater than would be expected in the actual containment.

6C.8 MISCELLANEOUS MATERIALS OF CONSTRUCTION

6C.8.1 Sealants

Candidate sealant materials for use in the reactor containment system were evaluated in simulated design-basis accident environments. Cured samples of various sealants were exposed in alkaline sodium borate solution, pH 10.0, 3000 ppm to a maximum temperature of 320°F.

Table 6C-8 presents a summary of the sealant materials tested, together with a description of the panel's appearance after testing. Three generic type of sealants were tested: butyl rubber, silicone, and polyurethane. Each of the materials was the "one package" type, that is, no mixing

of components was necessary prior to application. The materials were applied on stainless steel allowed to cure well prior to testing.

The test results showed that the silicone sealants tested were chemically resistant to the design-basis accident environment and are acceptable for use in containment. Sealant 780 by Dow Corning Corporation would be acceptable for use in the containment. Major applications of this sealant could be as concrete expansion joint sealant on the liner insulation panels. Sealant 780 will contribute no deterioration products to the emergency core cooling solution during the post design-basis accident period and will maintain its structural integrity and elastic properties.

6C.8.2 Polyvinyl Chloride Protective Coating

Tests were conducted to determine the stability of the polyvinyl chloride (PVC) protective coating, of the type, which might be used on conduit in the design-basis accident environment. Samples of the polyvinylchloride exposed to alkaline sodium borate solutions at design-basis accident conditions showed no loss in structural rigidity and no change in weight or appearance.

A sample of polyvinylchloride coated aluminum conduit (1-in. OD x 8-in. long) was irradiated by means of a Co-60 source, at an average dose rate of 3.2×10^6 rads/hr, to a total accumulated dose of 9.1×10^7 rads. The specimen was immersed in alkaline sodium borate solution (pH 10, 3000 ppm boron) at 70°F. Visual examination of the coating after the test showed no evidence of cracking, blistering, or peeling; and the specimen appeared completely unaffected by the gamma exposure. Chemical analysis of the test solution indicated that some bond breakage had occurred in the polyvinyl-chloride coating as evidenced by increase in the chloride concentration. The gamma exposure of approximately 10^8 rad resulted in a release to the solution of 26 mg of chloride per square foot of exposed polyvinylchloride surface. Considering a total surface area of polyvinylchloride coating present in containment (approximately 500-ft^2) and the emergency core cooling solution volume of 313,000 gal, the chloride concentration increase in the emergency core cooling solution due to irradiation of the coating, would be approximately 0.01 ppm.

Therefore, it is concluded that polyvinylchloride protective coating will be stable in the design basis accident environment.

6C.8.3 Fan Cooler Materials

Samples of the following air-handling system materials were exposed in an autoclave facility to the design-basis accident temperature-pressure cycle:

1. Moisture separator pad.
2. High efficiency particulate media.
3. Asbestos separator pads.
4. Adhesive for joining separator pads and high-efficiency particulate air filter media corners.
5. Neoprene gasketing material.

The materials were exposed in both the steam phase and liquid phase of a solution of Sodium Tetraborate (15ppm boron) to simulate the concentrations expected down stream of the fan-cooler coils. Examination of the specimens after exposure showed the following:

1. Moisture separator pads were somewhat bleached in color but maintained their structural form and showed good resiliency as removed in both liquid and steam phase exposure.

2. High-efficiency particulate filter media maintained its structural integrity in both the liquid and steam phase. No apparent change.

3. Asbestos separator pads showed some slight color bleaching. How ever, both steam and liquid phase samples maintained their structural integrity with no significant loss in rigidity.

4. Adhesive material for the HEPA/separator pad edges showed no deterioration or embrittlement and maintained its adhesive property.

5. Neoprene gasketing material is also satisfactory in both the steam and liquid phase. The material showed only weight gain and a shrinkage of 15 to 30-percent based on a superficial, one flat side area. The gasket thickness decreased about 10-percent. The gasket material was unrestrained during the exposure, and hence the dimensional changes experienced are greater than those which would result as installed in the air-handling side of the fan coolers.

6C.8.4 Polyvinyl Chloride Insulation

The containment liner is insulated with a polyvinyl chloride (PVC) insulation enclosed in a stainless steel jacket. Table 14.3-40 lists 6940 ft^2 of 1.25 inch thick PVC insulation enclosed with 0.019 inch jacket and 7434 ft^2 of 1.5 inch thick PVC insulation enclosed with a 0.025 inch jacket. This totals about 1652 ft^3 of PVC installation for this application. Approximately 25 ft^2 of the 1.25 inch thick insulation was replaced with fiberglass insulation. Materials with high chloride content (like PVC) are subject to thermal and/or radiological degradation and are not normally left in containment when a plant returns to power. This insulation material is enclosed in stainless steel jackets which if intact and containing no penetrations should keep the spray solution from contacting the insulation or its degradation products. However, if access paths in the jackets allow sump solution or spray to come in contact with the insulation material, it might be a large source of chlorides.

PVC degradation releases HCL which dissolves in water to form hydrochloric acid. Properties of PVC have been reported to change at ~1.9×10^7 rads (ref 39) and classified in reference 40 as having excellent resistance to radiation (noticeable changes in properties occur above 10^7 rads). Reference 40 also indicates that polymers are sensitive only to total radiation dose and not to dose rate so that the dose accumulated to the accident must also be considered. That reference also lists a continuous high temperature limit of 75 °C (167°F) for PVC. Degradation of PVC has caused failure of a number of components at nuclear power plants as reported in various NRC reports.

6C.9 ENVIRONMENTAL REQUALIFICATIONS

Qualification of electrical equipment for harsh environment is discussed in Section 7.1. The impact of changing the post accident spray solution pH control chemical from sodium hydroxide injected from the spray additive tank to Trisodium Phosphate allowed to dissolve in the solution collected in the sump on Westinghouse supplied Class 1E equipment was evaluated and

documented in WCAP-14495 (ref. 25). Some equipment installed at IP2 was of an older vintage, not qualified in accordance with WCAP-8587 (ref 33), or was provided and/or qualified by a vendor other than Westinghouse and was therefore outside of the scope of the WCAP-14495. A number of additional non-metallic materials, not considered here, contained in the Class 1E electrical equipment are also considered in that evaluation. That evaluation concluded that "spray additive tank elimination will not affect the qualification of Westinghouse supplied Class 1E equipment located inside containment at Indian Point Nuclear Generating Station Unit No. 2 (IP2) However, the Notes at the bottom of Table 2.0-1 states that "Evaluation of this equipment is representative of equipment installed at IP2. Additional evaluation will be necessary to affirm applicability/completeness for equipment installed at IP2".

An evaluation (Reference 43) was performed to determine the impact of the change in post-LOCA buffered sump chemistry on the existing EQ equipment qualified using Trisodium Phosphate. The evaluation concluded that due to the similarities between post-LOCA Trisodium phosphate and Sodium Tetraborate buffered sump solutions; the equipment qualified for Trisodium Phosphate remains qualified using the new Sodium Tetraborate buffered solution. Therefore, there would be no impact on existing IP2 EQ equipment as a result of the subject post-LOCA buffered sump chemistry change.

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TABLE 6C-1 (Sheet 1 of 2)
Review of Sources of Various Elements in Containment
and Their Effects on Materials of Construction

Group	Representative Elements	Corrosivity of Elements	Sources of Elements
0	He, Ne, Kr, Xe	No effect on any materials construction	Fission product release
I a	Li, Na, K	Generally corrosion inhibitive properties for steels and copper alloys - harmful to aluminum	Li - coolant pH-adjusting agent Na - spray additive solution, concrete leach product K - concrete leach product
II a	Mg, Ca, Sr, Ba	Generally not harmful to steel or copper base alloys	Concrete leach products - deteriorated insulation
III a	Y, La, Ac	Not considered harmful in low concentrations	Fission product release
IV a	Ti, Zr, Hf	Not considered harmful to any materials	Fuel rod cladding, control rod material, alloying constituent
V a	V, Nb, Ta	Not considered harmful to any materials	Alloying constituents in low concentration
VI a	Cr, Mo, W	Not considered harmful to any materials	Alloying constituents in equipment
VII a	Mn, Tc, Re	Not considered harmful	Mn - alloy constituent

TABLE 6C-1 (Sheet 2 of 2)
Review of Sources of Various Elements in Containment
and Their Effects on Materials of Construction

Group	Representative Elements	Corrosivity of Elements	Sources of Elements
VIII	Fe, Ni, Cr, Os	Fe, Ni, Cr - not harmful to any materials	Fe, Ni, Cr - alloying constituents. Others have no identifiable sources
I b	Cu, Ag, Au	Not harmful to any materials	Cu present as material of construction and alloying constituent
II b	Zn, Cd, Hg	Hg - harmful to stainless steel, Cu alloys, aluminum Zn - unknown Cd - unknown	Hg has been entirely excluded from use in the containment; Cd finish plating on components. Zn galvanizing and alloying constituent
III b	B, Al, Ga, In	Not harmful to material	B - neutron poison additive Al - materials of construction
IV b	C, Si, Sn, Pb	C, Si, Sn not harmful to materials Pb considered harmful to nickel alloys	Si - concrete leach product Pb- alloy constituent in some brazes
V b	N, P, As, Sb, Bi	No effect from N unless ammonia is formed; others unknown	N - containment air;Others not identified in significant materials
Group	Representative Elements	Corrosivity of Elements	Sources of Elements
VI b	O, S, Se, Te	S possibly harmful to nickel alloys	Te - fission product S - oils, greases, insulating materials
VII b	F, Cl, Br, I	F considered potentially harmful to Zircaloy Cl potentially harmful to stainless steel Br and I not generally harmful	Cl - concrete leach product, general contamination F - organic materials I and Br - fission products, low concentration

TABLE 6C-2
Materials of Construction in Reactor Containment

Material	Equipment Application
300 series stainless steel	Reactor coolant system, residual heat removal loop, spray system
400 series stainless steel	Valve materials
Inconel (600, 718)	Steam generator tubing, reactor vessel nozzles, core supports, and fuel rod grids
Galvanized steel	Ventilation duct work, CRDM shroud material, I and C conduit
Aluminum	Nuclear detectors, I and C equipment, CRDM connectors, paints
Copper	Service water piping, fan cooler material
70-30 Cu Ni	Fan cooler material
90-10 Cu Ni	Fan cooler material
Carbon steel	Component cooling loop, structural steel, main steam piping, etc.
Monel	Possibly instrument housings
Brass	Possibly instrument housings
Polyvinylchloride	Conduit sheathing, electrical insulation, containment liner insulation
Protective coatings	General use on carbon steel structures and equipment, concrete
Inorganic zincs Epoxy Modified phenolics	
Silicones - neoprene	Ventilation duct work gasketing, sealants

TABLE 6C-3
Inventory of Aluminum in Containment

	Item	Mass (lb)	Surface Area (ft^2)
1.	CRDM connectors	122	42
2.	Reactor vessel insulation foil	269	Very high
3.	Area monitors	6	4
4.	Source, intermediate, and power range detectors	140	40
5.	Process instrumentation and control	420	84
6.	Lighting fixtures and equipment	1061	380
7.	Paint on steam generator, pressurizer and reactor vessel	140	Very high
8.	Contingency	250	85

TABLE 6C-4
Corrosion of Aluminum Alloys in Alkaline Sodium Borate Solution

Data Point	Temperature (°F)	Alloy Type	Test Duration	Corrosion Rate mg/dm2/hr	pH	Exposure Condition	Reference
1	275	5053	3 hr	96.2	9	Solution	WCAP-7153, Table 9
2	275	5005	3 hr	840	9	Solution	WCAP-7153, Table 9
3	200	6061	320 hr	15.4	9.3	Solution	WCAP-7153, Table 8 WCAP-7153, Figure 9
4	210	5052	7 days	53.0	9	Solution	WCAP-7153, Table 7 WCAP-7153, Figure 8
5	210	5052	2 days	14.0	9	Solution	WCAP-7153, Table 5
6	210	5005	2 days	27.1	9	Solution	WCAP-7153, Table 5
7	284	5052	1 day	54	9.3	Spray	ORNL-TM-2425 Table 3.13
8	284	5052	1 day	31.5	9.3	Solution	ORNL-TM-2425 Table 3.13
9	212	6061	3 days	126	9.3	Spray	ORNL-TM-2368 Table 3.6
10	212	6061	3 days	110	9.3	Solution	ORNL-TM-2368, Table 3.6
11	150	6061	7 days	2.9	9.3	Solution	Westinghouse data
12	150	5052	7 days	4.2	9.3	Solution	Westinghouse data

TABLE 6C-5
Corrosion Products of Aluminum Following Design Basis Accident, Indian Point Unit 2

Time After Reactor Trip (days)	Mass of Aluminum Corroded ($lb \times 10^{-2}$)	Hydrogen Produced ($SCF \times 10^{-3}$)	Mass of $Al(OH)_3$ Formed ($lb \times 10^{-2}$)
1	1.71	3.41	4.94
5	4.31	8.60	12.4
10	4.50	8.98	13.0
20	4.88	9.75	14.1
30	5.26	10.5	15.2
40	5.66	11.3	16.4
50	6.06	12.1	17.5
60	6.41	12.8	18.5
70	6.81	13.6	19.7
80	7.21	14.4	20.9
90	7.61	15.2	22.0
100	7.97	15.9	23.0

TABLE 6C-6
Summary of Unit 2 Aluminum Corrosion Product Solubility Data

	Solution Temperature			
	77°F		150°F	
Parameter	pH 9.3	pH 8.3	pH 9.3	pH 8.3
Solubility product K_{sp}	2.28×10^{-11}	2.28×10^{-11}	4.16×10^{-10}	4.16×10^{-10}
Aluminum solubility, lb Al/gal	1.05×10^{-2}	1.05×10^{-3}	1.9×10^{-1}	1.9×10^{-2}
Soluble aluminum Limit[1] for emergency core cooling system, lb	4.69×10^{3}	4.69×10^{2}	8.49×10^{4}	8.49×10^{3}
Aluminum corrosion rate (normalized)	(Not used)	(Not used)	1	0.048
Aluminum corroded after 100 days, lb	(Not used)	(Not used)	795	438[2]
Aluminum solubility margin at 100 days	5.9[3]	1.1[3]	106	19

1. Indian Point Unit 2 solution volume 4.48×10^{5} gal.

2. Value assumes rapid corrosion of all aluminum paint and reactor vessel foil insulation.

3. Note corrosion rate at 150°F was used for “aluminum corroded” value; hence, value is very conservative.

TABLE 6C-7
Concrete Specimen Test Data

Concrete Boron Test No.	Total Exposure Period (Days)	Surface/ Volume (in.^{2}/gal)	Exposed Weight Change (Grams)	Initial Specimen Weight (Grams)	Visual Examination
1	24	28	-22.4	560.0	No apparent change
3	28	20	+21.5	404.0	Light, yellowish, deposition specimen
4[1]	72	38	0	641.2	No apparent change - coating adhesion excellent
5	72	43	-0.2	769.5	Light, hard deposit on specimen
6	-4[2]	54	-	601.4	No apparent change - small amount of sand particles in test can
7	175	23	+11.0	457.0	No apparent change
8[1]	175	38	+26.5	751.0	No apparent change - coating adhesion excellent
9[1]	-5[2]	78	+4.0	702.0	No apparent change - coating adhesion excellent

1. These specimens coated with Phenoline 305. All others were uncoated.

2. These tests were at high-temperature design-basis accident transient conditions. All others at 195°F - 205°F.

TABLE 6C-8
Evaluation of Sealant Materials For Use In Containment

Sealant Type	Manufacturer	Posttest Appearance
Butyl rubber	A	Unchanged, flexible.
Silicone	B	Unchanged, flexible.
Silicone	B	Unchanged, flexible
Polyurethane	C	Sealant bubbled and became very soft; solution permeated into bubbles.
Polyurethane	C	Sealant swelled and became soft; solution permeated into material.
Polyurethane	C	Sealant swelled and became very soft and tacky; solution permeated into material.

6C FIGURES

APPENDIX 6D
SPRAY SYSTEM MATERIALS COMPATIBILITY FOR LONG-TERM STORAGE OF SODIUM HYDROXIDE

(RETAINED FOR HISTROICAL PURPOSES)

SPRAY SYSTEM MATERIALS-COMPATIBILITY FOR LONG-TERM STORAGE OF SODIUM HYDROXIDE

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Appendix 6D

SPRAY MATERIALS COMPATIBILITY FOR LONG-TERM STORAGE OF SODIUM HYDROXIDE

[Historical Information] A materials compatibility review for the spray additive tank and associated equipment during long-term storage of sodium hydroxide is presented below. The exposure conditions are shown in Table 6D-1, and the materials for the various components are shown in Table 6D-2. The corrosion rates for the various materials at or near the long-term exposure conditions with air contamination are shown on Table 6D-3. The immunity of most of the materials in Table 6-2 to caustic cracking at the exposure conditions listed in Table 6D-1 has been reported by Logan[1] (see Figure 6D-1). No caustic cracking of 17-4 pH or Stellite has been reported.[2]

The effect of carbon dioxide from air exposure on the corrosion of iron is shown in Figure 6D-2.[3] At pH 14, no additional corrosion is observed over that observed in carbon-dioxide-free solution. In the Indian Point Unit 2 system, a nitrogen blanket is continuously maintained over the sodium hydroxide solution in the spray additive tank, thus essentially eliminating any carbon dioxide contamination of the solution.

The Nordel* rubber diaphragm material used in the tank valves was exposed in 33 wt percent sodium hydroxide solution at 110°F for 6 months and found to be unaffected by the simulated spray additive tank solution. The integrity of the structural materials in the spray additive tank system would not be adversely affected even using the corrosion rate presented in Table 6D-3 where air contamination is present. In the Indian Point Unit 2 system, where nitrogen blanketing of the spray additive tank would prevent air contamination, the corrosion rates would be even lower with even less effect on the material integrity.

Diamond Shamrock Company[4] reported that no galling of steel valves occurred after exposure to 50-percent sodium hydroxide at 120° to 140°F for greater than 3 years. Stainless steel valves, exhibiting lower corrosion rates, would have an even lower propensity toward galling than steel. Therefore, no galling should occur on the valves exposed to the long-term storage conditions. The total corrosion product released to the spray additive tank as oxide would be less than 1000 g/yr with aerated solution and would be much less with the air-free solution (i.e., the Indian Point Unit 2 solution). This small quantity of corrosion product should not present any problems with clogging of delivery lines.

No sodium hydroxide precipitation will occur for a 30 wt percent solution if the temperature of the tank and liners is maintained above 35°F. Since this system is located in an area of the auxiliary building, which is continuously heated to maintain a 50°F minimum temperature, no solid sodium hydroxide would be present and, therefore, no clogging of the lines could occur.

* Nordel is a product of E.I. Du Pont de Nemours and Company.

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TABLE 6D-1
Exposure Conditions

Temperature, ^{o}F	110
Nitrogen overpressure	Slight positive pressure
Sodium hydroxide concentration, w/o	30
Oxygen concentration, normal	Nitrogen blanketed
Carbon dioxide concentration, normal	Nitrogen blanketed

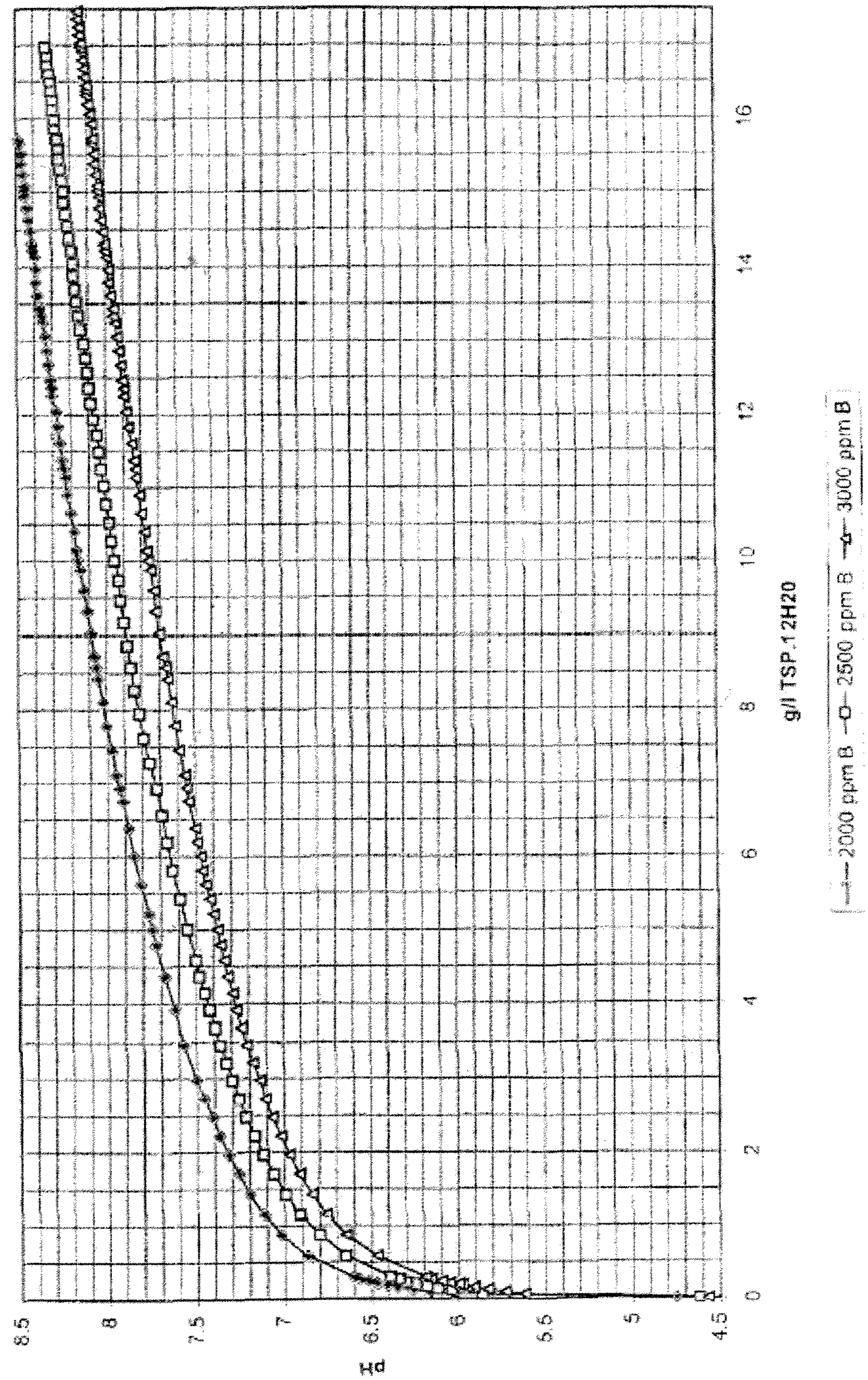
TABLE 6D-2
Component Materials

Spray additive tank	304 stainless steel cladding on steel A-516 GR-70
Piping	304 stainless steel
Valve bodies	304 and 316 stainless steel
Valve seats	Austenitic stainless steel or Stellite
Valve stems	17-4 pH and 410 stainless steel
Valve diaphragm	Ethylene-propylene dipolymer (Nordel rubber by DuPont)

TABLE 6D-3
Corrosion Rates

Material	Temperature (^{o}F)	NaOH Concentration (ppm)	Aeration	Corrosion Rates (mils/yr)	Reference No.
304 S/S	136	22 to 50	Yes	<0.1	1
316 S/S	125	30	Yes	<2	2
Steel	179	30 to 50	Yes	<20	2
410 S/S	125	30	Yes	<2	2
17-4 pH	176	30	Yes	3 to 6	7
Stellite	150	50	Yes	<0.6	4
Nordel rubber	110	33	Yes	<0.004	5

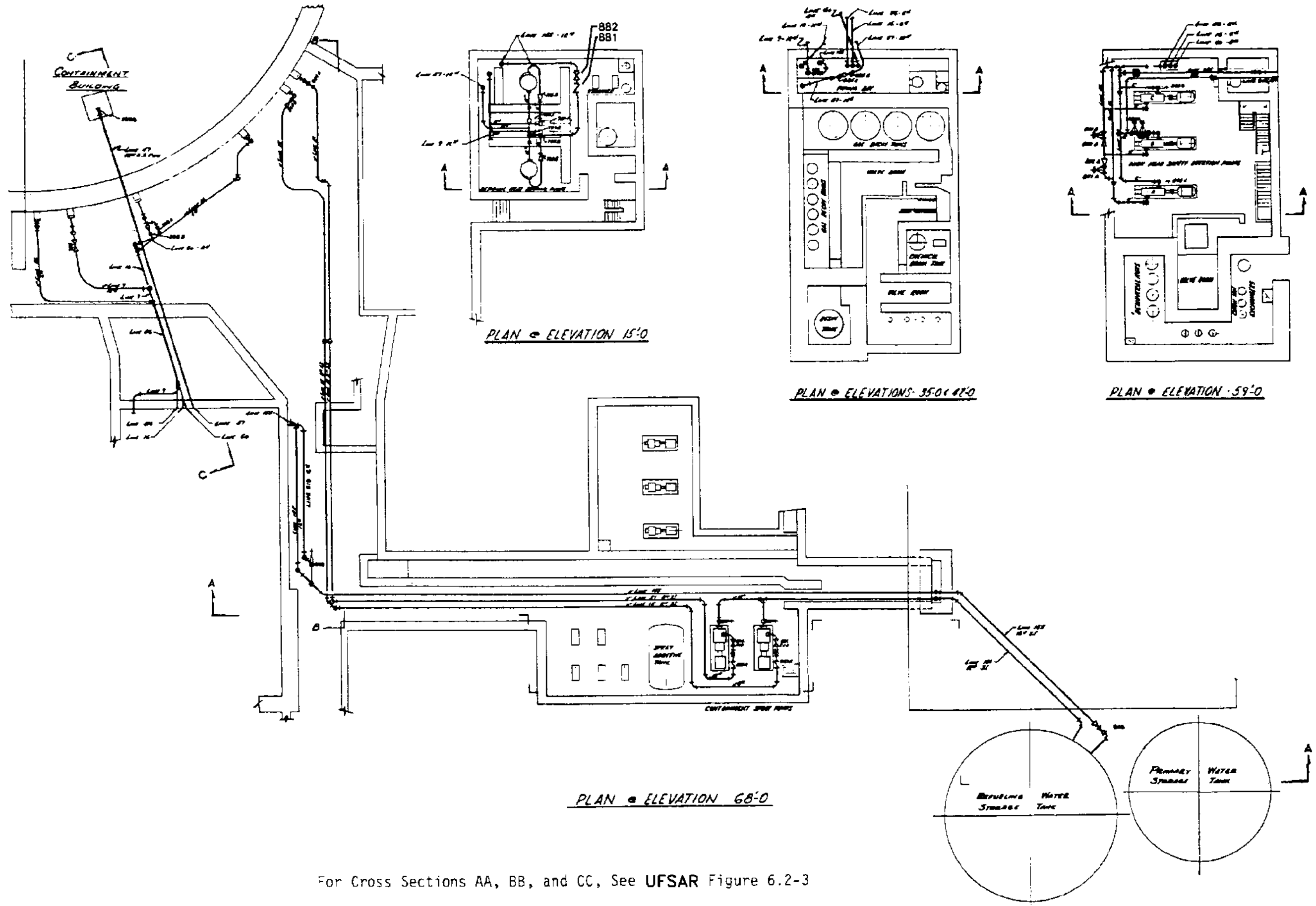
6D FIGURES



INDIAN POINT UNIT No. 2

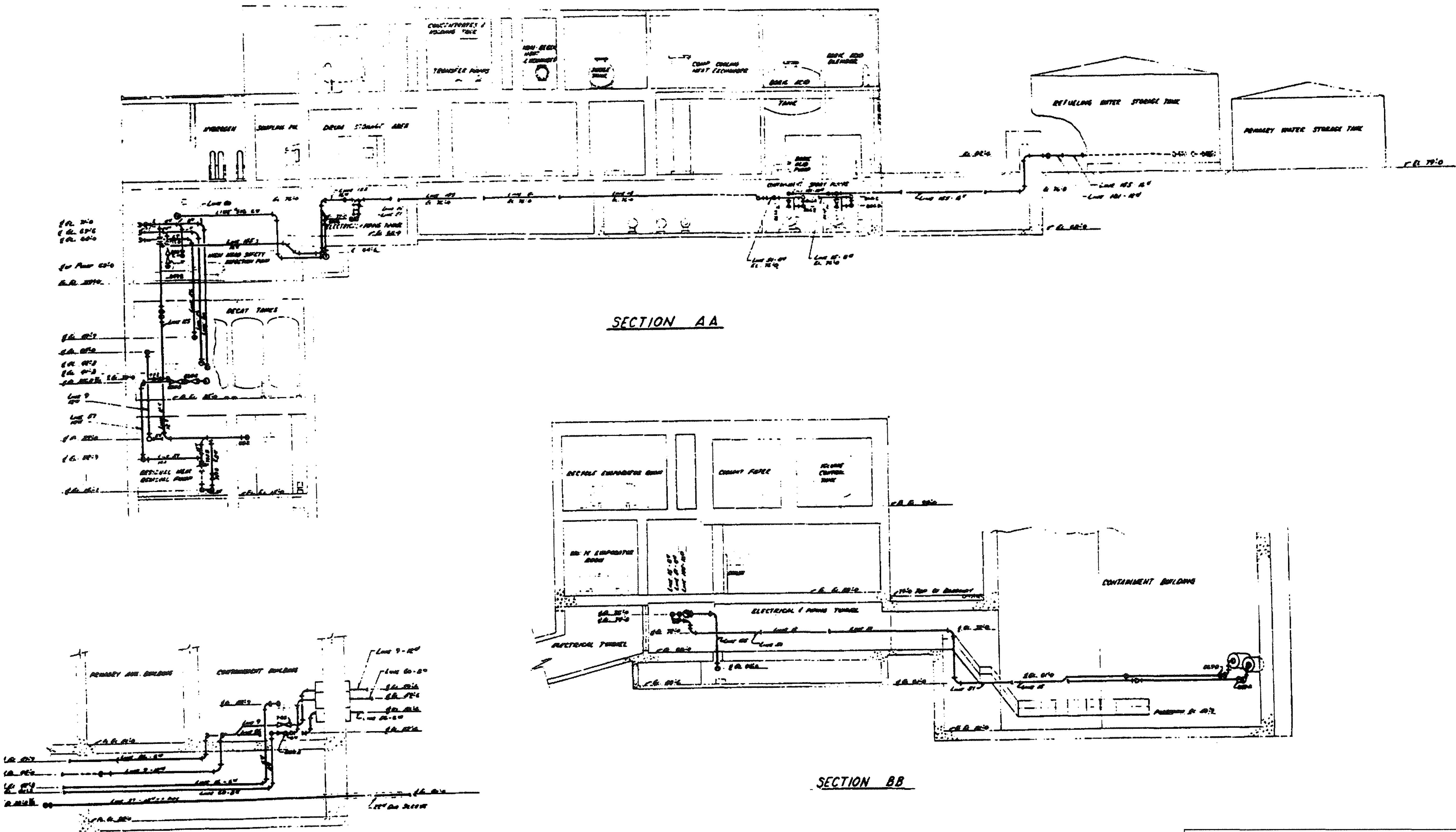
BORIC ACID
TSP.12H20 TITRATION CURVES

UFSAR FIGURE 6C-6	REV. No. 19



For Cross Sections AA, BB, and CC, See UFSAR Figure 6.2-3

INDIAN POINT UNIT No. 2	
PRIMARY AUXILIARY BUILDING SAFETY INJECTION SYSTEM PIPING-SCHEMATIC PLAN	
UFSAR FIGURE 6.2-2	REV. No. 20



SECTION AA

SECTION BB

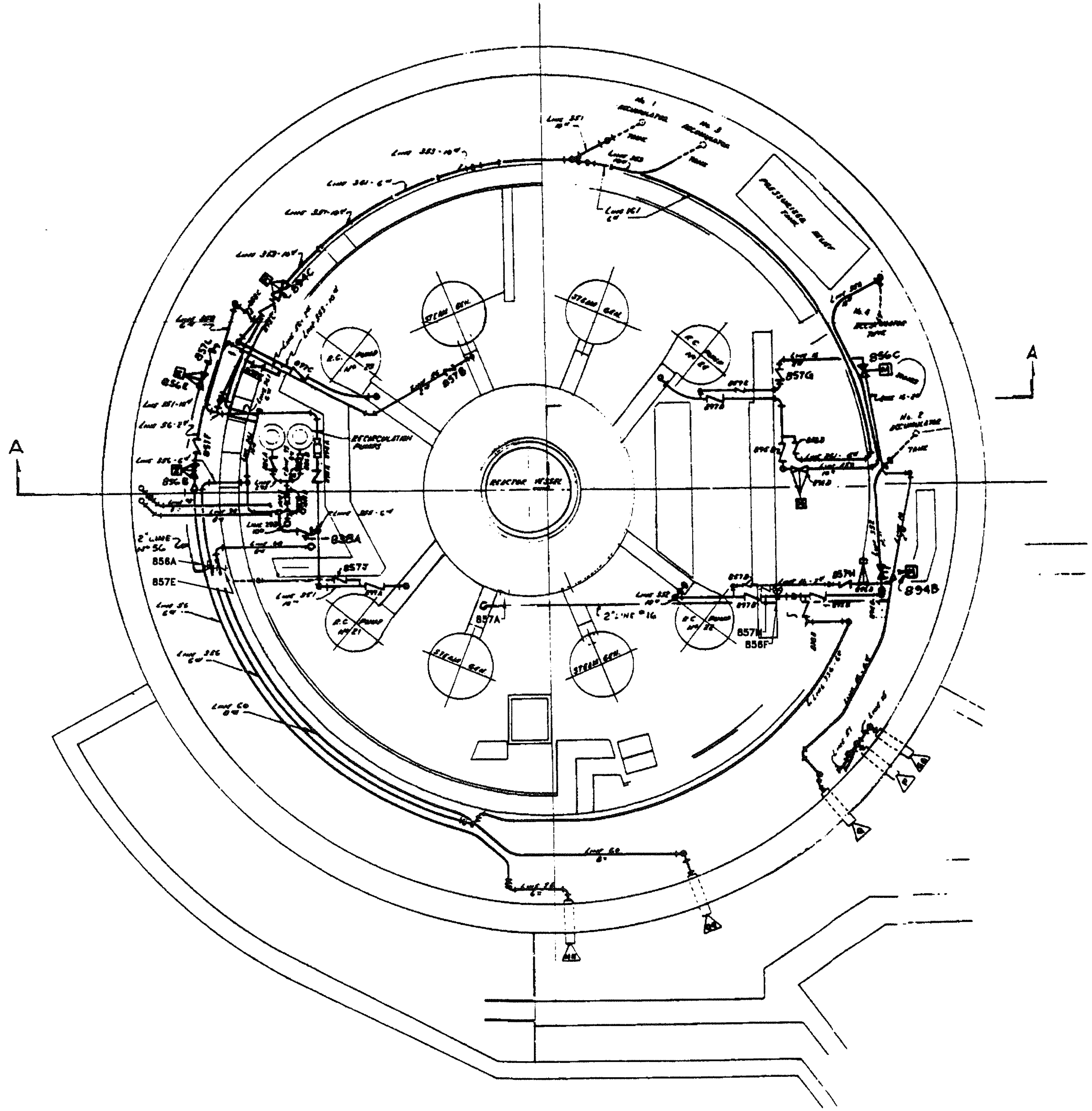
SECTION CC

INDIAN POINT UNIT No. 2

UFSAR FIGURE 6.2–3

PRIMARY AUXILIARY BUILDING SAFETY INJECTION SYSTEM PIPING SCHEMATIC ELEVATIONS

MIC. No. 1999MC3826	REV. No. 17A



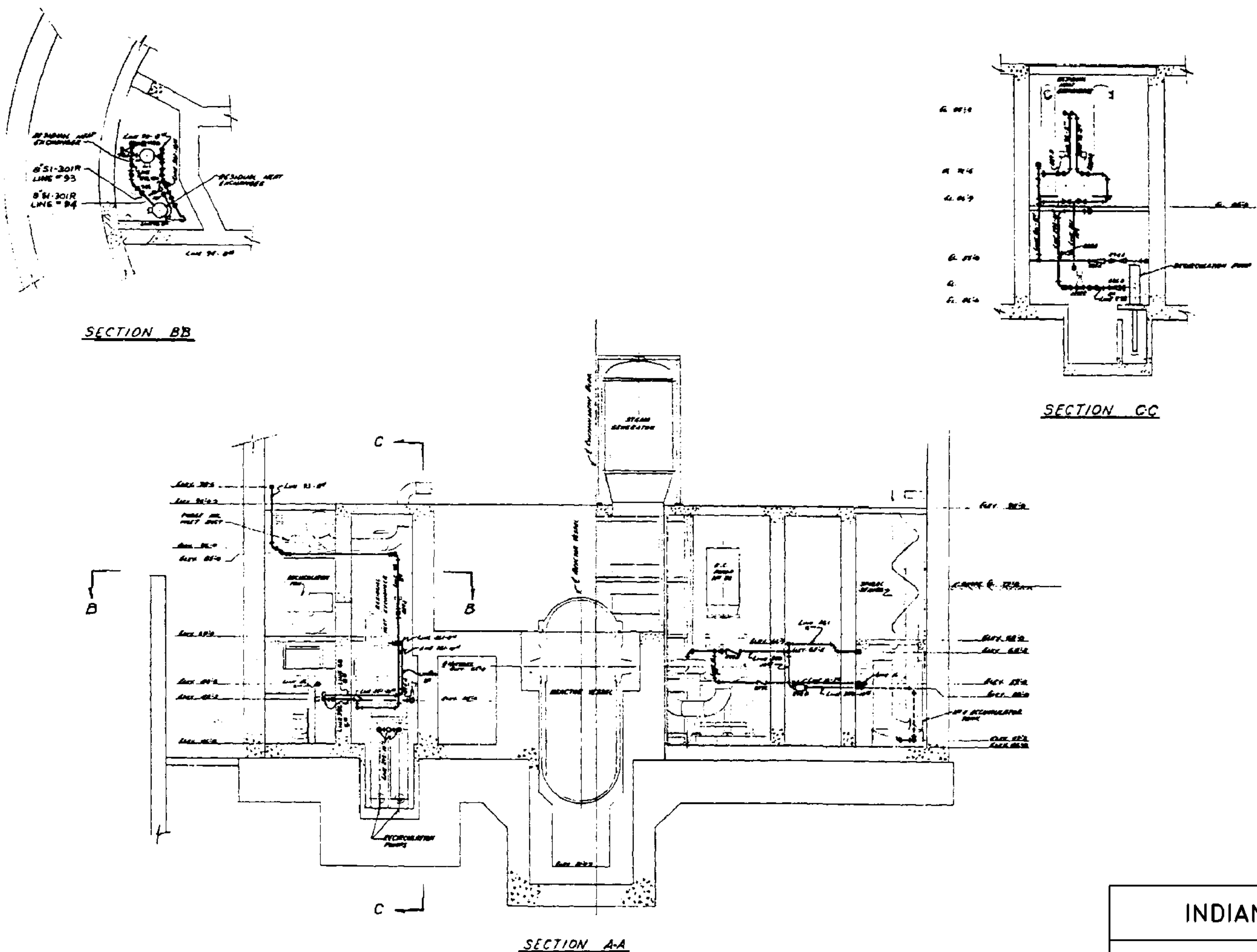
For Cross Section AA, See UFSAR Figure 6.2-5

INDIAN POINT UNIT No. 2

UFSAR FIGURE 6.2-4

CONTAINMENT BUILDING SAFETY INJECTION SYSTEM PIPING–PLAN

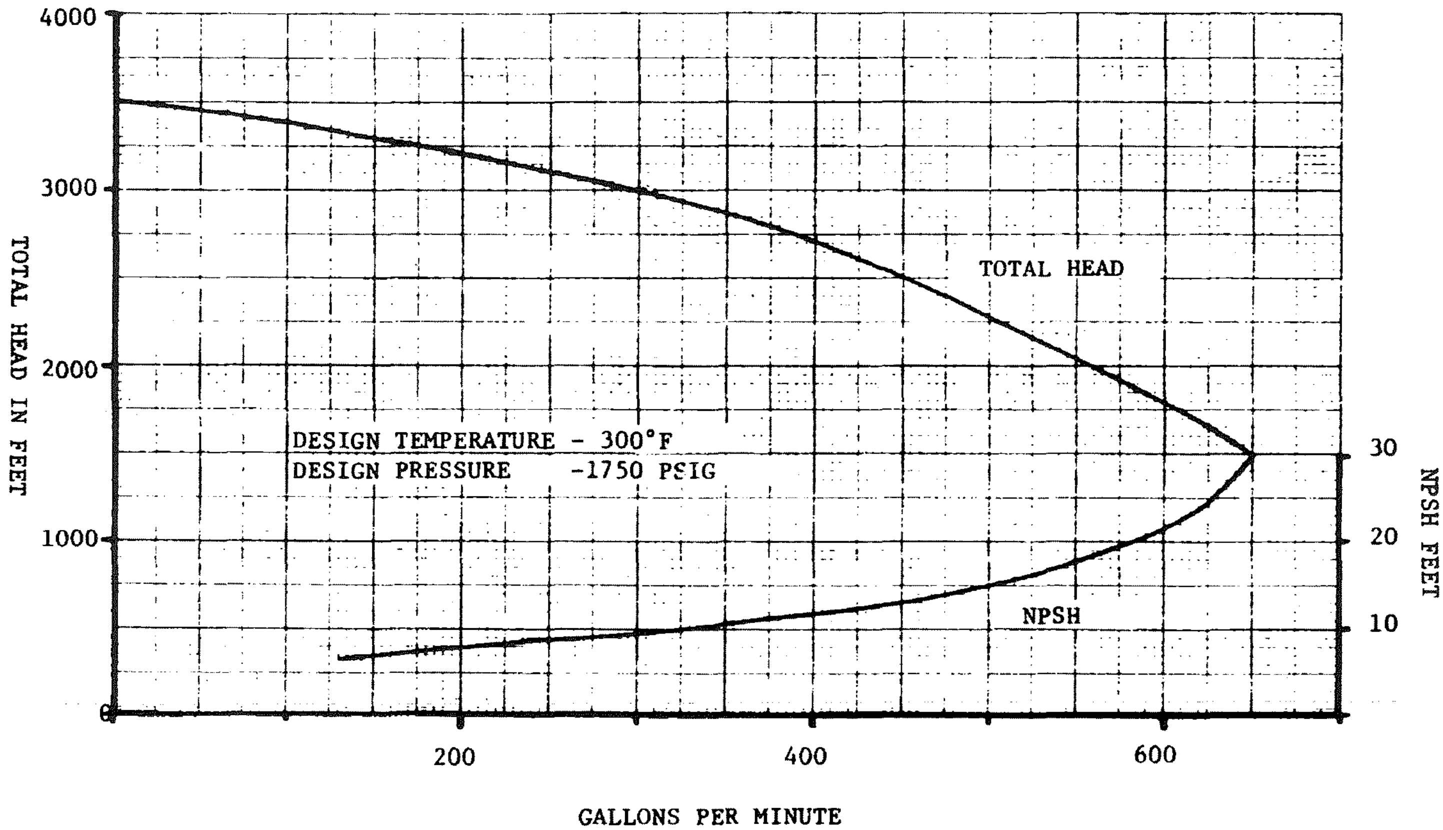
MIC. No. 1999MC3827	REV. No. 17A



INDIAN POINT UNIT No. 2

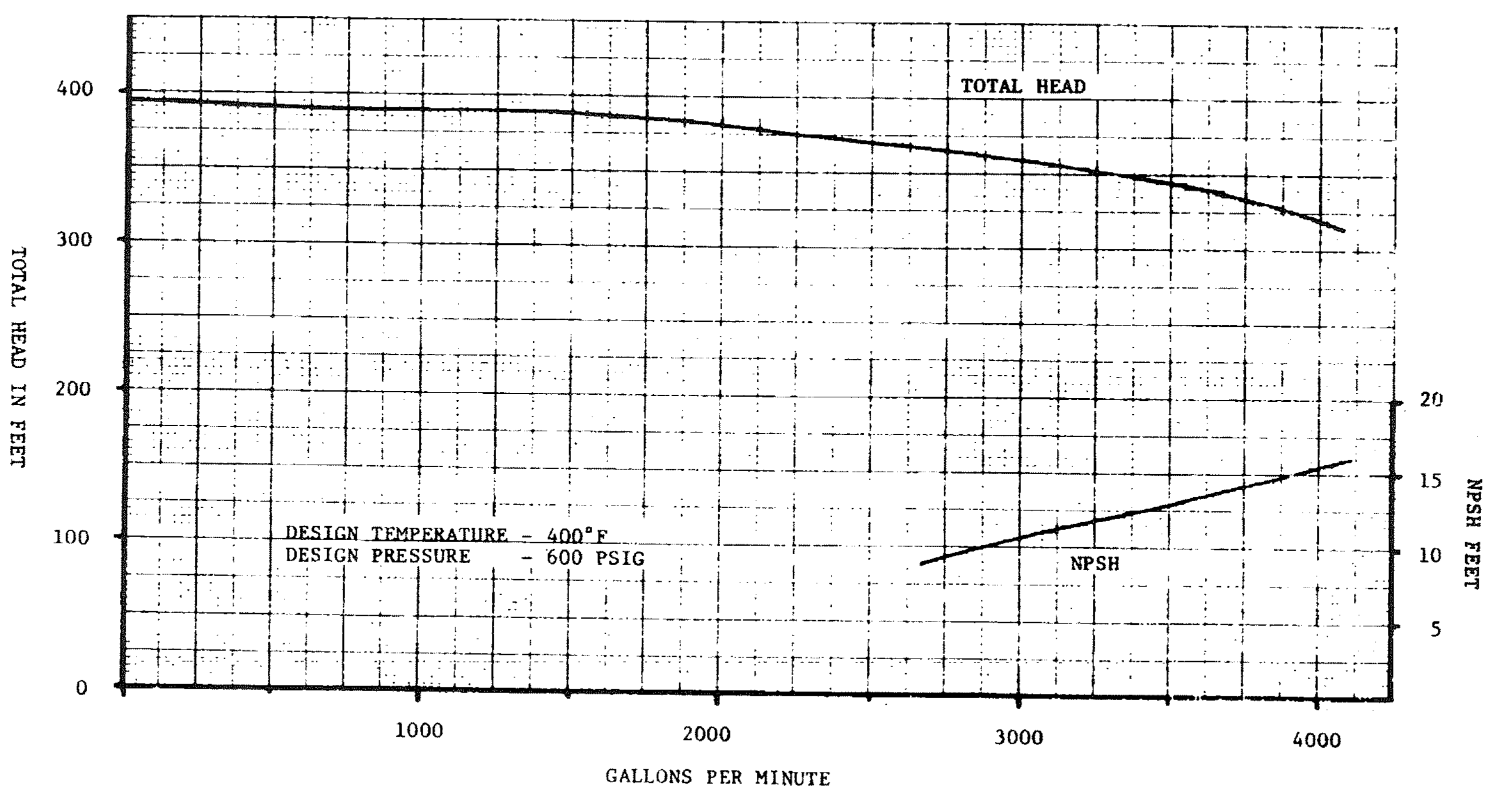
CONTAINMENT BUILDING SAFETY INJECTION PIPING ELEVATION

UFSAR FIGURE 6.2-5 | REV. No. 20



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 6.2-6	
SAFETY INJECTION PUMP PERFORMANCE	
MIC. No. 1999MC3829	REV. No. 17A

RESIDUAL HEAT REMOVAL PUMP PERFORMANCE

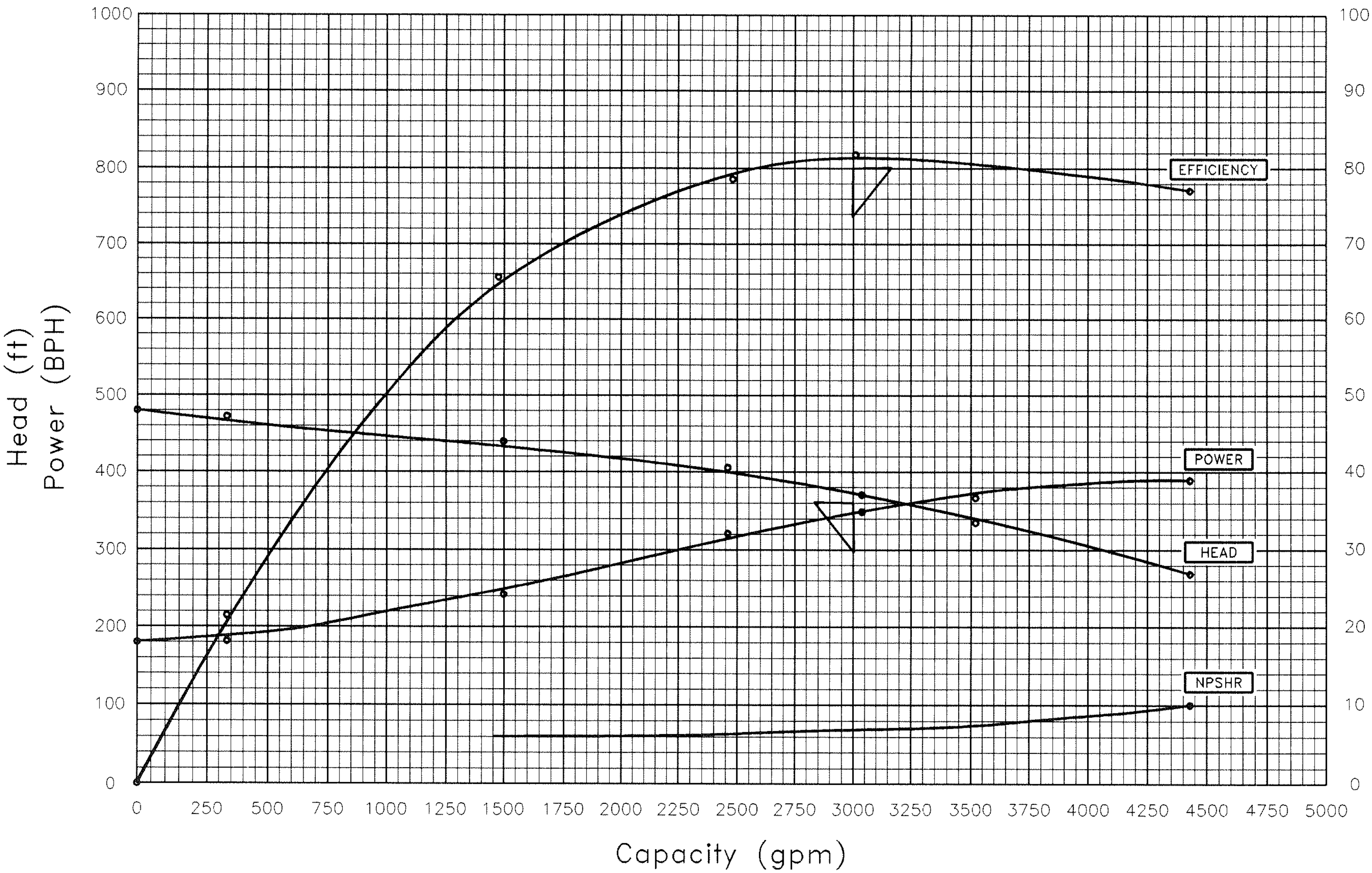


INDIAN POINT UNIT No. 2

UFSAR FIGURE 6.2-7

RESIDUAL HEAT REMOVAL PUMP PERFORMANCE

MIC. No. 1999MC3830	REV. No. 17A

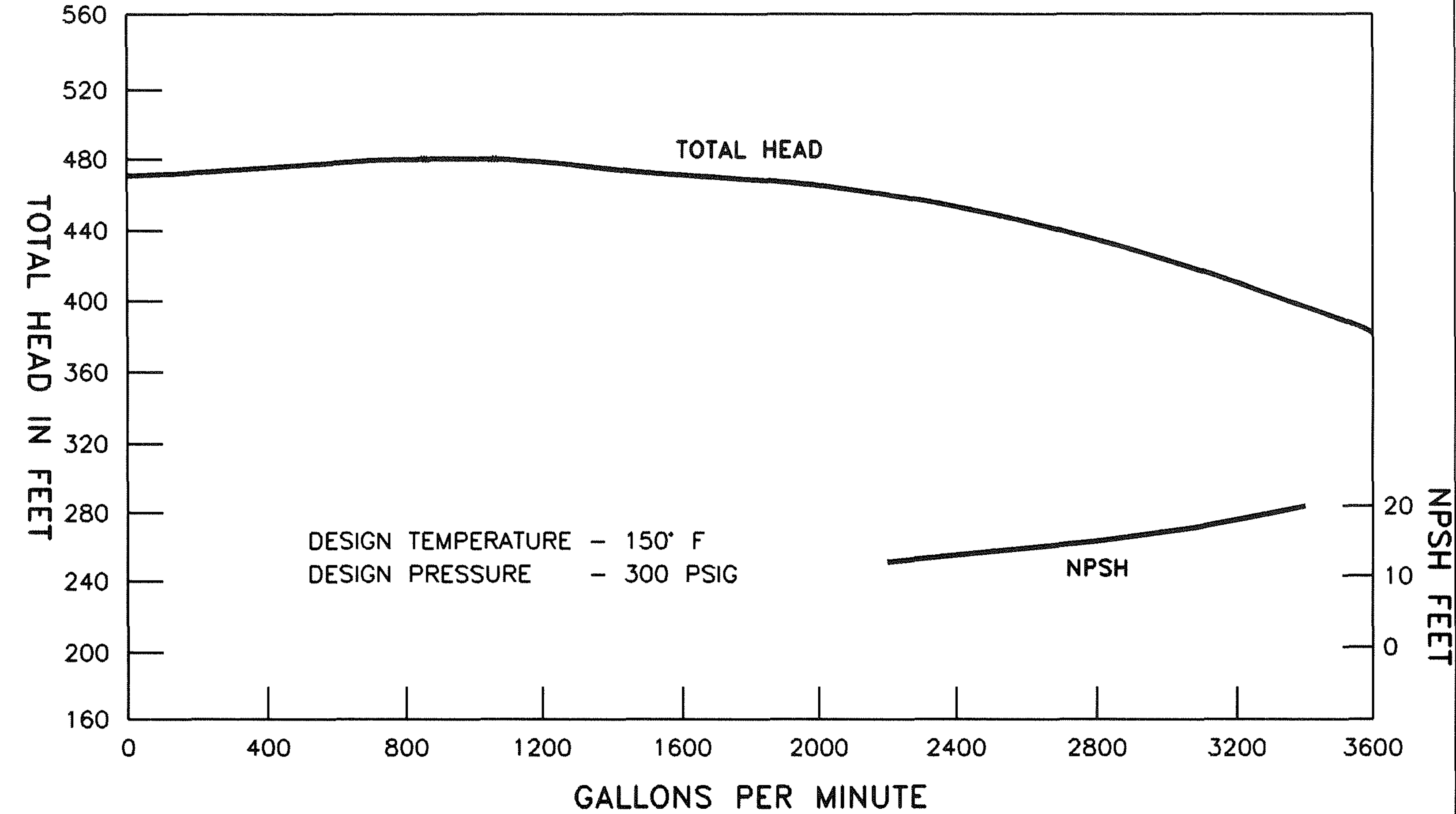


INDIAN POINT UNIT No. 2

UFSAR FIGURE 6.2-8

RECIRCULATION PUMP PERFORMANCE

MIC. No. 1999MC3831	REV. No. 17A



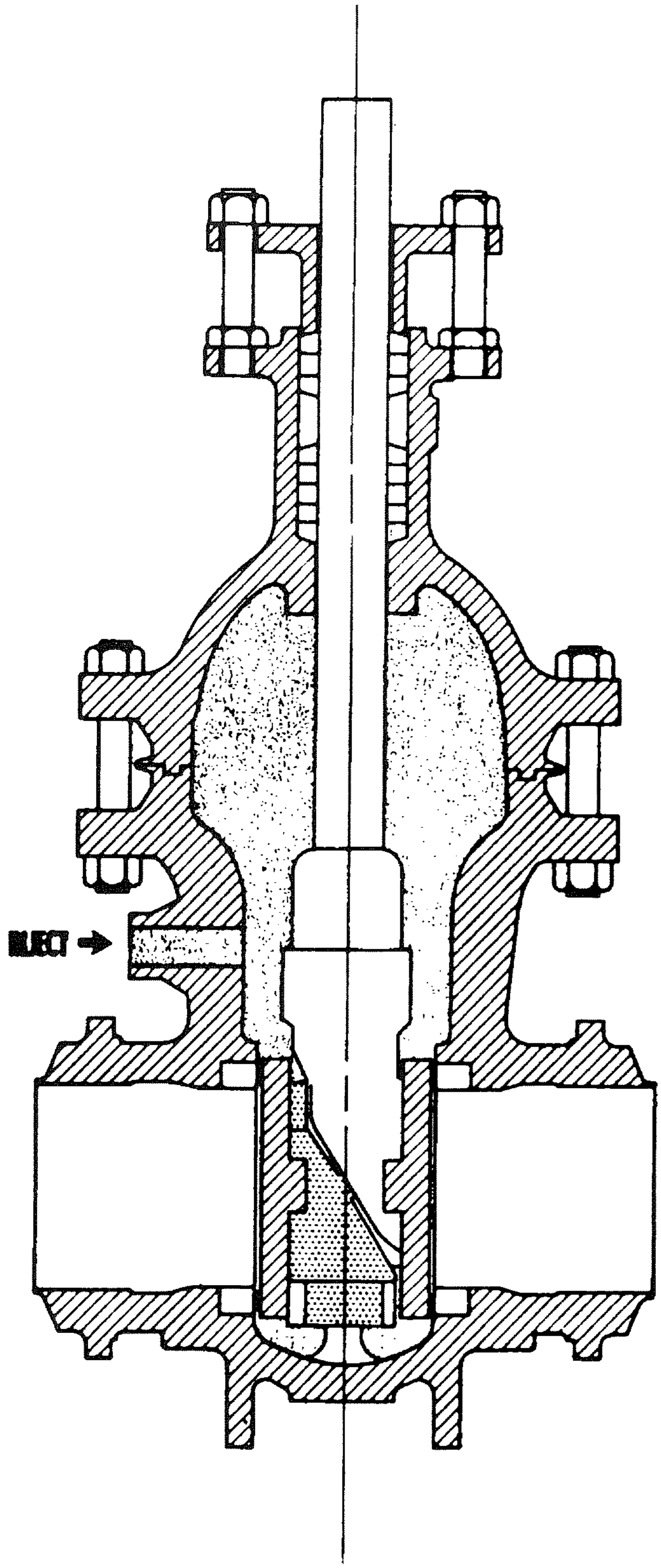
INDIAN POINT UNIT No. 2

UFSAR FIGURE 6.3–1

CONTAINMENT SPRAY PUMP
PERFORMANCE CHARACTERISTICS

MIC. No. 1999MC3833

REV. No. 17A

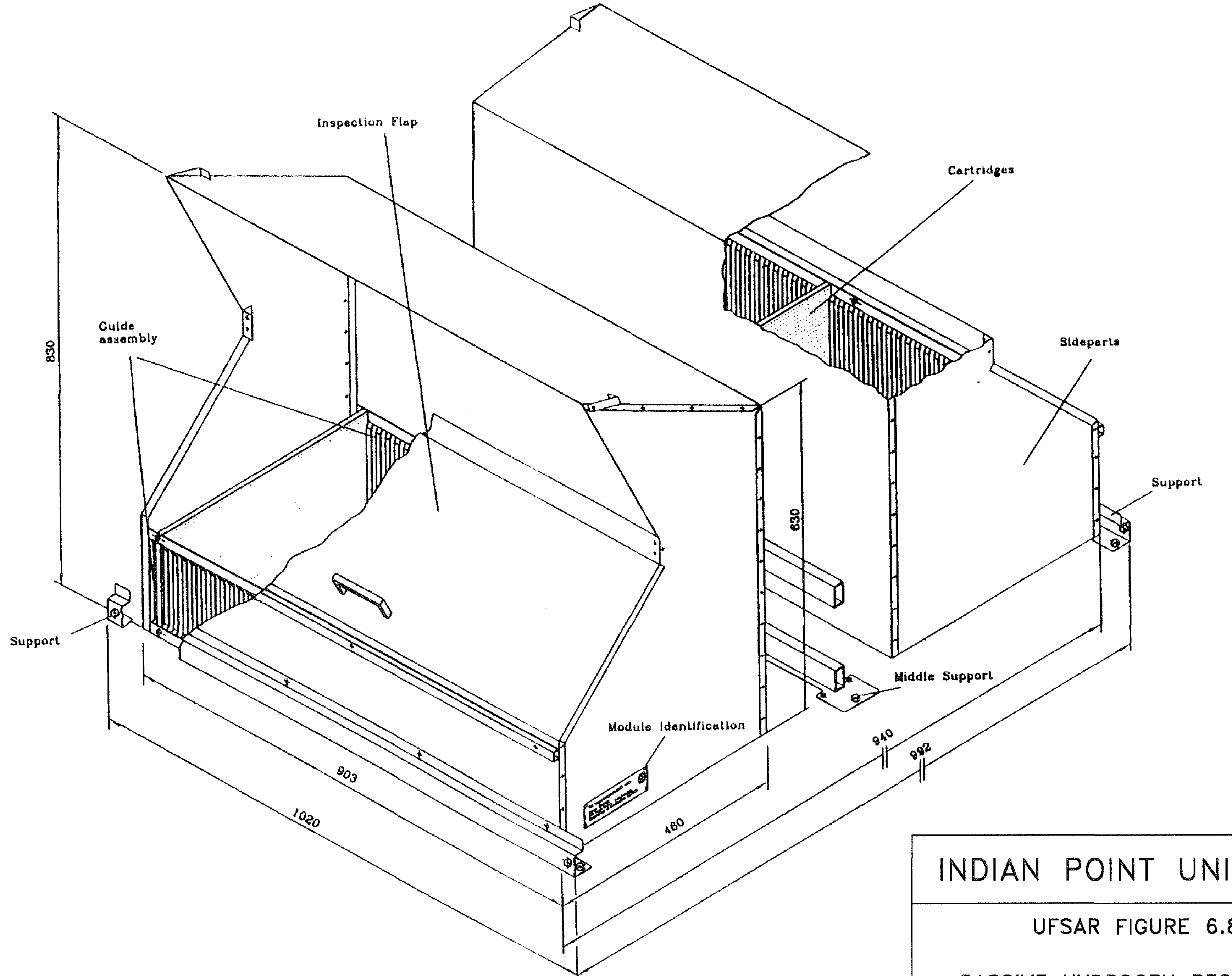


INDIAN POINT UNIT No. 2

UFSAR FIGURE 6.5-2

DOUBLE DISK ISOLATION VALVE WITH SEAL WATER INJECTION

MIC. No. 1999MC3836	REV. No. 17A

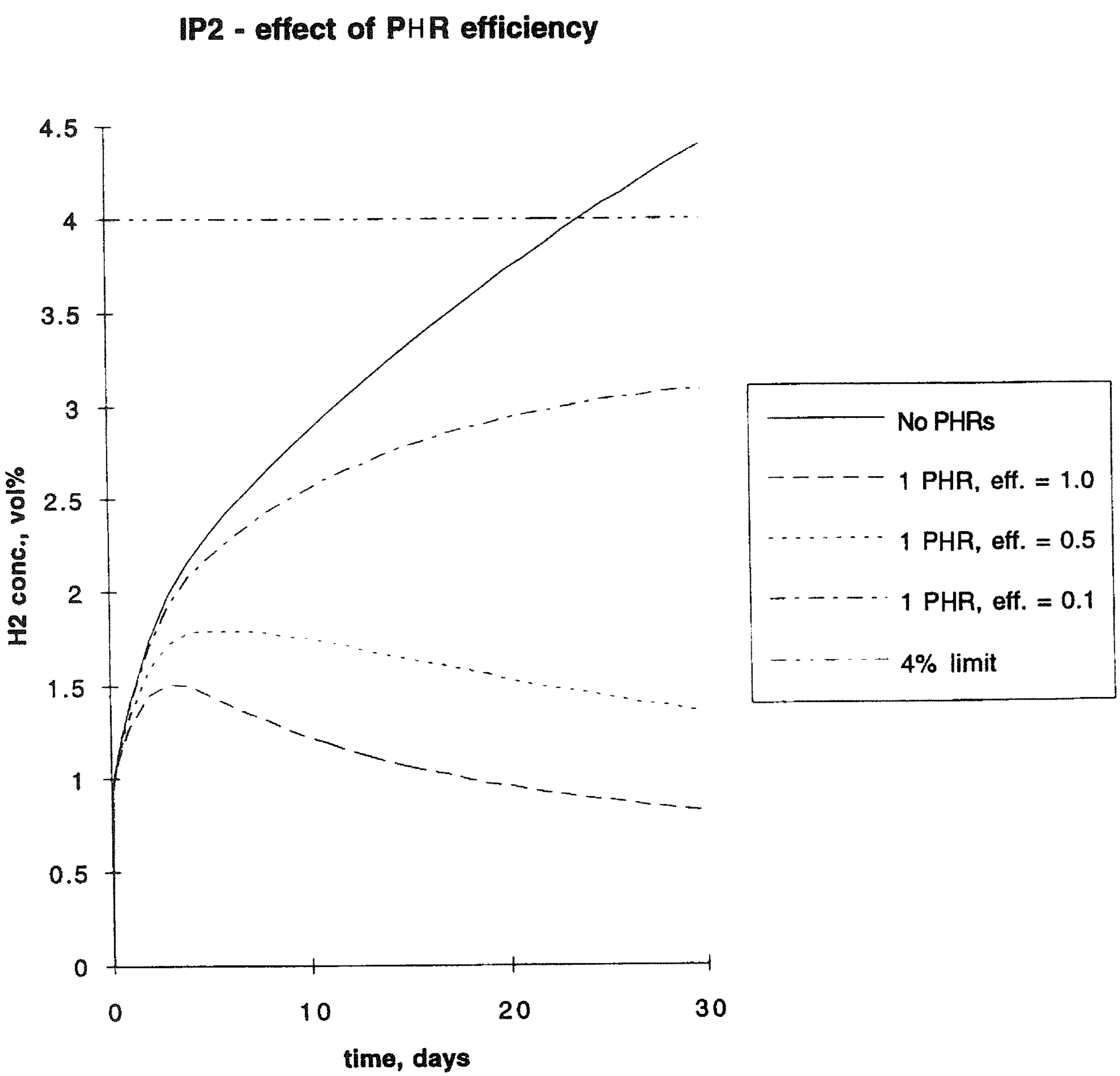


INDIAN POINT UNIT No. 2

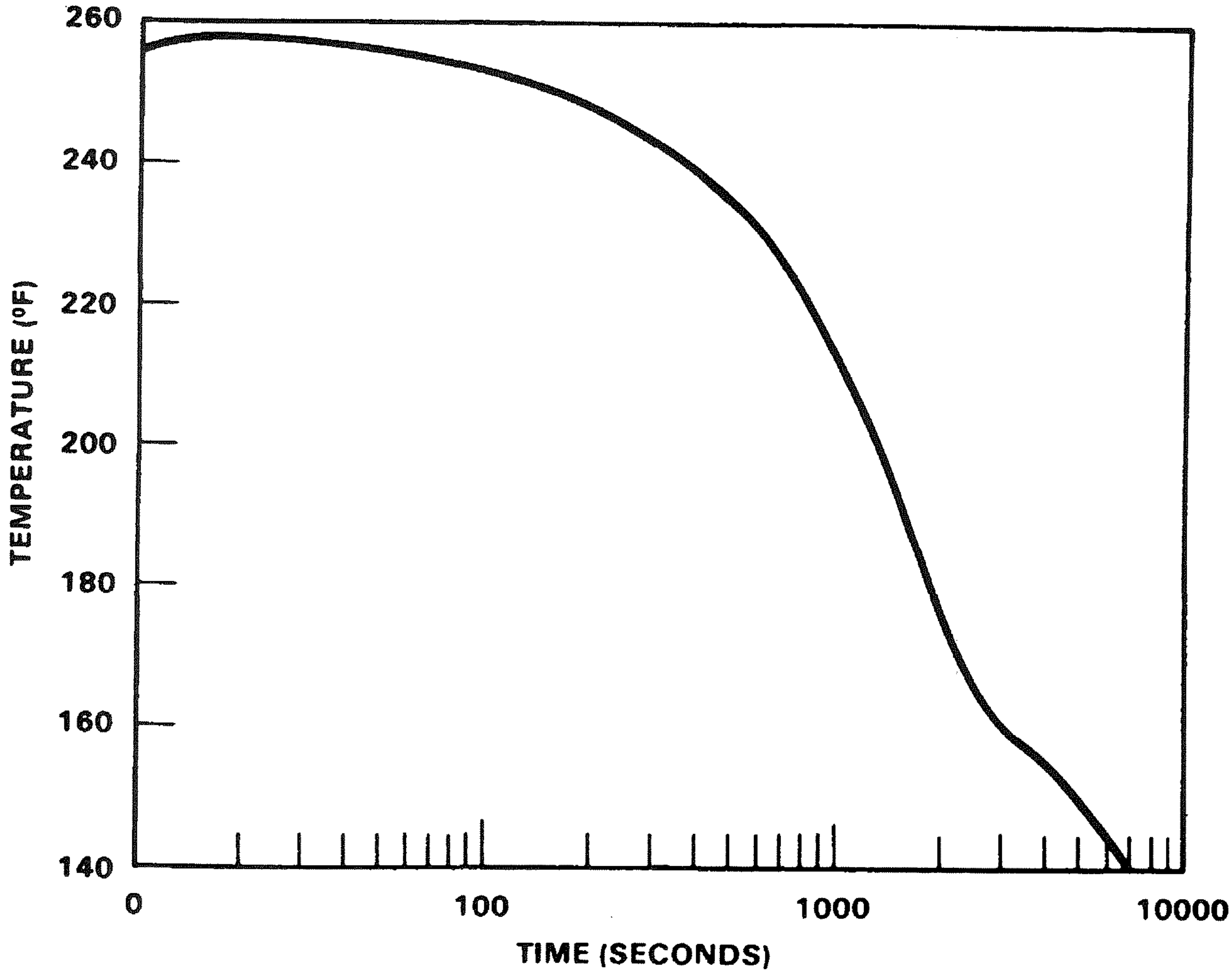
UFSAR FIGURE 6.8-1

PASSIVE HYDROGEN RECOMBINERS

MIC. No. 1999MC3528	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 6.8-2 CONTAINMENT HYDROGEN Vs TIME POST-LOCA	
MIC. No. 1999MC3619	REV. No. 17A

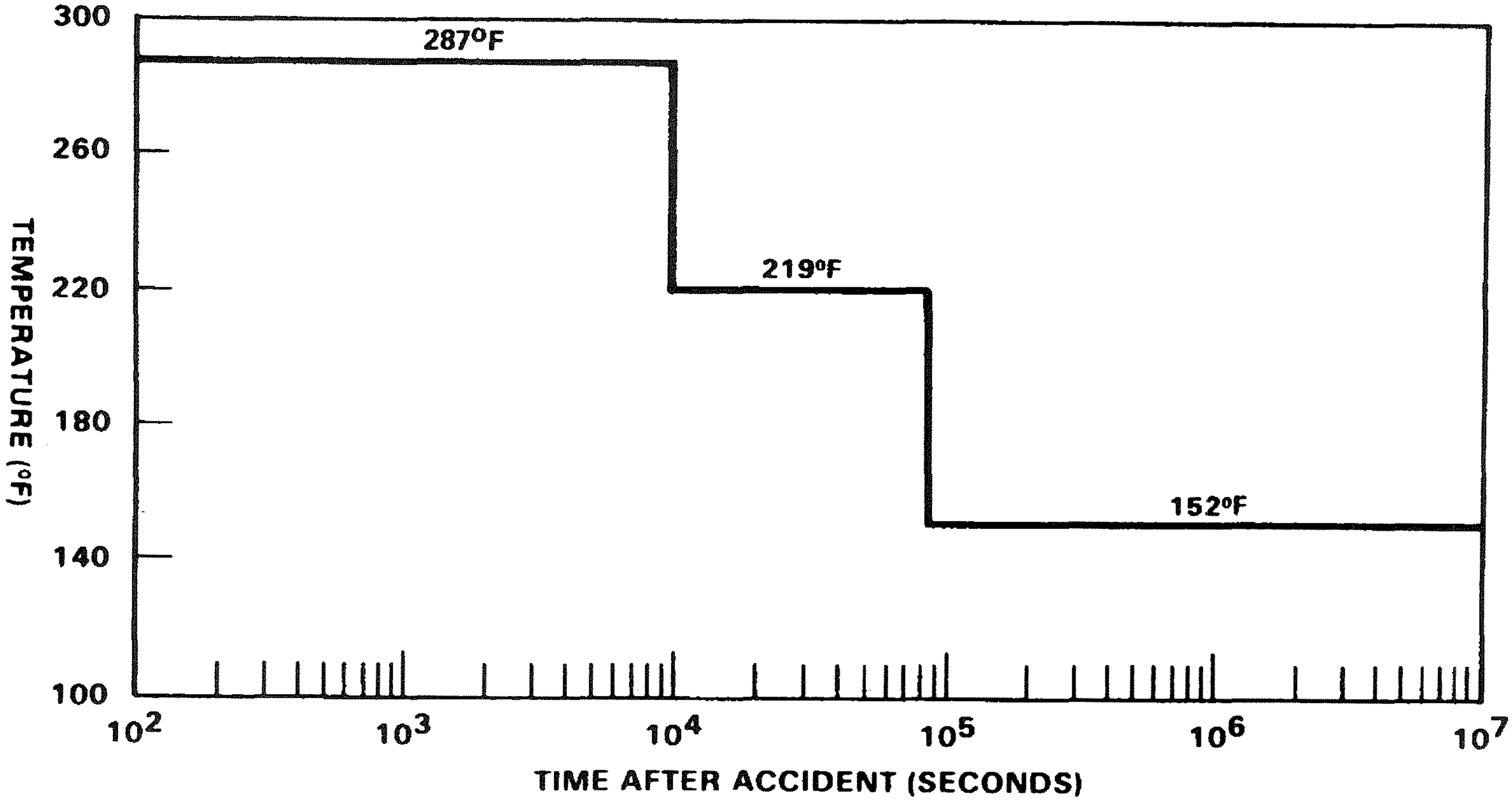


INDIAN POINT UNIT No. 2

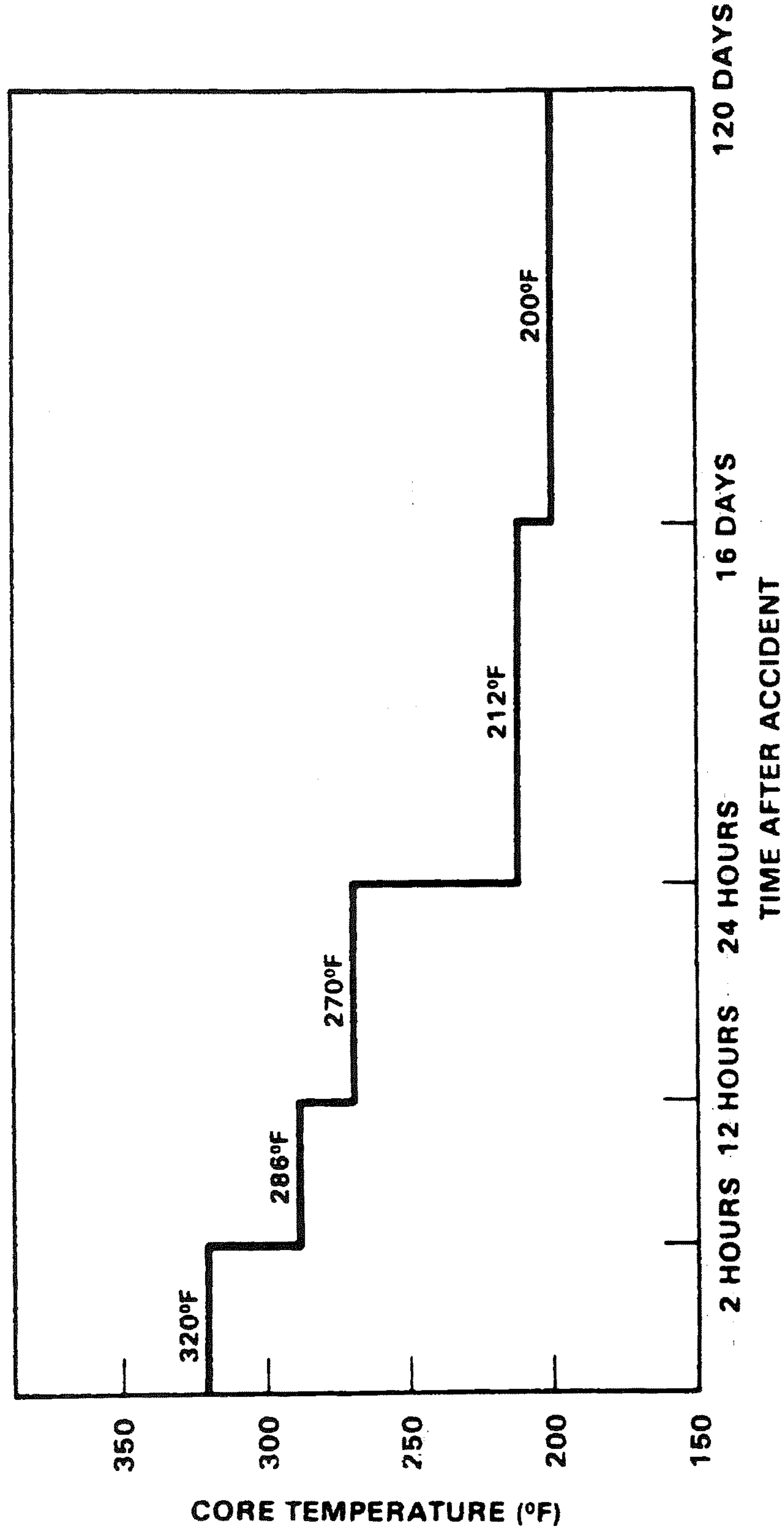
UFSAR FIGURE 6C-1

CONTAINMENT ATMOSPHERE TEMPERATURE
DESIGN BASES SAFETY INJECTION

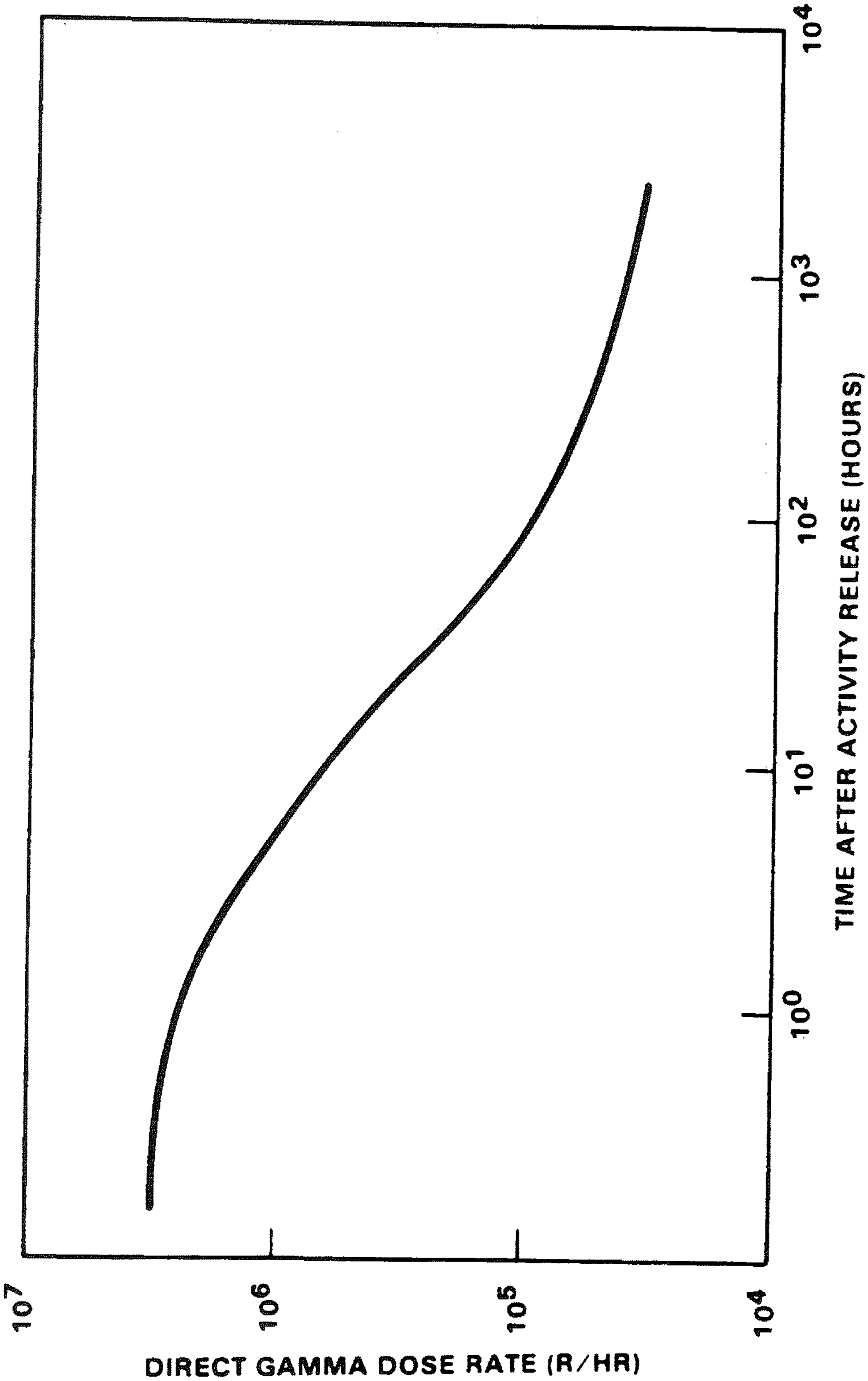
MIC. No. 1999MC3855	REV. No. 17A



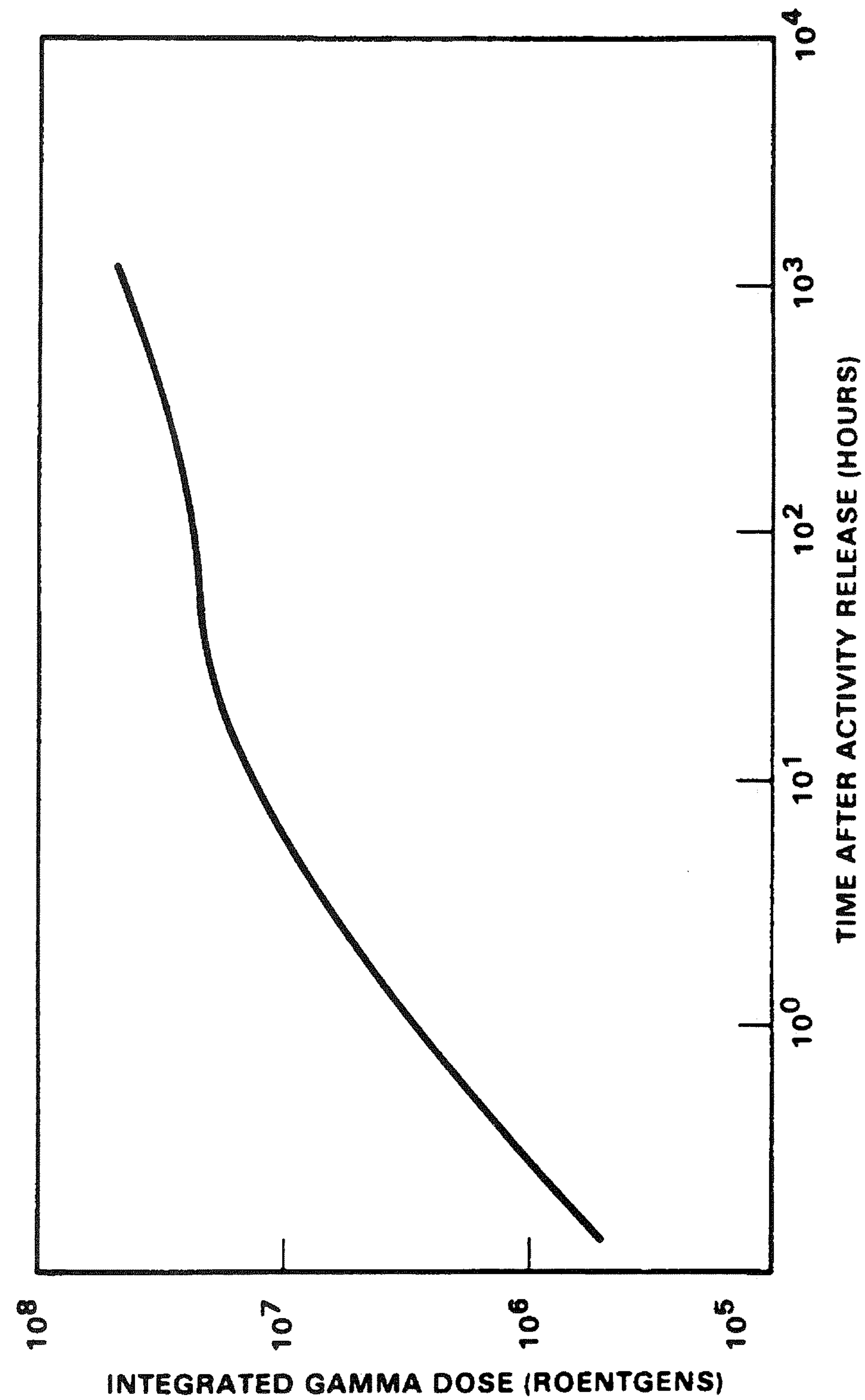
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 6C-2 INDIAN POINT UNIT 2 POSTACCIDENT CONTAINMENT MATERIALS DESIGN	
MIC. No. 1999MC3856	REV. No. 17A



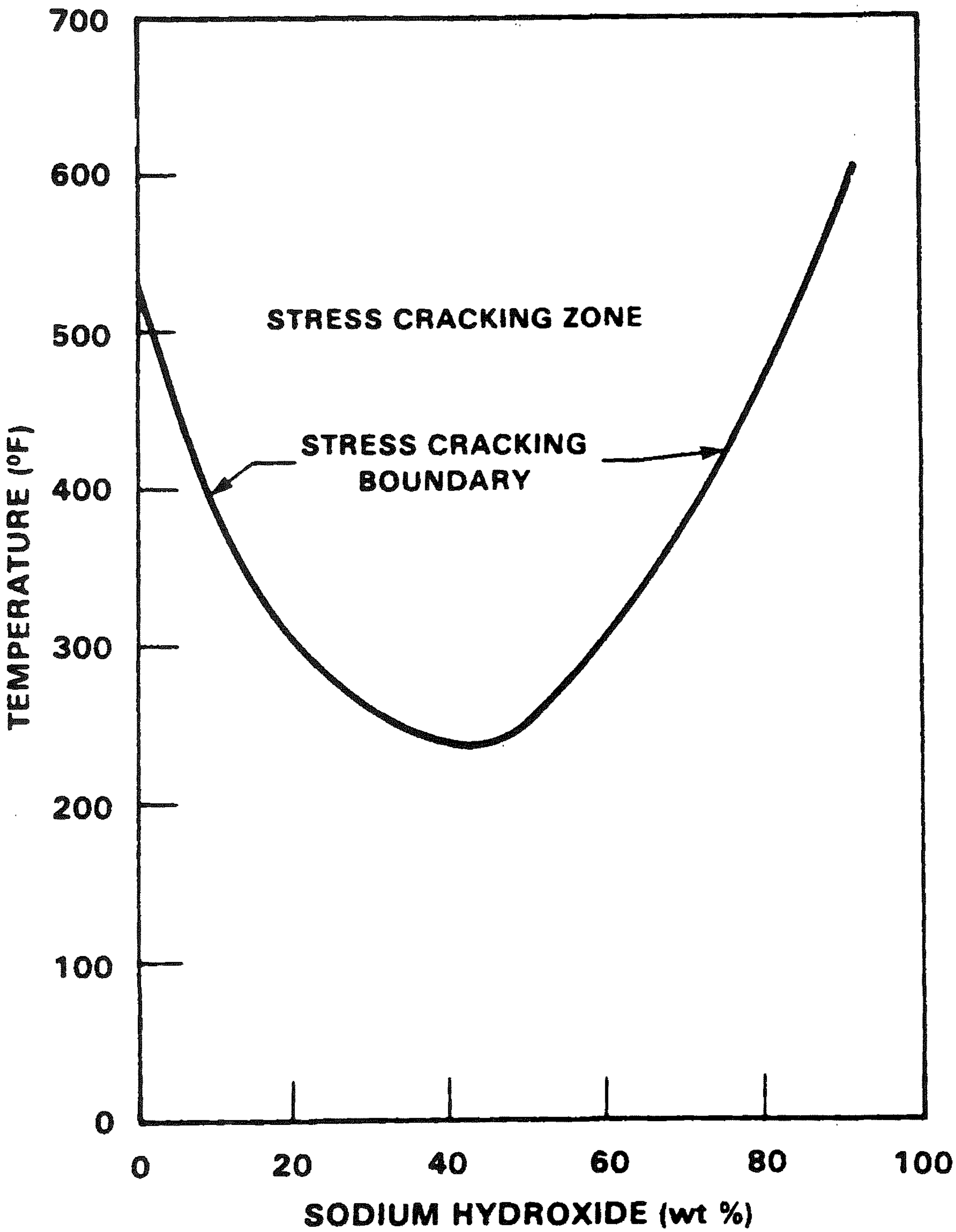
INDIAN POINT UNIT No. 2	
UFSAR FIGURE 6C-3 POSTACCIDENT CORE MATERIALS DESIGN CONDITIONS	
MIC. No. 1999MC3857	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 6C-4 INDIAN POINT UNIT 2 CONTAINMENT ATMOSPHERE DIRECT GAMMA DOSE RATE	
MIC. No. 1999MC3858	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 6C–5 INDIAN POINT UNIT 2 CONTAINMENT ATMOSPHERE INTEGRATED GAMMA DOSE LEVEL	
MIC. No. 1999MC3859	REV. No. 17A

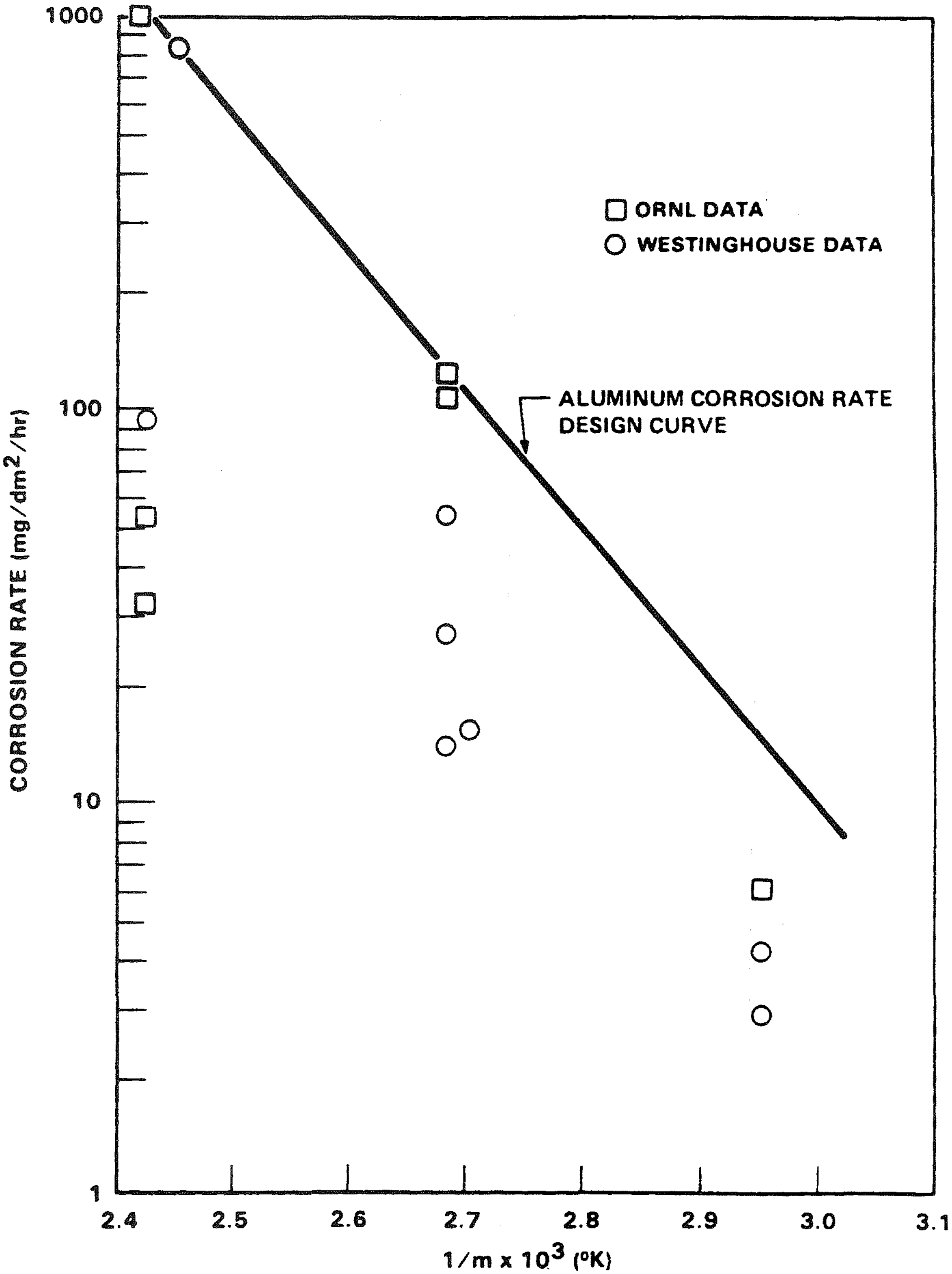


INDIAN POINT UNIT No. 2

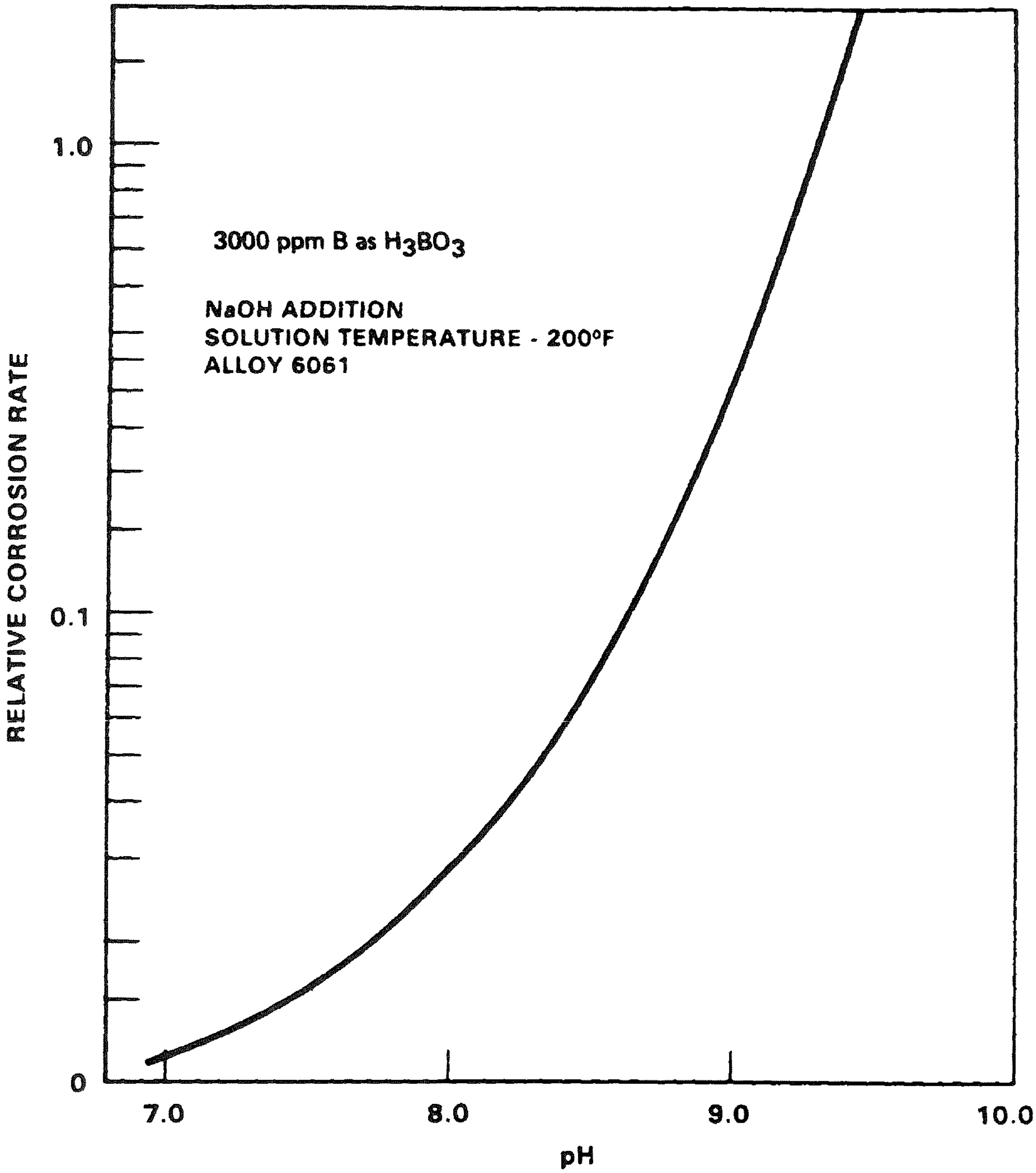
UFSAR FIGURE 6C–7

TEMPERATURE–CONCENTRATION RELATION FOR CAUSTIC CORROSION OF AUSTENITIC STAINLESS STEEL

MIC. No. 1999MC3861	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 6C-8	
ALUMINUM CORROSION IN DESIGN-BASIS-ACCIDENT ENVIRONMENT	
MIC. No. 1999MC3862	REV. No. 17A

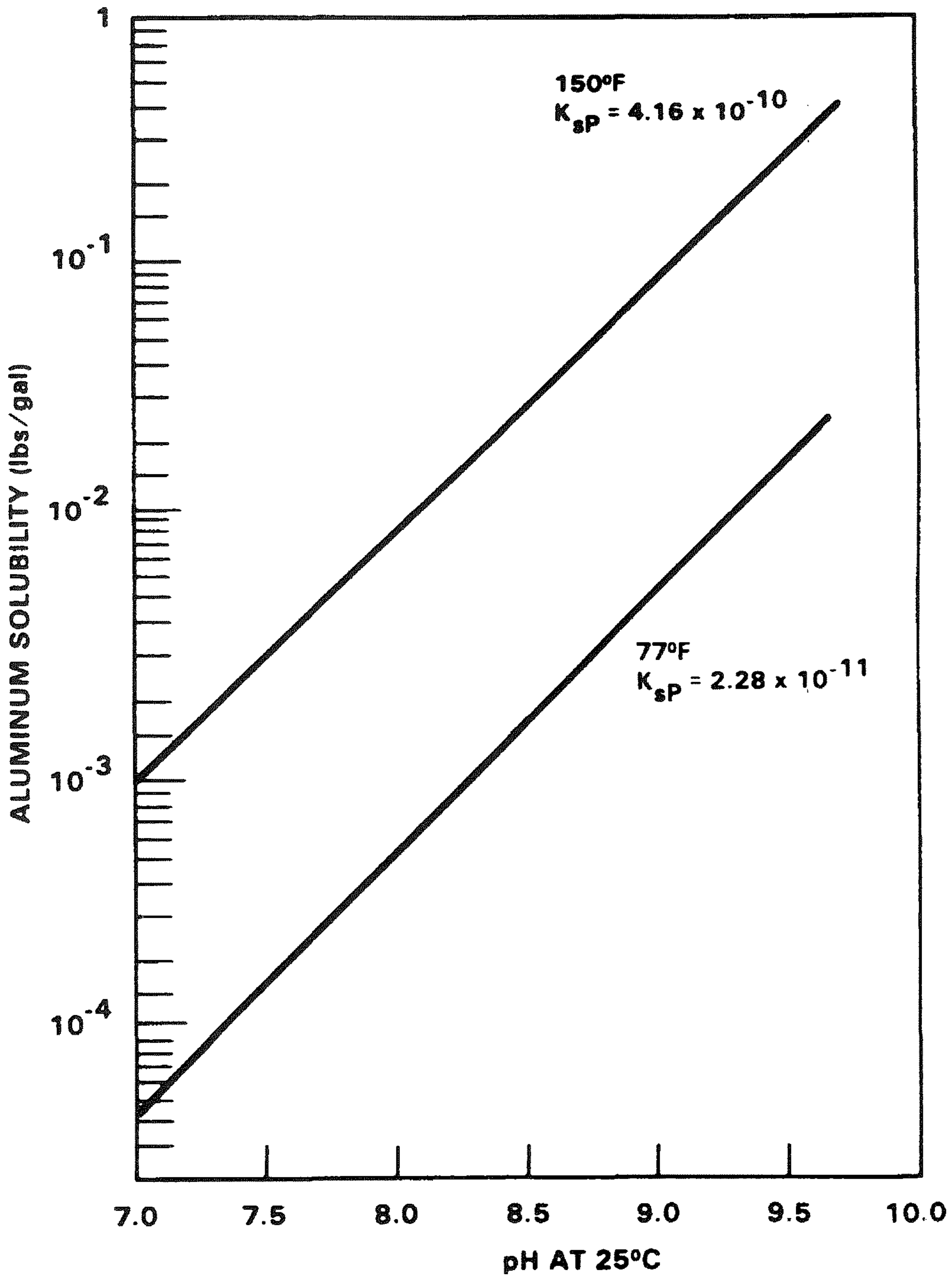


INDIAN POINT UNIT No. 2

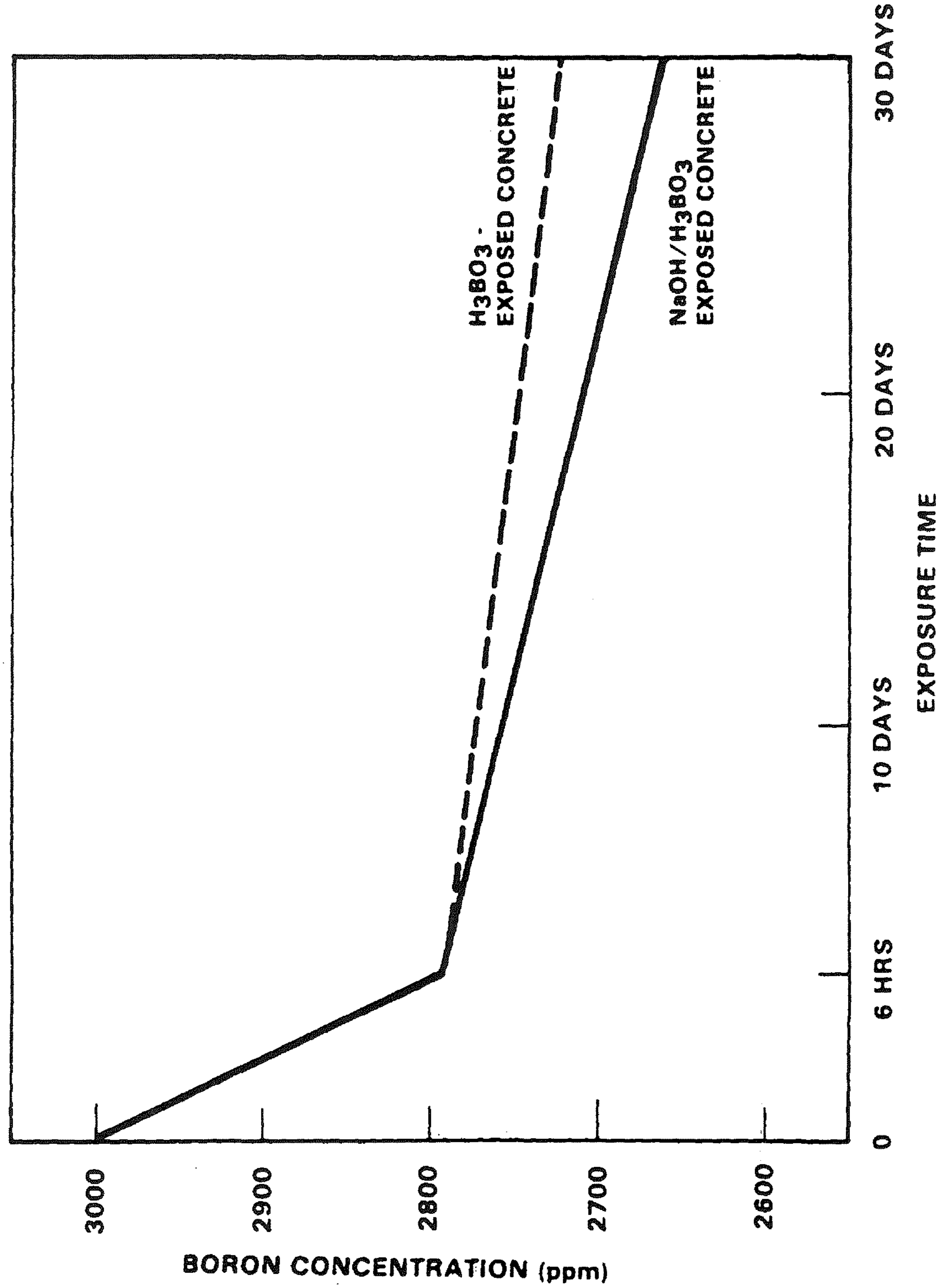
UFSAR FIGURE 6C-9

ALUMINUM CORROSION AS A FUNCTION OF PH

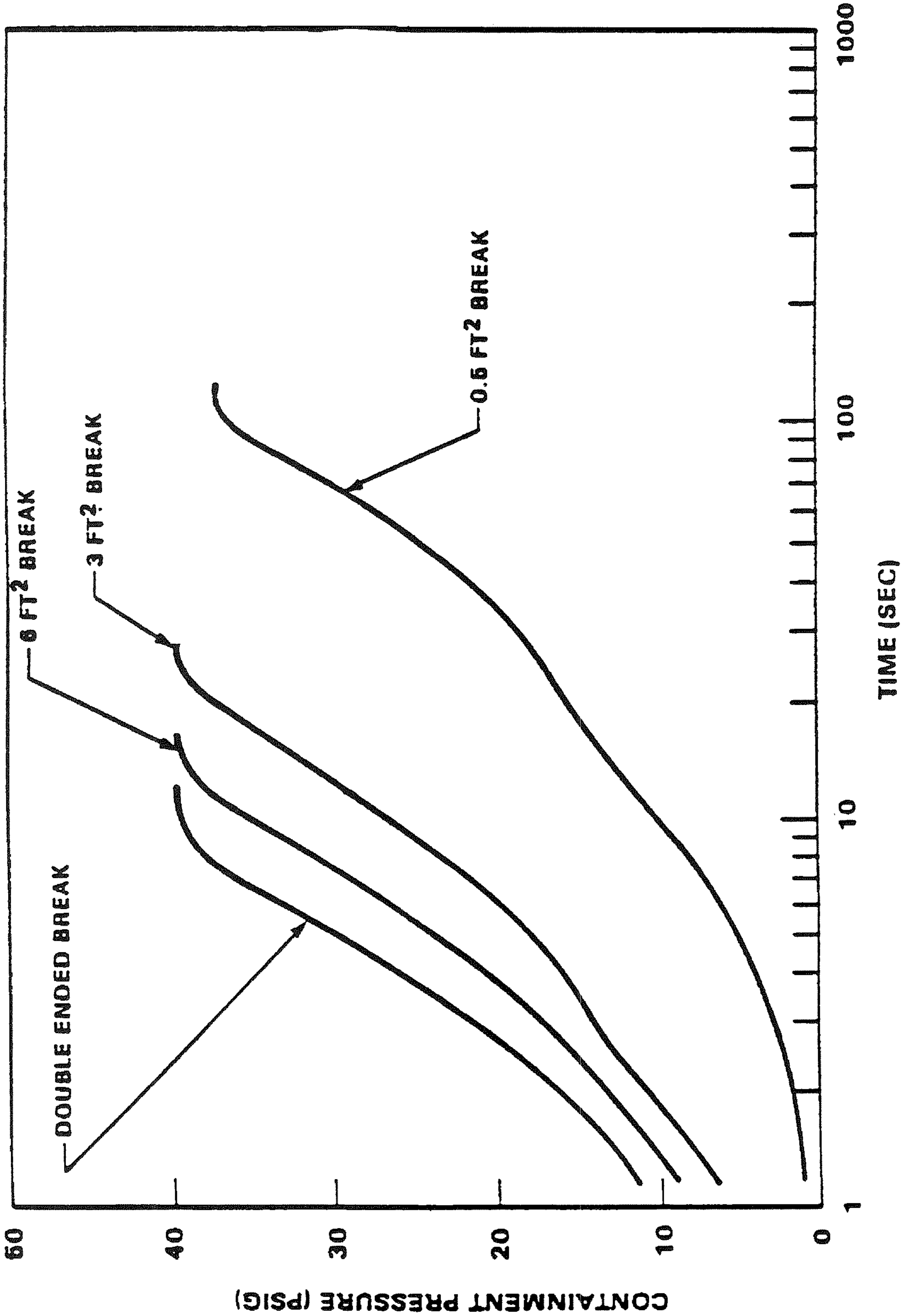
MIC. No. 1999MC3863	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 6C–10 SOLUBILITY OF ALUMINUM CORROSION PRODUCTS AS A FUNCTION OF PH AT 77°F AND 150°F	
MIC. No. 1999MC3864	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 6C–11 BORON LOSS FROM BORON–CONCRETE REACTION FOLLOWING A DESIGN–BASIS ACCIDENT	
MIC. No. 1999MC3865	REV. No. 17A

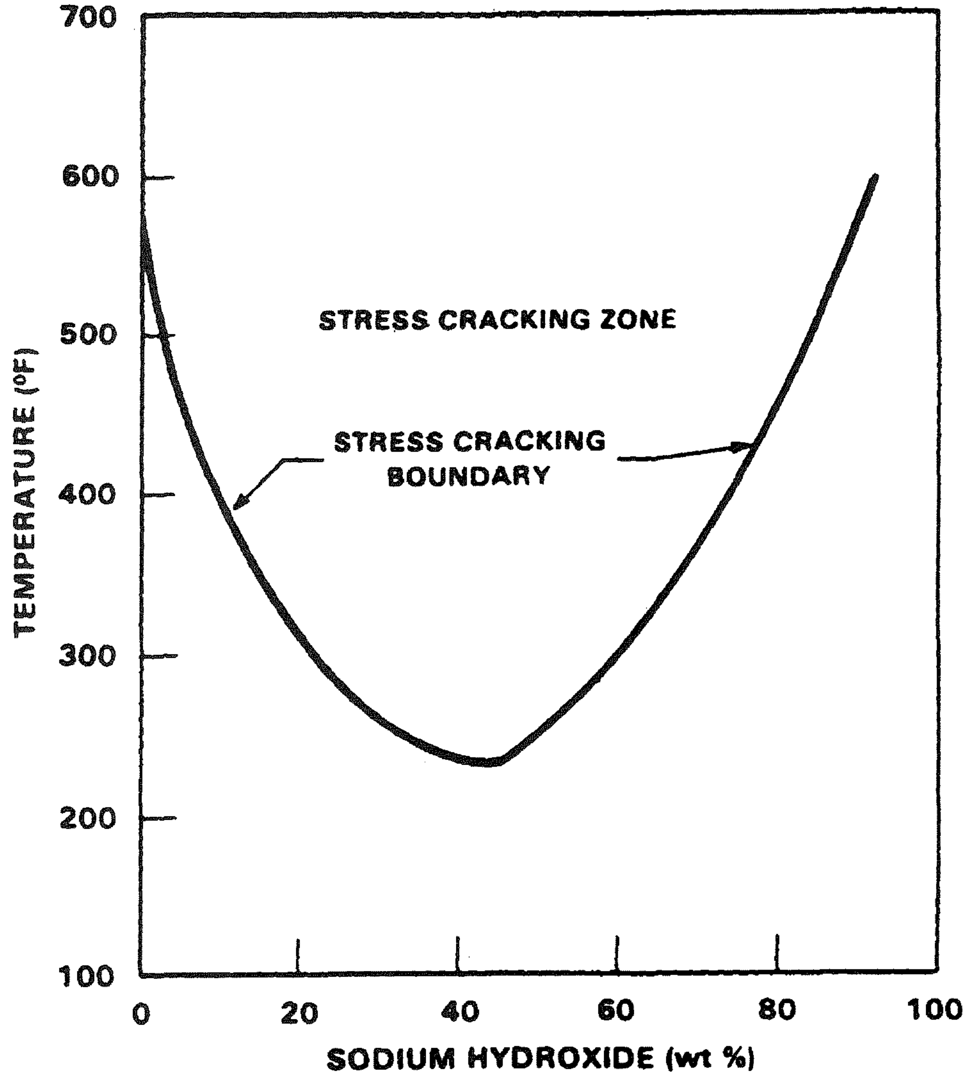


INDIAN POINT UNIT No. 2

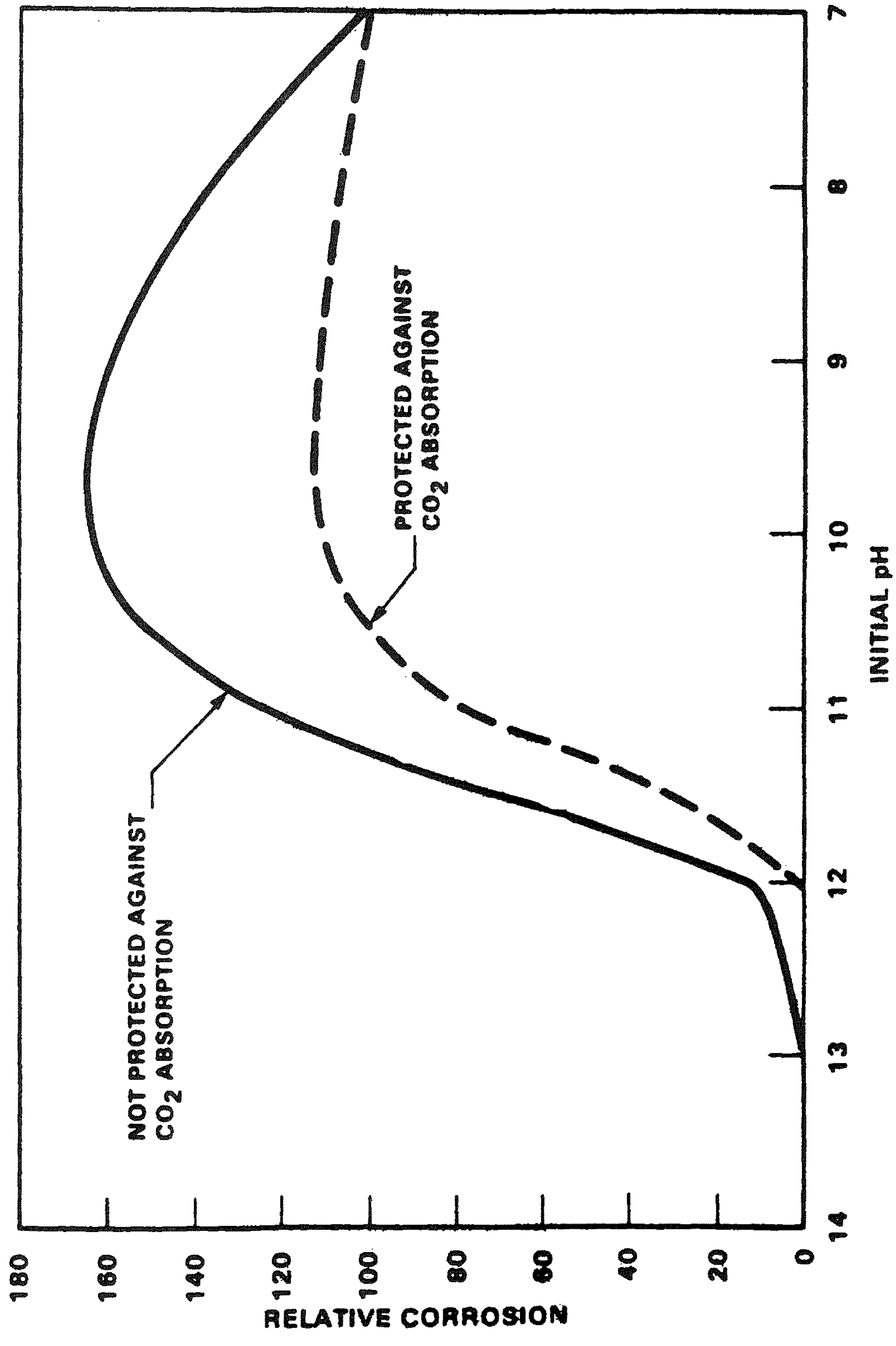
UFSAR FIGURE 6C-12

CONTAINMENT PRESSURE TRANSIENT – DURING BLOWDOWN PHASE vs TIME

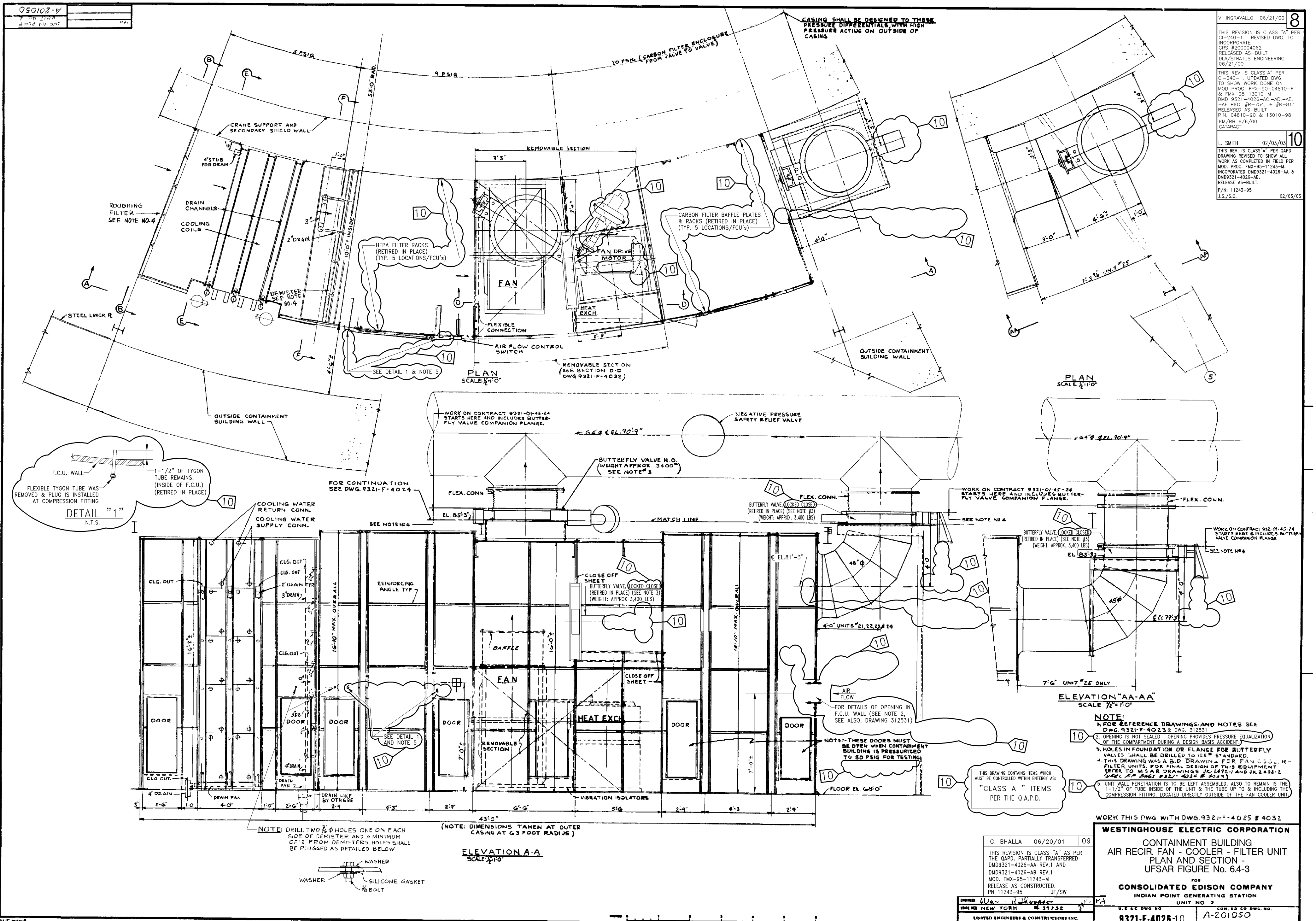
MIC. No. 1999MC3866	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 6D-1 TEMPERATURE - CONCENTRATION RELATIONS FOR CAUSTIC CORROSION OF AUSTENITIC STAINLESS STEEL	
MIC. No. 1999MC3867	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 6D–2 EFFECT OF CARBON DIOXIDE ON CORROSION OF IRON IN NaOH SOLUTION	
MIC. No. 1999MC3868	REV. No. 17A



A-201050
V. INGRAVALLO 06/21/00 8
THIS REVISION IS CLASS "A" PER CI-240-1. REVISED DWG. TO INCORPORATE CRS #200004062 RELEASED AS-BUILT DLA/STRATUS ENGINEERING 06/21/00
THIS REV IS CLASS"A" PER CI-240-1. UPDATED DWG. TO SHOW WORK DONE ON MOD PROC. FPX-90-04810-F & FMX-98-13010-M DMD 9321-4026-AC,-AD,-AE, -AF PKG. #R-754, & #R-814 RELEASED AS-BUILT P.N. 04810-90 & 13010-98 KM/RB 6/6/00 CATARACT
L. SMITH 02/03/03 10
THIS REV. IS CLASS"A" PER QAPD. DRAWING REVISED TO SHOW ALL WORK AS COMPLETED IN FIELD PER MOD. PROC. FMX-95-11243-M. INCORPORATED DMD9321-4026-AA & DMD9321-4026-AB. RELEASE AS-BUILT.
P/N: 11243-95
J.S./S.O. 02/03/03
CASING SHALL BE DESIGNED TO THESE PRESSURE DIFFERENTIALS, WITH HIGH PRESSURE ACTING ON OUTSIDE OF CASING
5 PSIG
9 PSIG
20 PSIG (CARBON FILTER ENCLOSURE FROM VALVE TO VALVE)
53'-0" RAD.
CRANE SUPPORT AND SECONDARY SHIELD WALL
REMOVABLE SECTION
4" STUB FOR DRAIN
ROUGHING FILTER SEE NOTE NO. 4
DRAIN CHANNELS
COOLING COILS
2" DRAIN
10'-0" INSIDE
DEMISTER SEE NOTE NO. 4
STEEL LINER R
HEPA FILTER RACKS (RETIRED IN PLACE) (TYP. 5 LOCATIONS/FCU's)
FAN
FAN DRIVE MOTOR
HEAT EXCH.
CARBON FILTER BAFFLE PLATES & RACKS (RETIRED IN PLACE) (TYP. 5 LOCATIONS/FCU's)
FLEXIBLE CONNECTION
AIR FLOW CONTROL SWITCH
SEE DETAIL 1 & NOTE 5
REMOVABLE SECTION (SEE SECTION D-D DWG 9321-F-4032)
OUTSIDE CONTAINMENT BUILDING WALL
PLAN SCALE ½"=1'-0"
7'-3¾" UNIT #25
PLAN SCALE ½"=1'-0"
F.C.U. WALL
1-1/2" OF TYGON TUBE REMAINS. (INSIDE OF F.C.U.) (RETIRED IN PLACE)
FLEXIBLE TYGON TUBE WAS REMOVED & PLUG IS INSTALLED AT COMPRESSION FITTING
DETAIL "1" N.T.S.
WORK ON CONTRACT 9321-01-45-24 STARTS HERE AND INCLUDES BUTTERFLY VALVE COMPANION FLANGE.
64"φ ℄ EL. 90'-9"
NEGATIVE PRESSURE SAFETY RELIEF VALVE
BUTTERFLY VALVE N.O. (WEIGHT APPROX 3400#) SEE NOTE #3
FOR CONTINUATION SEE DWG. 9321-F-4024
FLEX. CONN
EL. 85'-3"
SEE NOTE #4
MATCH LINE
BUTTERFLY VALVE, LOCKED CLOSED (RETIRED IN PLACE) (SEE NOTE #3) (WEIGHT: APPROX. 3,400 LBS)
COOLING WATER RETURN CONN.
COOLING WATER SUPPLY CONN.
CLG. OUT
2" DRAIN TYP
3" DRAIN
16'-10" MAX. OVERALL
REINFORCING ANGLE TYP
CLOSE OFF SHEET
BAFFLE
FAN
HEAT EXCH
CLOSE OFF SHEET
REMOVABLE SECTION
DOOR
SEE DETAIL 1 AND NOTE 5
4" DRAIN
DRAIN PAN
DRAIN LINE BY OTHERS
VIBRATION ISOLATORS
℄ EL. 81'-3"
48"φ
4'-0" UNITS #21, 22, 23 & 24
AIR FLOW
FOR DETAILS OF OPENING IN F.C.U. WALL (SEE NOTE 2, SEE ALSO, DRAWING 312531)
NOTE:- THESE DOORS MUST BE OPEN WHEN CONTAINMENT BUILDING IS PRESSURIZED TO 50 PSIG FOR TESTING.
FLOOR EL. 68'-0"
43'-0"
(NOTE: DIMENSIONS TAKEN AT OUTER CASING AT 63 FOOT RADIUS)
NOTE: DRILL TWO ⅜"φ HOLES ONE ON EACH SIDE OF DEMISTER AND A MINIMUM OF 12" FROM DEMISTERS. HOLES SHALL BE PLUGGED AS DETAILED BELOW
WASHER
SILICONE GASKET
⅜ BOLT
ELEVATION A-A SCALE ½"=1'-0"
WORK ON CONTRACT 9321-01-45-24 STARTS HERE AND INCLUDES BUTTERFLY VALVE COMPANION FLANGE.
FLEX. CONN.
EL. 83'-3"
SEE NOTE #4
45"φ
℄ EL 79'-3"
7'-6" UNIT #25 ONLY
ELEVATION "AA-AA" SCALE ½"=1'-0"
NOTE:
1. FOR REFERENCE DRAWINGS AND NOTES SEE DWG. 9321-F-4023 & DWG. 312531
2. OPENING IS NOT SEALED. OPENING PROVIDES PRESSURE EQUALIZATION OF THE COMPARTMENT DURING A DESIGN BASIS ACCIDENT.
3. HOLES IN FOUNDATION OR FLANGE FOR BUTTERFLY VALVES SHALL BE DRILLED TO 125# STANDARD.
4. THIS DRAWING WAS A BID DRAWING FOR FAN COOLER-FILTER UNITS. FOR FINAL DESIGN OF THIS EQUIPMENT REFER TO MSAR DRAWINGS JK-2492-1 AND JK-2492-2
5. UNIT WALL PENETRATION IS TO BE LEFT ASSEMBLED, ALSO TO REMAIN IS THE 1-1/2" OF TUBE INSIDE OF THE UNIT & THE TUBE UP TO & INCLUDING THE COMPRESSION FITTING, LOCATED DIRECTLY OUTSIDE OF THE FAN COOLER UNIT
THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN ENTERGY AS: "CLASS A" ITEMS PER THE Q.A.P.D.
WORK THIS DWG WITH DWG. 9321-F-4025 & 4032
G. BHALLA 06/20/01 09
THIS REVISION IS CLASS "A" AS PER THE QAPD. PARTIALLY TRANSFERRED DMD9321-4026-AA REV.1 AND DMD9321-4026-AB REV.1 MOD. FMX-95-11243-M RELEASE AS CONSTRUCTED. PN 11243-95 JF/SW
WESTINGHOUSE ELECTRIC CORPORATION
CONTAINMENT BUILDING
AIR RECIR. FAN - COOLER - FILTER UNIT
PLAN AND SECTION -
UFSAR FIGURE No. 6.4-3
FOR
CONSOLIDATED EDISON COMPANY
INDIAN POINT GENERATING STATION
UNIT NO 2
NEW YORK 39732
UNITED ENGINEERS & CONSTRUCTORS INC.
9321-F-4026-10
A-201050

2/20/2004 11:30:38 PM 208479-0-25.DWG

MECH.

NOTES:

1. FOR GENERAL NOTE SEE DWG. 9321-F-2748
2. FLOW DIAGRAM SYMBOLS AS PER UE & C 9321-01-248-18
3. ALL NEW SOLENOID VALVES AT CONTAINMENT ARE LOCKED CLOSED
3. ALL SOLENOID VALVES ENERGIZED TO OPEN AND FAIL CLOSED
5. SAMPLE INLET LINES SHALL BE ELECTRICALLY HEAT TRACED TO TO MAINTAIN TEMPERATURE BETWEEN 280 -°320 F°
6. ALL VALVES ARE 3/8" O.D. S.W. ENDS UNLESS OTHERWISE INDICATED, S.S. 3/8"-2500#
7. PIPING AND OR TUBING SHALL BE IN ACCORDANCE WITH UE & C SPEC. 9321-01-248 CLASS 153R AND CLASS 2505R
8. THE QUALITY GROUP A,B,C AND SEISMIC BOUNDARIES EXTEND TO THE FIRST SEISMIC SUPPORT/RESTRAINT BEYOND THE BOUNDARIES SHOWN.

EVERYTHING ON THIS DRAWING (EXCEPT DASHED LINES REPRESENTING PIPING) IS PART OF CONTAINMENT BUILDING POST ACCIDENT CONTAINMENT AIR SAMPLING SYSTEM (PACAS) DISCRIBED AS PART OF POST ACCIDENT HYDROGEN CONTROL SYSTEM LISTED IN EXHIBIT A, CI-240-1 EXCEPT AS SPECIFICALLY INDICATED HEREIN

N_2 NITROGEN SUPPLY SYSTEM

REFERENCE DWGS.

- 9321-C-2016 – FLOW DIAG SYMBOLS
- 9321-F-2748 – CONTAINMENT BLDG. POST ACCIDENT CONTAINMENT SAMPLING SYSTEM
- 9321-F-7040 – PAB INSTRUMENT ARRANREMENT SHEET NO. 1
- 9321-F-7041 – DITTO – SHEET NO. 2
- 9321-F-2725 – FLOW DIAGRAM AUTOMATIC GAS ANAYLZER SYSTEM
- 9321-F-2723 – FLOW DIAGRAM NITROGEN TO NUCLEAR EQUIPMENT
- 9321-F-2745 – FLOW DIAGRAM SAMPLING SYSTEM
- 9321-F-2730 – WASTE DISPOSAL SYSTEM SHEET NO. 2 .

VENDOR DWGS.

- 099620 – COMSIP INC. MODEL K-IV PIPING AND INSTR. DIAGRAM

THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLED WITHIN ENTERGY AS

"CLASS A" ITEMS

PER THE QAPD

COMPUTER GENERATED DRAWING NOT TO BE HAND REVISED

REVISION	REVIEWER		DATE	CKR.	SUPV.	DESIGN ENG	DISC. ENG.	ENG. MGR.
	G.BHALLA 7/29/02 ENG	THIS REVISION IS NON CLASS PER QAPD. UPDATED DRAWING. PER CR# 200206324 AND FIELD VERIFICATION. DELETED VALVE #5723. RELEASED AS CONSTRUCTED. P.N. 69901-AF D.B./G.B. 7/29/02	12-19-2001			02-01-01		
				REVIEW		APPROVALS		

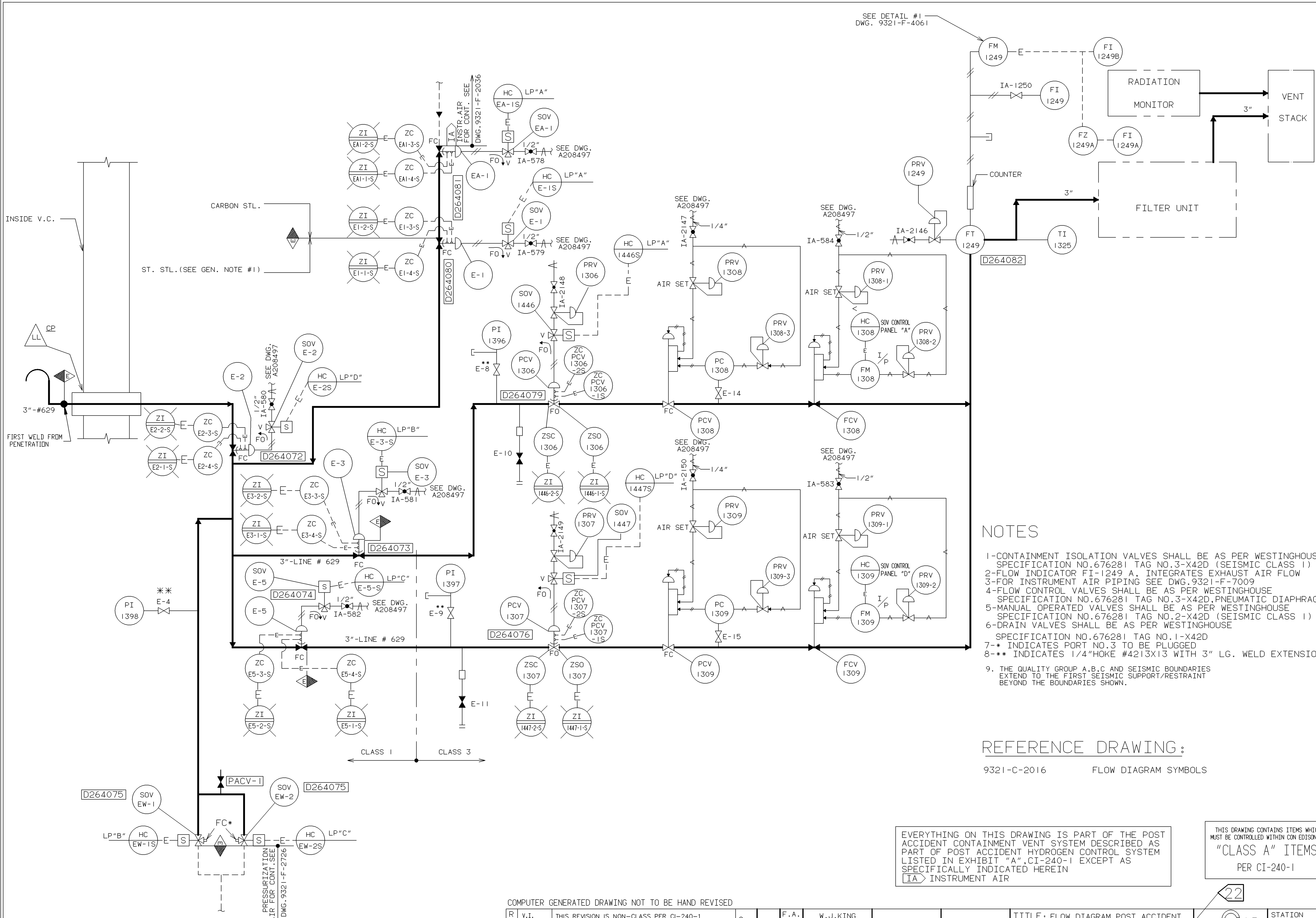
TITLE: FLOW DIAGRAM CONTAINMENT BLD. POST ACCIDENT CONTAINMENT AIR SAMPLING SYST. – UFSAR FIGURE No. 6.8-4

DRWN. BY D. BALADO | SCALE | REC'D

Con Edison | STATION: INDIAN POINT

DWG. NO. A208479 -25 | MY

MECHANIC



NOTES

1-CONTAINMENT ISOLATION VALVES SHALL BE AS PER WESTINGHOUSE SPECIFICATION NO.676281 TAG NO.3-X42D (SEISMIC CLASS 1)
2-FLOW INDICATOR FI-1249 A. INTEGRATES EXHAUST AIR FLOW
3-FOR INSTRUMENT AIR PIPING SEE DWG.9321-F-7009
4-FLOW CONTROL VALVES SHALL BE AS PER WESTINGHOUSE SPECIFICATION NO.676281 TAG NO.3-X42D,PNEUMATIC DIAPHRAGM
5-MANUAL OPERATED VALVES SHALL BE AS PER WESTINGHOUSE SPECIFICATION NO.676281 TAG NO.2-X42D (SEISMIC CLASS 1)
6-DRAIN VALVES SHALL BE AS PER WESTINGHOUSE SPECIFICATION NO.676281 TAG NO.1-X42D
7-* INDICATES PORT NO.3 TO BE PLUGGED
8-** INDICATES 1/4"HOKE #4213X13 WITH 3" LG. WELD EXTENSION
9. THE QUALITY GROUP A.B.C AND SEISMIC BOUNDARIES EXTEND TO THE FIRST SEISMIC SUPPORT/RESTRAINT BEYOND THE BOUNDARIES SHOWN.

REFERENCE DRAWING:

9321-C-2016 FLOW DIAGRAM SYMBOLS

EVERYTHING ON THIS DRAWING IS PART OF THE POST ACCIDENT CONTAINMENT VENT SYSTEM DESCRIBED AS PART OF POST ACCIDENT HYDROGEN CONTROL SYSTEM LISTED IN EXHIBIT "A",CI-240-1 EXCEPT AS SPECIFICALLY INDICATED HEREIN
IA INSTRUMENT AIR

THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN CON EDISON AS:
"CLASS A" ITEMS
PER CI-240-1

22

COMPUTER GENERATED DRAWING NOT TO BE HAND REVISED

REVISION							
V.I. 11/15/99	THIS REVISION IS NON-CLASS PER CI-240-1 UPDATED DWG. PER FIELD CONDITIONS RELEASED AS RECORD P/N 69911-NP PP/VI 11/12/99	2/5/87	A.G.	F.A.	W.J.KING 2/5/87		
DES		DATE	CKR.	SUPV.	DESIGN ENG	DISC. ENG.	ENG. MGR.
ENG			REVIEW		APPROVALS		

TITLE: FLOW DIAGRAM POST ACCIDENT CONTAINMENT VENTING SYSTEM - UFSAR FIGURE No. 6.8-3

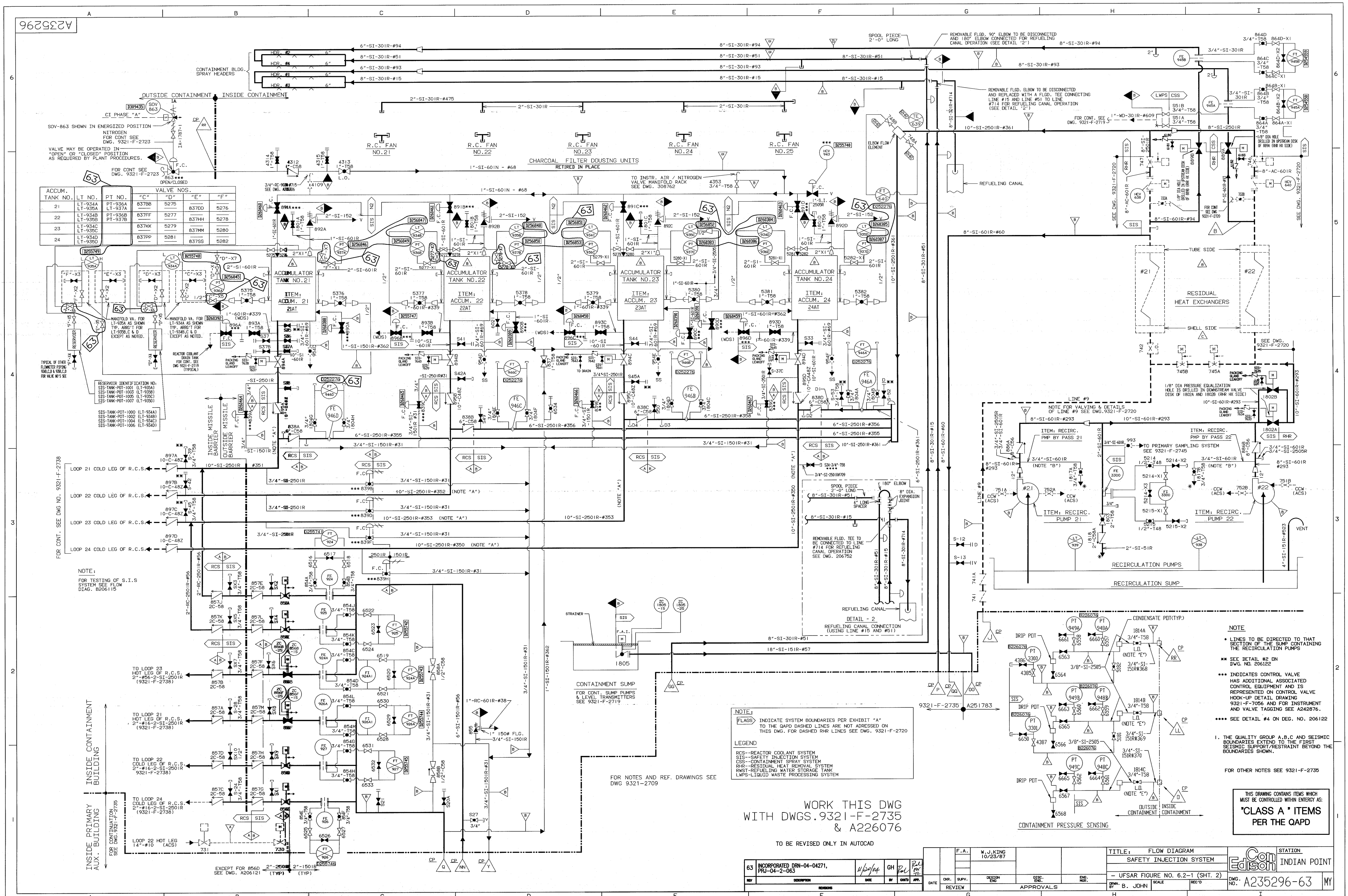
MOD.PROC.OR SPEC.
DRWN. BY D.BALADO | SCALE NONE | REC'D

Con Edison

STATION: INDIAN POINT

DWG. NO. B208879-22 MY

A235296



ACCUM. TANK NO.	LT NO.	PT NO.	VALVE NOS. "C"	"D"	"E"	"F"
21	LT-934A LT-935A	PT-936A LT-937A	837BB	5275	837DD	5276
22	LT-934B LT-935B	PT-936B PT-937B	837FF	5277	837HH	5278
23	LT-934C LT-935C		837KK	5279	837MM	5280
24	LT-934D LT-935D		837PP	5281	837SS	5282

NOTE:

FLAGS INDICATE SYSTEM BOUNDARIES PER EXHIBIT "A" TO THE QAPD DASHED LINES ARE NOT ADRESSED ON THIS DWG. FOR DASHED RHR LINES SEE DWG. 9321-F-2720

LEGEND

RCS--REACTOR COOLANT SYSTEM
SIS--SAFETY INJECTION SYSTEM
CSS--CONTAINMENT SPRAY SYSTEM
RHR--RESIDUAL HEAT REMOVAL SYSTEM
RWST-REFUELING WATER STORAGE TANK
LWPS-LIQUID WASTE PROCESSING SYSTEM

NOTE

* LINES TO BE DIRECTED TO THAT SECTION OF THE SUMP CONTAINING THE RECIRCULATION PUMPS

** SEE DETAIL #2 ON DWG. NO. 206122

*** INDICATES CONTROL VALVE HAS ADDITIONAL ASSOCIATED CONTROL EQUIPMENT AND IS REPRESENTED ON CONTROL VALVE HOOK-UP DETAIL DRAWING 9321-F-7056 AND FOR INSTRUMENT AND VALVE TAGGING SEE A242876.

**** SEE DETAIL #4 ON DEG. NO. 206122

1. THE QUALITY GROUP A.B.C AND SEISMIC BOUNDARIES EXTEND TO THE FIRST SEISMIC SUPPORT/RESTRAINT BEYOND THE BOUNDARIES SHOWN.

FOR OTHER NOTES SEE 9321-F-2735

THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN ENTERGY AS:

"CLASS A" ITEMS PER THE QAPD

WORK THIS DWG WITH DWGS.9321-F-2735 & A226076

TO BE REVISED ONLY IN AUTOCAD

REV	DESCRIPTION	DATE	BY	CHK'D	APP.
63	INCORPORATED DRN-04-04271, PRJ-04-2-063	11/20/04	GH	Rul	Rul for TC

REVISIONS

F.A. W.J.KING 10/23/87

DATE | CKR. | SUPV. | DESIGN ENG | DISC. ENG. | ENG. MGR.

REVIEW | APPROVALS

TITLE: FLOW DIAGRAM
SAFETY INJECTION SYSTEM
– UFSAR FIGURE NO. 6.2–1 (SHT. 2)

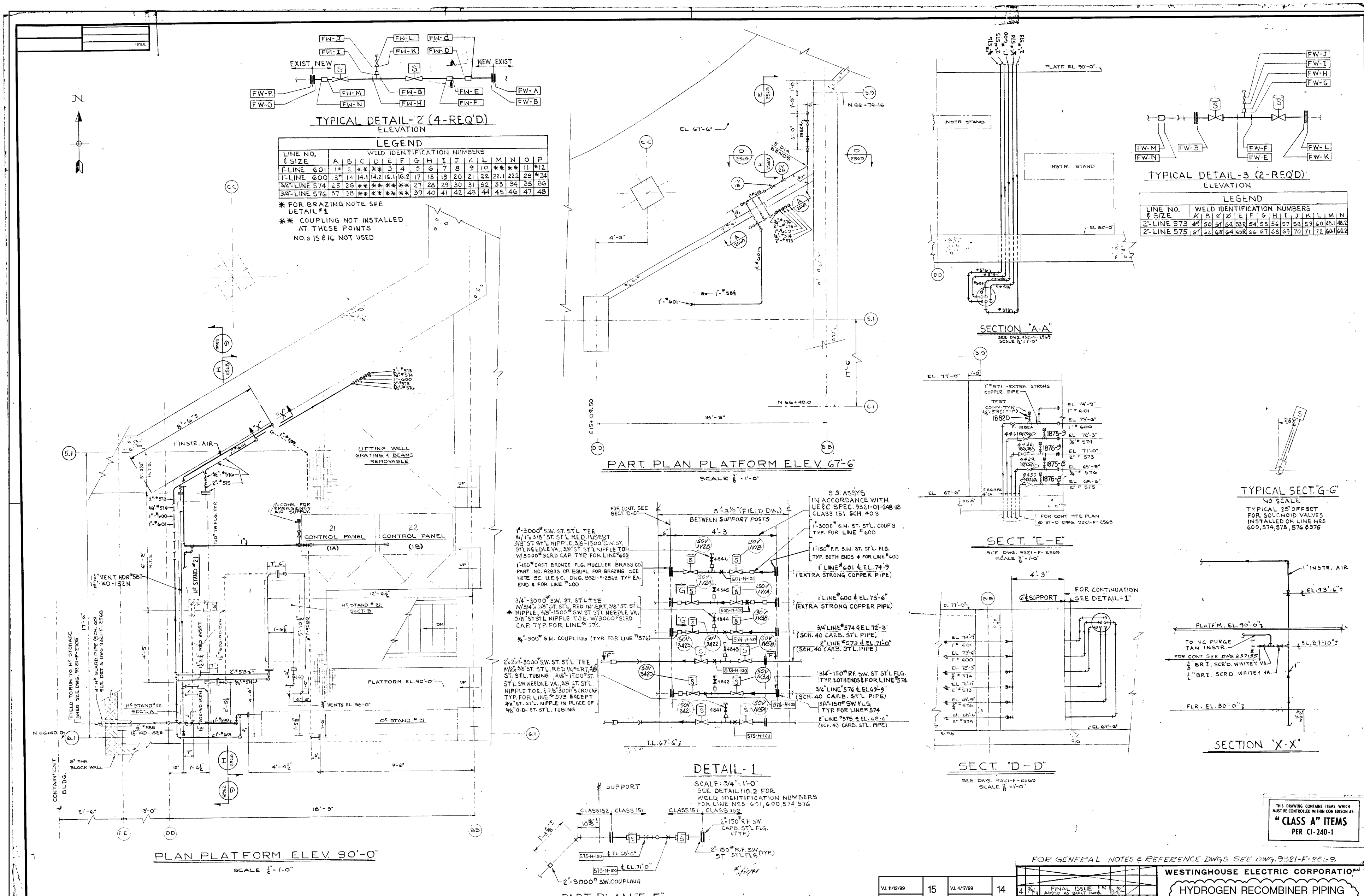
DRWN. BY B. JOHN | SCALE | REC'D

Con Edison

STATION INDIAN POINT

DWG. NO. A235296-63 MY

2/20/2004 10:05:17 AM 9321-2569-0-15.dwg



LEGEND (Typical Detail-2)

LINE NO. & SIZE	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P
1"-LINE 601	1*	2	**	**	3	4	5	6	7	8	9	10	**	**	11	*12
1"-LINE 600	.3*	14	14.1	14.2	16.1	16.2	17	18	19	20	21	22	22.1	22.2	23	*24
3/4"-LINE 574	25	26	**	**	**	**	27	28	29	30	31	32	33	34	35	36
3/4"-LINE 576	37	38	**	**	**	**	39	40	41	42	43	44	45	46	47	48

* FOR BRAZING NOTE SEE DETAIL #1.

** COUPLING NOT INSTALLED AT THESE POINTS

NO.s 15 & 16 NOT USED

LEGEND (Typical Detail-3)

LINE NO. & SIZE	A	B	C	D	E	F	G	H	I	J	K	L	M	N
2"-LINE 573	~~49~~	50	~~51~~	~~52~~	53R	54	55	56	57	58	59	60	48.1	48.2
2"-LINE 575	~~61~~	62	~~63~~	~~64~~	65R	66	67	68	69	70	71	72	60.1	60.2

THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN CON EDISON AS:
"CLASS A" ITEMS
PER CI-240-1

FOR GENERAL NOTES & REFERENCE DWGS. SEE DWG. 9321-F-2568

V.I. 11/12/99	15
THIS REVISION IS NON-CLASS PER CI-240-1 UPDATED DRAWING TITLE BLOCK No. IN TITLE P/N 69911-NP RELEASED FOR RECORD PP/VI 11/12/99	

V.I. 4/17/99	14
THIS REVISION IS NON-CLASS PER CI-240-1 UPDATED DRAWING TO INCLUDE UFSAR FIGURE No. IN TITLE P/N 69911-NP RELEASED FOR RECORD DLA/4/17/99	
STRATUS ENGINEERING	

REV NO.	DATE	DESCRIPTION	ENGR	SUP ENG
4	10/9/73	FINAL ISSUE ADDED AS BUILT INFO.		
3	9/14/	REVISED SCHEDULE OF COPPER PIPE	JK	
2		ADDED CONTROL PANEL DESIG.	FJS	
1		ADDED EMER. AIR SUPPLY ... PER ... 3125	JK	
		FIRST ISSUE		

ENGINEER
STATE REG NO.

UNITED ENGINEERS & CONSTRUCTORS INC.

WESTINGHOUSE ELECTRIC CORPORATION

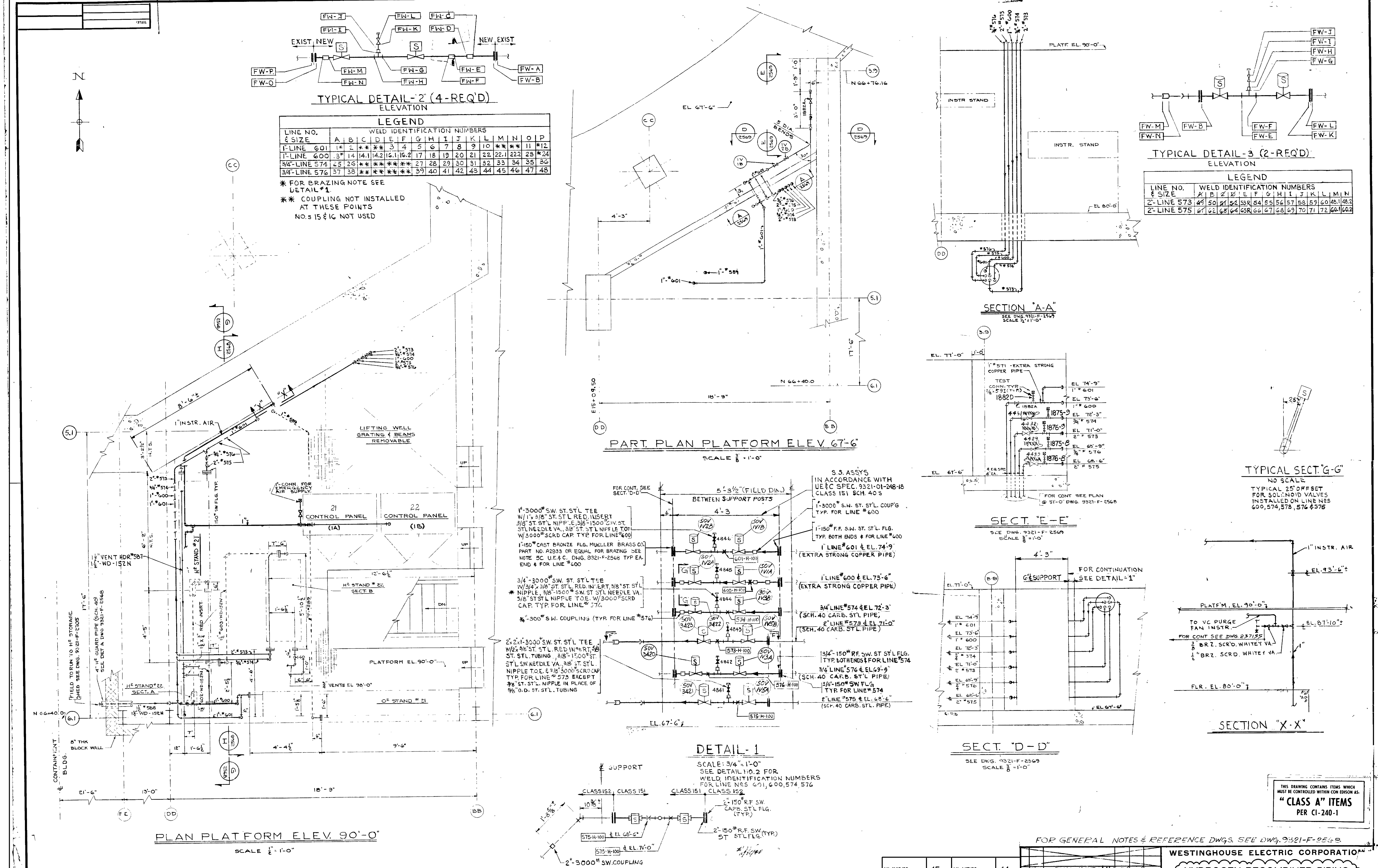
HYDROGEN RECOMBINER PIPING
SHEET No. 2

FOR

CONSOLIDATED EDISON COMPANY
INDIAN POINT GENERATING STATION
UNIT NO. 2

U.E.&C. DWG. NO. 9321-F-2569-15

CON. ED. CO. DWG. A 2009



LEGEND (Typical Detail-2)

LINE NO. & SIZE	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O	P
1"-LINE 601	1*	2	**	**	3	4	5	6	7	8	9	10	**	**	11	*12
1"-LINE 600	.3*	14	14.1	14.2	16.1	16.2	17	18	19	20	21	22	22.1	22.2	23	*24
3/4"-LINE 574	25	26	**	**	**	**	27	28	29	30	31	32	33	34	35	36
3/4"-LINE 576	37	38	**	**	**	**	39	40	41	42	43	44	45	46	47	48

(Columns A–P: WELD IDENTIFICATION NUMBERS)

* FOR BRAZING NOTE SEE DETAIL #1.

** COUPLING NOT INSTALLED AT THESE POINTS

NO.s 15 & 16 NOT USED

LEGEND (Typical Detail-3)

LINE NO. & SIZE	~~A~~	B	~~C~~	~~D~~	E	F	G	H	I	J	K	L	M	N
2"-LINE 573	~~49~~	50	~~51~~	~~52~~	53R	54	55	56	57	58	59	60	48.1	48.2
2"-LINE 575	~~61~~	62	~~63~~	~~64~~	65R	66	67	68	69	70	71	72	60.1	60.2

(Columns A–N: WELD IDENTIFICATION NUMBERS)

THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN CON EDISON AS:
"CLASS A" ITEMS
PER CI-240-1

FOR GENERAL NOTES & REFERENCE DWGS. SEE DWG. 9321-F-2568

V.I. 11/12/99 — 15
THIS REVISION IS NON-CLASS PER CI-240-1 UPDATED DRAWING TITLE BLOCK No. IN TITLE P/N 69911-NP RELEASED FOR RECORD PP/VI 11/12/99

V.I. 4/17/99 — 14
THIS REVISION IS NON-CLASS PER CI-240-1 UPDATED DRAWING TO INCLUDE UFSAR FIGURE No. IN TITLE P/N 69911-NP RELEASED FOR RECORD DLA/4/17/99

STRATUS ENGINEERING

REV NO.	DATE	DESCRIPTION	ENGR	SUP ENG
4	[illegible]	FINAL ISSUE ADDED AS BUILT INFO.	[illegible]	
3	[illegible]	REVISED SCHEDULE OF COPPER PIPE	JK	
2	[illegible]	ADDED CONTROL PANEL DESIG. [illegible]	FJS	
1	[illegible]	ADDED EMER AIR SUPPLY [illegible]	JK	
	[illegible]	FIRST ISSUE [illegible]	DH	

ENGINEER
STATE REG NO.

UNITED ENGINEERS & CONSTRUCTORS INC.

WESTINGHOUSE ELECTRIC CORPORATION

HYDROGEN RECOMBINER PIPING
SHEET No. 2

FOR
CONSOLIDATED EDISON COMPANY
INDIAN POINT GENERATING STATION
UNIT NO. 2

U.E.&C. DWG. NO. 9321-F-2569- 15
CON. ED. CO. DWG. A [illegible]

THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN CON EDISON AS:
"CLASS A" ITEMS PER THE QAPD

REFERENCE DRAWINGS

9321-C-2016 FLOW DIAGRAM SYMBOLS
9321-F-7033 PENETRATION AND LINER WELD CHANNEL PIPING - SHT.1
9321-F-7034 - SHT.2
9321-F-7035 - SHT.3
9321-F-7039 - SHT.4
9321-F-7052 - SHT.5
9321-F-7019 - SHT.6

NOTES:
THE QUALITY GROUP A,B,C AND SEISMIC BOUNDARIES EXTEND TO THE FIRST SEISMIC SUPPORT/RESTRAINT BEYOND THE BOUNDARIES SHOWN.

COMPUTER GENERATED DRAWING NOT TO BE HAND REVISED

REVISION	DES		DATE		CKF.	SUPV.	DESIGN ENG	DISC. ENG.	ENG. MGR.
	G. BHALLA 05/15/01	THIS REVISION IS NON-CLASS PER THE QAPD. UPDATED DRAWING PER CR #200104549 RELEASED AS-BUILT P.N. 69901-AF JWR/S&L	9/15/87	E.C.	F.A.		W.J.KING 9-15-87		
				REVIEW			APPROVALS		

TITLE: FLOW DIAGRAM-PENETRATION AND LINER WELD JOINT CHANNEL PRESSURIZATION SYSTEM - UFSAR FIGURE No. 6.6-1

Con Edison — STATION: INDIAN POINT

DRWN. BY: GIBBS&HILL — SCALE: NONE — REC'D

DWG. NO. 9321-F-2726-75 MY

NOTE:
ORIGINAL INFORMATION SHOWN ON THIS DWG. WAS TRANSFERRED FROM FABRICATING ENGINEERING CORP. DWG.# D-11483 (A207655)

NOTE
PER MMC-80-02-13 TWELVE SOLENOID OPERATED VALVES NUMBERS SOV-1V1A,-1V2A,-1V1B,-1V2B,-1V3A,-1V3B, 1V5A,-1V5B,-3420,-3421,-3422 & -3423 ARE INSTALLED SO THAT THE FLOW ARROW STAMPED ON THE VALVE BODY ARE OPPOSITE TO THE NORMAL FLOW TO THE CONTAINMENT HYDROGEN RECOMBINERS.THIS WAY UNDER THE POSTULATED ** ACCIDENT CONDITION THE HIGH CONTAINMENT PRESSURE WILL NOT LIFT THEM OPEN.

THE QUALITY GROUP A,B,C & SEISMIC BOUNDARIES EXTEND TO THE FIRST SEISMIC SUPPORT/RESTRAINT BEYOND THE BOUNDARIES SHOWN

REFERENCE DWG.

9321-C-2016 - FLOW DIAGRAMS SYMBOLS

THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN CON EDISON AS: "CLASS A" ITEMS PER CI-240-1

COMPUTER GEN. DWG. NOT TO BE HAND REVISED.

THIS REVISION IS CLASS 'A' AS PER THE QAPD PARTIALLY TRANSFERRED DMD9321-2727-AE REV. 0 AND DMD9321-2727-AF REV. 1 MOD. FMX-95-10912-M RELEASE AS CONSTRUCTED. P/N 10912-95 JF/SW

G. BHALLA 06/26/01

W.J.KING 6-19-87

TITLE: FLOW DIAGRAM
HYDROGEN RECOMBINER SYSTEM
- UFSAR FIGURE No. 6.8-5

DRWN. BY B.JOHN — SCALE NONE

Con Edison — STATION: INDIAN POINT

DWG. NO. 9321-F-2727-40 MY

2/21/2004 1:04:13 PM 9321-2735-0-136.DWG

9321-F-2735

REFERENCE DRAWINGS: U.E. & C. NO.

RCS- -REACTOR COOLANT SYSTEM -----9321-F-2738
CVSC -CHEMICAL AND VOLUME CONTROL SYSTEM
SHEET #1. ------9321-F-2736
SHEET #2. ------9321-F-2737
SHEET #3. ------9321-F-2737
ACS- -AUXILIARY COOLANT SYSTEM ----9321-F-2720
SS- -SAMPLING SYSTEM ------9321-F-2745
WDS- -WASTE DISPOSAL SYSTEM
SHEET #1. ------9321-F-2719
SHEET #2. ------9321-F-2730
IVSWS-ISOLATION VALVE SEAL WATER SYS.---9321-F-2746
N2- -NITROGEN TO NUCLEAR EQUIPMENT - 9321-F-2723
TEMPERATURE STRAINER TABULATION --9321-H-2739

REFERENCES:

1. PROCESS FLOW DIAGRAM DWG.541-F-154
2. DEFINITION OF SYMBOLS
   E SPEC. G675176 REV.2 AND
   E SPEC. G675164 REV. 0
3. INSTALLATION OF INSTRUMENTATION
   PROC. SPEC. CAP.-294367 REV.1
4. MATERIAL SPEC. PIPE AND FITTINGS
   E. SPEC G569866 REV. 2 AND
   E. SPEC G676398 REV. 0

NOTES:

A. PIPING IS SCHEDULE 140
B. LOCATE IN HIGH SIDE OF PIPE IN LOCAL HIGH POINT.
D. ORIFICE IS SUPPLIED WITH PUMP PACKAGE.
E. MOUNT WITH STEM VERTICAL
F. BALANCE PISTON FLOW LINE FOR S.I.PUMP #22 MUST BE INSIDE VALVES 887A & B.
G. LOCAL NITROGEN BOTTLE STATION - SEE DWG. 9321-F-2723.
H. LOCATE AT LOW POINT OF PIPE WORK.
J. INSTALL VALVES FOR FOR VERTICAL DISCHARGE.
K. FLUSH LINE TO BE INSTALLED AS CLOSE TO VALVE INLET AS POSSIBLE
L. HEAT TREATING TO CONTINUE APPROXIMATELY ONE FOOT PAST THE JUNCTURE.
M. *** INDICATES CONTROL VALVE HAS ADDITIONAL ASSOCIATED CONTROL EQUIPMENT & IS REPRESENTED ON CONTROL VALVE HOOK-UP DETAIL DWG. 9321-F-7056.
N. ALL BORIC ACID HEAT TRACE CIRCUITS HAVE BEEN DISCONNECTED. RWST FREEZE PROTECTION CIRCUITS REMAIN ENERGIZED.
O. THE QUALITY GROUP A,B,C AND SEISMIC BOUNDARIES EXTEND TO THE FIRST SEISMIC SUPPORT/RESTRAINT BEYOND THE BOUNDARIES SHOWN.

NOTES:
1. THE QUALITY GROUP A,B,C & SEISMIC BOUNDARIES EXTEND TO THE FIRST SEISMIC SUPPORT/RESTRAINT BEYOND THE BOUNDARIES SHOWN.
2. FLAGS INDICATE SYSTEM BOUNDARIES PER EXHIBIT "A" TO CI-240-1. DASHED LINES ARE NOT ADDRESSED ON THIS DRAWING FOR DASHED RHR LINES SEE DWG.9321-F-2720 & DWG A251783

LEGEND:
SIS - SAFETY INJECTION SYSTEM.
CSS - CONTAINMENT SPRAY SYSTEM.
RHR - RESIDUAL HEAT REMOVAL SYSTEM.
RWST- REFUELING WATER STORAGE TANK.
LWPS- LIQUID WASTE PROCESSING SYSTEM.
SFPCS- SPENT FUEL POOL COOLING SYSTEM.

REFERENCE DRAWINGS
9321-C-2016------ FLOW DIAGRAM SYMBOLS

WORK THIS DWG. WITH A235296

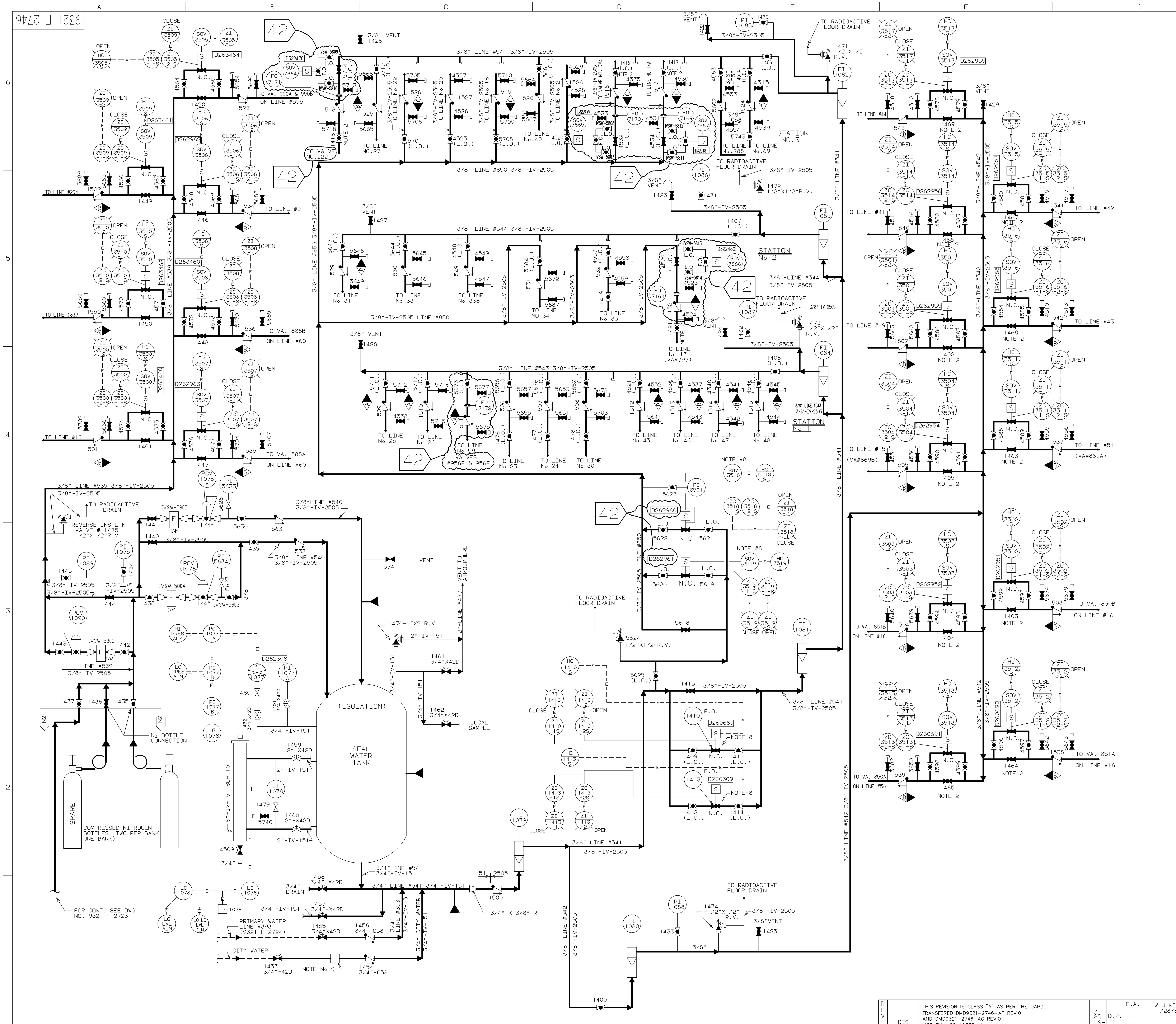
THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN ENTERGY AS:
"CLASS A" ITEMS
PER THE QAPD

THIS DWG. TO BE REVISED ONLY IN AUTOCAD.

REVISION	DES	THIS REVISION IS NON-CLASS PER THE QAPD. DRAWING UPDATED IN RESPONSE TO CR-IP2-2004-00396 RELEASE FOR RECORD	2/12/88	A.G.	F.A.	W.J.KING 2/12/88
	T.CZEMIEWSKI 02/11/04 ENG	PN 69901-AF RL/TC	DATE	CKR.	SUPV.	DESIGN MGR. / CHIEF DESIGN ENG. / DISC. ENG. / ENG. MGR.
			REVIEW			APPROVALS

TITLE: FLOW DIAGRAM
SAFETY INJECTION SYSTEM
- UFSAR FIGURE No. 6.2-1 (SHT. 1)
DRAWN BY: B.JOHN SCALE: NONE

Con Edison
STATION: INDIAN POINT
DWG. NO. 9321-F-2735-136 MY



PENETRATION LIST

SEAL CONN TO LINE NO.	INJECTION POINTS	CONTAINMENT PENETRATION SLEEVE	LINE DESCRIPTION	Q/G	REMARKS
9	VALVE #744	J	RESIDUAL HEAT REMOVAL LOOP IN	B	NITROGEN INJ.
10	VALVE #732	K	DITTO OUT	B	DITTO
337	IN LINE #337	NONE	DITTO OUT	B	DITTO
13	VALVE #797	N	R.C. PUMPS COOLING WATER IN	C	
14	VALVE #784	O	DITTO OUT	C	
14A	VALVE FCV #625	O	DITTO OUT	C	
15	VALVE #869B	P	CONTAINMENT SPRAY HEADER	B	
16	VALVE #850B	Q	SAFETY INJECTION LINE	B	IN P.A.B.
16	VALVE #851B	Q	DITTO	B	DITTO
17	VALVE #222	R	R.C. PUMPS SEAL WATER RETURN	B	
18	IN LINE #18	R	EXCESS LETDOWN HT. EXCH. COOLING WTR.RET.	C	
19	IN LINE #19	R	CHARGING LINE	B	
20	VALVE #UH43	S	AUXILIARY STEAM	NONE	
21	VALVE #UH44	T	AUXILIARY STEAM CONDENSATE	NONE	
22	IN LINE #22	U	EXCESS LETDOWN HT. EXCH. COOLING WTR. SUPPLY	C	
23	IN LINE #23	V	CONTAINMENT VENT HEADER	NONE	
24	IN LINE #24	V	P.R.T. GAS ANALYZER LINE	NONE	
25	IN LINE #25	W	PRESSURIZER STEAM SAMPLE	A	
26	IN LINE #26	W	DITTO LIQUID DITTO	A	
27	IN LINE #27	X	LETDOWN LINE	B	
30	IN LINE #30	V	R.C. DRAIN TANK GAS ANALYZER LINE	NONE	
31	IN LINE #31	Y	S.I.S. TEST LINE	B	
33	IN LINE #33	Y	P.R.T. MAKE-UP LINE	NONE	
34	IN LINE #34	Y	SERVICE AIR	NONE	
35	IN LINE #35	Y	CITY WATER	NONE	
40	IN LINE #40	Z	R.C. DRAIN TANK PUMP DISCHARGE	NONE	
41	IN LINE #41	Z	R.C. PUMP SEAL WATER SUPPLY	B	
42	IN LINE #42	Z	DITTO	B	
43	IN LINE #43	Z	DITTO	B	
44	IN LINE #44	Z	DITTO	B	
45	IN LINE #45	AA	STEAM GENERATOR BLOWDOWN LINE	B	
46	IN LINE #46	BB	DITTO	B	
47	IN LINE #47	CC	DITTO	B	
48	IN LINE #48	DD	DITTO	B	
51	VALVE #869A	GG	CONTAINMENT SPRAY HEADER	B	
294	IN LINE #294	NONE	RESIDUAL HEAT REMOVAL LOOP SAMPLE	B	NITROGEN INJ.
16	VALVE #851A	NN	SAFETY INJECTION LINE	B	IN P.A.B.
56	VALVE #850A	NN	DITTO	B	DITTO
59	IN LINE #59	W	REACTOR COOLANT SAMPLE LINE	A	
60	VALVE #888A	QQ	RESIDUAL HT.EXCH. TO SAFETY INJ. PUMPS	B	NITROGEN INJ.
60	VALVE #888B	QQ	DITTO	B	DITTO
69	IN LINE #69	RR	ACCUMULATOR SAMPLE LINE	B	
338	IN LINE #338	Y	SUMP PUMP DISCHARGE	NONE	
595	IN LINE #595	TT	RECIRC. PUMP DISCH. SAMPLE	B	NITROGEN INJ.
788			HRSS RETURN TO VC SUMP	NONE	

REFERENCES:

1. PROCESS FLOW DIAGRAM-------------- (W)
2. DEFINITIONS OF SYMBOLS ----------- (W) SPEC. G675176 REV D.
3. INSTRUMENT & CONTROL STD --------- (W) SYMBOLS & APPLICATIONS FOR INST. DIAGRAMS SECTION - 1.1,ISSUED AUG.12 1966.
   (W) INST. INSTALLATION: SECT.3.0 ISSUE
4. MATERIAL SPEC & FITTINGS --------- (W) SPEC. G.569866 REV.2
   (W) SPEC. G.676398 REV. 0

REFERENCE DRAWINGS

UE & C. DWGS
9321-F-2738 ------------- REACTOR COOLANT SYSTEM
9321-F-2736 ------------- CHEMICAL VOLUME CONTROL SH. 1

9321-F-2737 ------------- CHEMICAL VOLUME CONTROL SH. 2
CHEMICAL VOLUME CONTROL SH. 3
9321-F-2719 ------------- WASTE DISPOSAL SYSTEM SH. 1
9321-F-2730 ------------- WASTE DISPOSAL SYSTEM SH. 2

9321-F-2735 ------------- SAFETY INJECTION SYSTEM
9321-F-2745 ------------- SAMPLE SYSTEM
9321-F-2729 ------------- STEAM GEN. BLOWDOWN SYSTEM
9321-F-2720 ------------- AUXILIARY COOLANT
9321-F-2724 ------------- PRIMARY WATER MAKE-UP SYSTEM
9321-F-2768 ------------- ISOLATION SEAL WATER PIPING ARRGT.
9321-F-2769 ------------- ISOLATION SEAL WATER PIPING ARRGT.
9321-C-2016 ------------- FLOW DIAGRAM SYMBOLS
B209776 ------------- I.V.S.W.S. INJECTION LINE TEST CONNECTIONS

GENERAL NOTES

1. ALL PIPING TO BE 3/8" S.S., O.D. TUBING (LINE DESIGNATION 3/8" - IV-2505 (EXCEPT AS NOTED).
2. ALL GLOBE VALVES INSTALLED WITH FLOW UNDER SEAT, EXCEPT NO. S 1402-1405.1416-1418. 1421, 1463-1469
3. ALL GLOBE VALVES ARE TYPE 3/8" T58 EXCEPT NO. S 1434-1448 (NITROGEN-SUPPLY) WHICH ARE TYPE 3/8" T58S(SOFT SEATS AND LARGE ORIFICES)
4. ALL CHECK VALVES ARE TYPE 3/8"-C58,EXCEPT NO. S 1454,1456 & 1533 TO 1536. VALVE 1533-1536 ARE TYPE 3/8"-C58S (SOFT SEAT).
5. ALL GLOBE & CHECK VALVES ARE WELDED OR EQUIPTED WITH SWAGELOK FITTINGS, (SWAGELOK PREFERRED FOR MAINTENANCE EASE)
6. ADDITIONAL VENTS AND DRAINS MAY BE REQUIRED.
7. THE INDIVIDUAL SEAL WATER LINES ARE ROUTED TO THE CONTAINMENT ISOLATION POINTS BY THE LINE NUMBERS INDICATED.
8. VALVES 1410, 1413, SOV-35-18 & SOV-3519 ARE OPENED AUTOMATICALLY BY A PHASE "A" CONTAINMENT ISOLATION SIGNAL.
9. FLANGES AND BLANK OFF DISC.
10. THE QUALITY GROUP A,B,C AND SEISMIC BOUNDARIES EXTEND TO THE FIRST SEISMIC SUPPORT/RESTRAINT BEYOND THE BOUNDARIES SHOWN.

LEGEND

PW----------------PRIMARY WATER
FC----------------FAIL CLOSE
CW----------------CITY WATER
N.C. ------------NORMALLY CLOSED
LO----------------LOCKED OPEN

THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN CON EDISON AS:
"CLASS A" ITEMS
PER CI-240-1

REVISION			DATE						
DES G. BHALLA 06/04/01 ENG	THIS REVISION IS CLASS "A" AS PER THE QAPD TRANSFERED DMD9321-2746-AF REV.0 AND DMD9321-2746-AG REV.0 MOD FMX-00-12338-M. RELEASED AS CONSTRUCTED PN 12338-00 JF/DB	1/28/87	D.P.	F.A.	W.J.KING 1/28/87				
		DATE	CKR.	SUPV.	DESIGN ENG	DISC. ENG.	ENG. MGR.		
			REVIEW		APPROVALS				

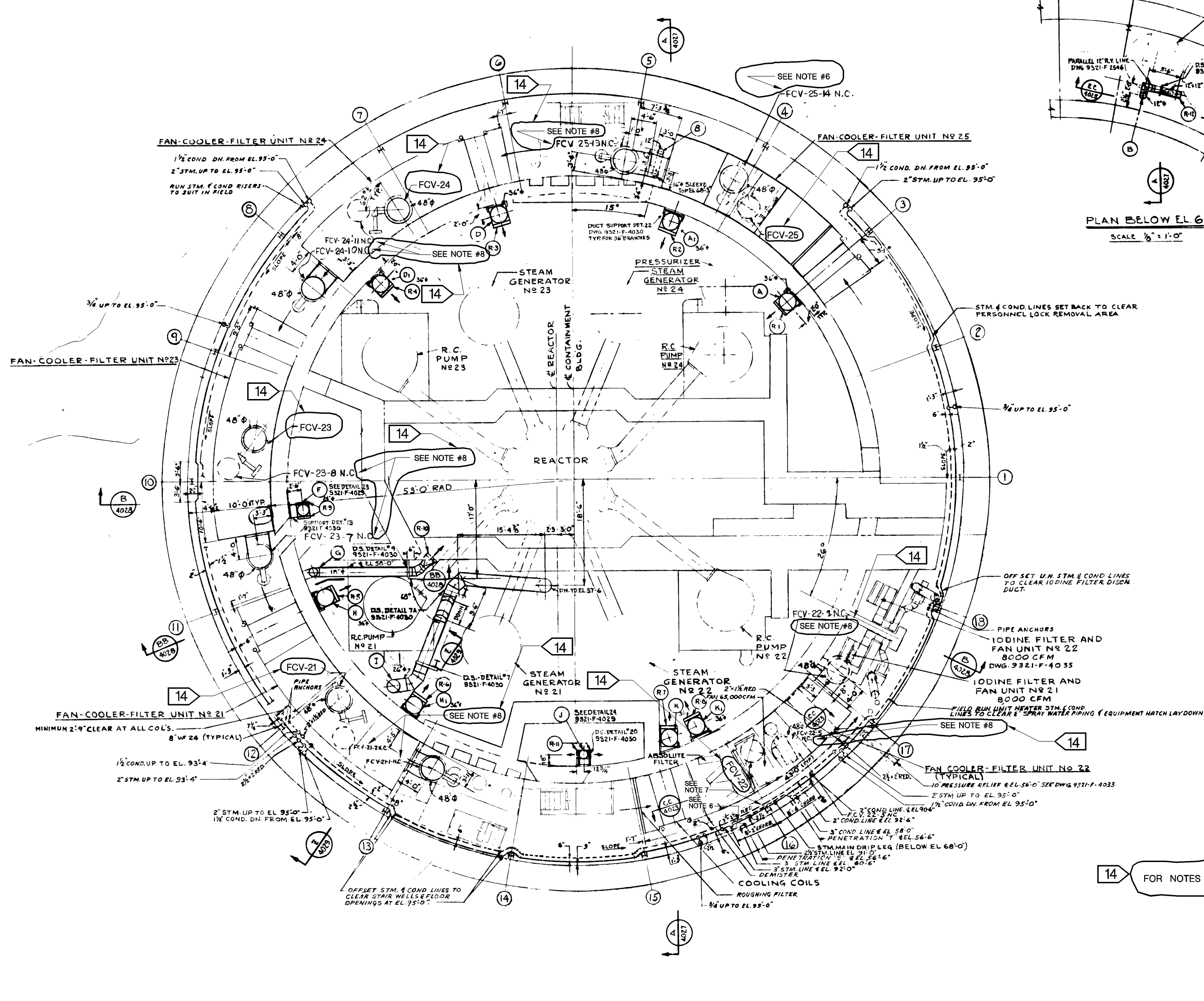
TITLE: FLOW DIAGRAM
ISOLATION VALVE SEAL
WATER SYSTEM
- UFSAR FIGURE No. 6.5-1

DRWN. BY GIBBS&HILL | SCALE NONE | REC'D

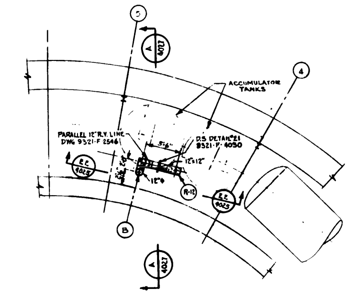
Con Edison

STATION INDIAN POINT

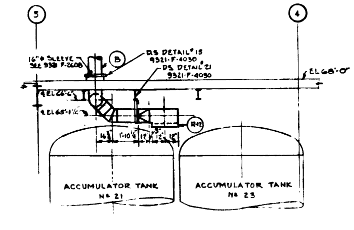
DWG. NO. 9321-F-2746-42 MY



PLAN ABOVE EL. 68'-0"
SCALE 1/8" = 1'-0"



PLAN BELOW EL. 68'-0"
SCALE 1/8" = 1'-0"



SECTION EE-EE
9321-F-4026
SCALE 1/4" = 1'-0"

14 THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN ENTERGY AS: "CLASS A " ITEMS PER THE Q.A.P.D.

14 FOR NOTES AND REF. DWGS. SEE 9321-2709.

NOTES:
1. FOR REFERENCE DRAWINGS AND NOTES SEE DWG. 9321-F-4023 & CON ED. DWG. A206838
2. PITCH UNIT HEATER CONDENSATE LINE 1" IN 40' TOWARD PENETRATION "T"
3. PITCH UNIT HEATER STEAM SUPPLY LINE 1" IN 40' TOWARD PENETRATION "S"
4. PROVIDE PIPE GUIDES AT COLUMNS FOR STEAM AND CONDENSATE LINES
5. ALL PIPES 2" AND SMALLER ARE SHOWN DIAGRAMMATICALLY AND SHALL BE RUN TO SUIT IN FIELD
6. SEE DETAIL 1 ON DWG. #311904
7. SEE DETAIL 2 ON DWG. #311904
8. ALL F.C.U.'S INCIDENT INLET/OUTLET F.C.V.'S & ASSOCIATED PANELS ARE RETIRED IN PLACE. 14

14	13	12
L. SMITH 5/27/03	V.I. 1/20/00	V.I. 4/17/99
THIS REVISION IS CLASS "A" PER THE QAPD. DRAWING REVISED TO INCORPORATE CHANGES IN FIELD PER MOD. FMX-95-11243-M. INCORPORATED DMD9321-4025-AB & DMD9321-4025-AC RELEASED AS BUILT P.N. 11243-95 SL/JS 5/27/03	THIS REVISION IS CLASS "A" PER CI-240-1 UPDATED DRAWING TO SHOW WORK DONE ON MOD. PROC. MFI-88-01408-M DMD9321-4025-AE PKG. R-812 RELEASED AS-BUILT P.N. 01408-88 THB/WD 01/20/00 CATARACT	THIS REVISION IS NON-CLASS PER CI-240-1 UPDATED DRAWING TO INCLUDE UFSAR FIGURE No. IN TITLE P/N 69911-NP RELEASED FOR RECORD DLA/4/17/99 STRATUS ENGINEERING

11 10/25/84
THIS REVISION IS CLASS "A" PER CI-240-1 UPDATED DWG TO SHOW WORK OF REV 10 & MOD PROCD MPC-84-15743-35. RELEASE AS CONSTRUCTED P/N 15743-35 DM/E.D. 10-15-84

10
THIS REVISION CONTAIN CLASS "A" ITEMS PER CI-240-1, MOD PROC MPC-84-15743.35 REVISED TAG NOS. FCV-23-8NC, FCV-23-9NC, FCV-23-7NO, FCV-24-12NC, FCV-24-10NO, FCV-25-15NC, FCV-25-13NO, FCV-25-14NC, FCV-22-6NC & FCV-22-4 NO TO READ FCV-23-7NC, FCV-23-8NC, FCV-23-9NO, FCV-24-10NC, FCV-24-12NO, FCV-25-13NC, FCV-14-NC, FCV-25-15NC, FCV-22-4NC & FCV-22-6NO, RESPECTIVELY RELEASE FOR CONST. PN 15743-35 WG/...

5-18-84 C

REV	DATE	DESCRIPTION	ENGR	SUP ENG	
8	11-3-77	THIS REV. CONTAINS CLASS "A" ITEMS PER CI-240-1 ADDED REF. DWG. FOR FCV-22-4N.C. RELEASE FOR RECORD ONLY MOD. PROCEDURE MMC-77-2-06 P.N. 6777			CON ED. REV.
7	11/1/73	FINAL ISSUE			
6		RELEASED FOR CONSTRUCTION			
5		DIMENSION 4'-5" WAS 3'-3" ADDED DIMENSIONS			
4		ADDED ... PLAN BELOW ...			
3	3-17-69	CHANGED DEMISTER LOCATION ADDED RED IN STEAM LINE			
2		ADDED STEAM HEATING PIPING & NOTES 2 TO 5			
1		ADDED VALVE NOS, REVISED DUCT			
		FIRST ISSUE ISSUED FOR BIDDING			

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STATE NEW YORK NO. 39...
UNITED ENGINEERS & CONSTRUCTORS INC.

MH

WESTINGHOUSE ELECTRIC CORPORATION

CONTAINMENT BUILDING
AIR RECIRCULATION SYSTEM
PLAN ABOVE 68'-0"
- UFSAR FIGURE No. 6.4-4

FOR

CONSOLIDATED EDISON COMPANY
INDIAN POINT GENERATING STATION
UNIT NO. 2

U.E. & C. DWG. NO.	CON. ED. CO. DWG. NO.
9321-F-4025-14	A-201049

CHAPTER 7
INSTRUMENTATION AND CONTROL

7.1 GENERAL DESIGN CRITERIA

Complete supervision of both the nuclear and turbine-generator sections of the plant is accomplished by the instrumentation and control systems from the control room. The instrumentation and control systems are designed to permit periodic online tests to demonstrate the operability of the reactor protection system.

Criteria applying in common to all instrumentation and control systems are given in the following listing. Thereafter, criteria that are specific to one of the instrumentation and control systems are discussed in the appropriate portion of the description of that system.

7.1.1 Instrumentation and Control Systems Criteria

Criterion: Instrumentation and controls shall be provided as required to monitor and maintain within prescribed operating ranges essential reactor facility operating variables. (GDC 12)

Instrumentation and controls essential to avoid undue risk to the health and safety of the public are provided to monitor and maintain neutron flux, primary coolant pressure, flow rate, temperature, and control rod positions within prescribed operating ranges.

Westinghouse designed and procured all systems that actuate reactor trip and safety feature actions for Indian Point Unit 2. The design of protective grade instrumentation and logic systems are in accordance with the proposed IEEE criteria for nuclear power plant protection system (IEEE-279 Code), dated August 28, 1968. The functional design is originated at Westinghouse with equipment procurement through vendor supplies. Equipment compatibility and integration of component hardware was factored into the design by Westinghouse or under the direct supervision of Westinghouse.

The non-nuclear regulating process and containment instrumentation measures temperatures, pressure, flow, and levels in the reactor coolant system, steam systems, containment, and other auxiliary systems. Process variables required on a continuous basis for the startup, power operation, and shutdown of the plant are controlled from and indicated or recorded in the control room. The quantity and types of process instrumentation provided ensure the safe and orderly operation of all systems and processes over the full operating range of the plant.

7.1.2 Related Criteria

The following criteria are related to all instrumentation and control systems but are more specific to other plant features or systems, and therefore are discussed in other sections, as listed.

Name	Discussion
Suppression of power oscillations (GDC 7)	Chapter 3
Reactor core design (GDC 6)	Chapter 3
Quality standards (GDC 1)	Chapter 4
Performance standards (GDC 2)	Chapter 4
Fire protection (GDC 3)	Chapters 5 and 9

Missile protection (GDC 40)	Chapters 4, 5, and 6
Emergency power (GDC 39)	Chapter 8

7.1.3 Environmental Qualifications - Original Plant Operations

As part of the original plant design for Indian Point Unit 2, environmental requirements were established for all safety-related electrical equipment in the facility. These requirements included environmental conditions, testing, and qualifications as discussed in this section. For a discussion of equipment requalification in accordance with recent NRC guidelines, see Section 7.1.4. Table 7.1-1 lists the equipment located within the primary containment (reactor containment building), which is required to be operable during or following a loss-of-coolant or a steam line break accident. In addition, Table 7.1-1 lists the equipment operational and environmental testing requirements as established as part of the original facility operations. Figures 7.1-1 and 7.1-2 present the environmental conditions of pressure and temperature, respectively, for both the Table 7.1-1 required equipment test conditions and for the containment post loss-of-coolant design accident (LOCA) conditions. Figures 7.1-3 and 7.1-4 present the maximum calculated instantaneous and integrated radiation dose levels inside the containment as a function of time following a TID-14844 model LOCA.

7.1.3.1 Category 1 – Instrumentation

Except for sump level channels LT-938, 939, and 941, the supplier completed preliminary qualification tests on pressure and differential pressure transmitters. These are reported in WCAP-7354-L,[2] which has been superceded by WCAP-7410-L.[3]

Additional instrumentation tests were performed by Westinghouse on equipment obtained from the Indian Point Unit 2 plant equipment supplier. The results of these tests confirmed that the equipment would provide the required signals in the post-LOCA environment.

The test conditions of the Westinghouse test were as follows:

steam environment, a 5-sec period rise to 286°F and 60-psig pressure and the maintenance of these conditions for 2 hr. All equipment, listed below, continued to operate throughout the test and are typical of transmitter ranges used in the containment.

Static Pressure Transmitters	Differential Pressure Transmitters
0-2500 psig	0-240-in. of water
1700-2500 psig	0-300 psid

Containment sump and recirculation sump level channels consist of hermetically sealed magnetic switches in a stainless steel housing. The instrumentation was designed for submerged service in borated water at 295°F at a pressure of 69 psig. Since instruments of this design have seen considerable actual service in applications more severe than the post-LOCA design conditions, environmental testing for these instruments was not required.

7.1.3.2 Category 2 – Valves

The Indian Point Unit 2 valve operator supplier conducted loss-of-coolant environmental tests on a motor operated valve with a Class H unit similar to those used in this plant. Reports of results indicated that the unit operated satisfactorily at test conditions more severe than those expected in the Indian Point Unit 2 loss-of-coolant or steam break accident environment.

In addition, Westinghouse performed environmental tests on a unit similar to that being used in the Indian Point Unit 2 plant. The results of the Westinghouse tests indicated that the equipment would perform its required function in the post-LOCA environment.

Tests performed on motor operated valve operators, both Class H and Class B, included the following:

1. Preliminary heat tests (dry heat for 16 hr at 375°F) on limit and torque switches. All parts operated freely.

2. Preliminary heat tests on actuator. A complete operator assembled and baked at 325°F for 12 hr. Unit operated every 0.5 hr for 2 min, full open to full close. All operations were satisfactory.

3. Preliminary live steam test. Live steam injected into switch compartment. Unit operated every 0.5 hr for 2 min over a period of 9 hr. All operations were satisfactory.

4. Heat aging of motor. Heat aging at 180°C for 100 hr (equivalent to 40-year life) was performed. Comparison of insulation resistance between new and aged motor indicated no significant insulation degradation.

5. Life cycle test. 150 life cycle test under loaded conditions (valve operator produced ~ 16,500 lb of thrust). No noticeable change in operator following test.

6. Environmental test. Valve operators subjected to environmental conditions shown in Figures 7.1-1 and 7.1-2 and sprayed continuously for a period of 3 hr with a solution of boric acid and sodium hydroxide.

Results of the tests are as follows:

1. Class H operator (actual peak test conditions 320°F at 90 psig): operator survived 1st day of exposure during which 12 complete reversing cycles were accomplished. Following 1-week exposure to 247°F and 14.7 psig, the unit was operated for two complete reversing cycles. The unit operated satisfactorily.

2. Class B operator: operator survived the 1st day of exposure with 12 complete reversing cycles, However, after 5 days of exposure, the operator failed (failure found to be a short in the motor winding).

As a result of the above tests, Class H operators were supplied where long-term operation is required. Class B operators were supplied where short-term (less than 12 hr) operation is required.

A production line valve motor was irradiated to a level of 2 x 10^8 rads using a cobalt-60 irradiation source. The irradiated motor and an identical unirradiated motor underwent a series of reversing tests at room temperature, followed by a series of reversing tests at 275°F. The room temperature test was repeated while vibrating the motors at a frequency of 30 cycles per sec. Both motors operated satisfactorily during all of the tests. No significant difference was evident in the comparison of the data for the two units throughout the test period.

7.1.3.3 [Deleted]

7.1.3.3.2 Hydrogen Recombiner System

The original flame recombiner system has been replaced with a Passive Autocatalytic Recombiner System (PARS). Qualification testing of the PARS is discussed in Section 6.8.4.1.

7.1.3.3.3 Cable and Splice Tests

Cabling of the type that is installed in the Indian Point Unit 2 plant was tested under simulated loss-of-coolant accident conditions. The tests were conducted by the cable manufacturer and Westinghouse and consisted of the following:

1. A test was performed by the cable manufacturer in a steam environment of 214°F for 436 hr. During this test, some cable was energized and was carrying current. A visual inspection following this test showed the cables to be in excellent condition. High voltage, tensile elongation, and stretch showed insignificant changes in their characteristics.

2. A test was performed by the cable manufacturer where the specimens were exposed to a gamma radiation field of 2.8 x 10^7 rads followed by exposure in a steam atmosphere of 85 psig for two 30-min cycles. Following these tests, the physical appearance of the cables was excellent. Changes in electrical characteristics were as follows:

    a. Insulation resistance - 90-percent of original value.
    b. Specific inductance (SIC) - No change.
    c. Dissipation factor - Change from 2.2 to 2.1-percent.
    d. AC breakdown - 82-percent of original value.

These percentages represent an average of seven samples.

Westinghouse performed cable testing in a postaccident steam and chemical environment of 80 psig (maximum) and a temperature in excess of 300°F. The durations of these tests were in excess of 200 hr in the postaccident steam and chemical environment, 68 hr of which was at a steam pressure higher than containment design pressure. The general appearance of the cables following these tests was good. Some loosening of the jacket from the insulation at the cable ends occurred, generally believed to be due to the rapid decrease in pressure during the tests. Had the cable ends been properly made up, this separation could have been prevented.

Westinghouse performed additional testing on 18 cable and cable splice test specimens. The testing consisted of the following:

1. Thermal aging to an equivalent of 40 years of operation. (Kerite cable - 150°C for 192 hr; silicone cable - 210°C for 30 days.)

2. Irradiation to levels up to 2 x 10^8 rads.

3. Exposing the cables for three weeks to the environmental test conditions shown in Figures 7.1-1 and 7.1-2.

4. Applying a potential of 480-V (with respect to ground) to the cables and conducting rated current through the cables on a daily schedule of 8 hr on and 16 hr off.

Before the admission of steam to the test chamber, it was found that four of the test specimens had open conductors. These four specimens were removed from the chamber, and subsequent examination indicated that the conductors had not been crimped properly.

Of the 14 specimens tested in the environment, 13 survived. The 14th was found to have shorted against the test grounding pipe surrounding the cable. The short appeared to have occurred because of the whipping of the cable caused by the steam injection against the cable.

According to the above test results, the safeguards cable and splices used on the Indian Point Unit 2 plant will maintain their required integrity under post-LOCA conditions.

7.1.3.4 Postaccident Equipment Radiation Exposure

The design basis for the reactor protection system and engineered safety feature equipment radiation exposure is that the equipment must function after exposure associated with the TID-14844 model accident. The maximum anticipated exposure for components located within the containment was originally specified as 1.6 x 10^8 rads, which is accumulated during 1 year following the accident (note that the integrated exposure for safeguards equipment during 40 years of operation is less than 5 x 10^5 rads). In the determination of exposure no credit was taken for containment cleanup or other removal mechanism other than isotope decay. The expected integrated exposure on the outside of the containment building, again assuming TID-14844 releases and no credit for cleanup, was originally specified to be less than 10^2 rads integrated over a year at the containment outside surface. These radiation exposure values are updated and are maintained by the ongoing Environmental Qualification Program described in Section 7.1.4.

To establish the combined effect of long-term operation followed by exposure to accident conditions inside the containment, selected components were subjected to thermal aging followed by irradiation. In addition, components were first irradiated and then subjected to thermal aging. Results of the tests indicated that the components would perform satisfactorily following a design-basis accident.

Indian Point Unit 2 cables were tested using the same approach as described above, that is, first irradiation, and then thermal aging followed by steam exposure. During exposure to steam the cables carried nominal voltage and current.

7.1.4 Environmental Qualifications

An ongoing program of evaluating the environmental qualification of safety-related electrical equipment at the Indian Point Unit 2 facility has been in progress since early 1980. On May 23, 1980, the NRC Commissioners issued Memorandum and Order CLI-80-21, which describes the NRC environmental qualification requirements. CLI-80-21 states that the Division of Operating Reactors guidelines and NUREG-0588 set the requirements that licensees and applicants must meet regarding the environmental qualification of safety-related electrical equipment to satisfy 10 CFR 50, Appendix A, GDC 4.

In February 1980, the NRC included Indian Point Unit 2 in the systematic evaluation program for the purpose of the equipment environmental qualification review.

On March 5, 1980, Con Edison was formerly asked to address the environmental qualification of safety-related electrical equipment for Indian Point Unit 2. This evaluation information was detailed in several responses. The original response was transmitted to the NRC by Con Edison on May 9, 1980,[7] with additional information in subsequent transmittals.[8-10] This information was evaluated and a safety evaluation report (SER) was issued by the NRC on May 21, 1981. The Con Edison response to this SER was made on September 4, 1981.[11] As a result of this SER, containment accident pressure was changed to 40.6 psig with a saturation temperature of 287°F, and this information was recorded as a part of the review for the September 4, 1981 response[11] to the NRC Staff SER. The assessment of additional electrical equipment was provided to the NRC Staff by Con Edison submittal dated May 4, 1982.[12]

Included in this program were evaluations of the following variables and environmental parameters: function, service, location, operating time, temperature, pressure, relative humidity, chemical spray, radiation, aging, submergence, and qualifying method. This evaluation program was based on the provisions of the Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors (DOR Guidelines), or NUREG-0588, Interim Staff Position on Environmental Qualification of Safety Related Electrical Equipment, December 1979.

The information submitted by Con Edison was evaluated by the Franklin Research Center and a technical evaluation report (TER) was issued in June 1982. A safety evaluation report was subsequently issued in January 1983 with the TER as an attachment. Con Edison responded to the safety evaluation report items and also presented the general methodology for compliance with 10 CFR 50.49, Environmental Qualification of Electrical Equipment, which had become effective in February 1983, in a submittal to the NRC in October 1984.[13] In accordance with schedule requirements for environmental qualification contained in 10 CFR 50.49, all components falling within the scope of this program were to be qualified, replaced, or modified to ensure their operation. The submittal was evaluated by the NRC and a final resolution was issued in December 1984.[14] The NRC evaluation concluded the following:

1. Con Edison's electrical equipment environmental qualification program complies with the requirements of 10 CFR 50.49.

2. The proposed resolution for each of the environmental qualification items identified in the safety evaluation report of January 1983 is acceptable.

3. Continued operation until completion of the licensee's environmental qualification program will not present undue risk to the public health and safety.

A comprehensive list of all electrical equipment important to safety, pursuant to the environmental qualification rule, 10 CFR 50.49, has been submitted to the Nuclear Regulatory Commission by Reference 1. This list called the "EQ Master List" is periodically updated to reflect plant modifications, procedure changes, or new analysis.

Further EQ evaluations were performed to support plant operation at an intermediate core thermal power level of 3071.4 MW (3083.3 MW NSSS) and a 1.4% power uprate to 3114.4 MW and were reevaluated for steam generator replacement and stretch power level of 3216 MWt.

LOCA containment re-analysis performed as part of IP2 stretch power uprate (3216 MWt) conditions resulted in a peak containment pressure of 45.71 psig and a peak temperature of 266.81^{o}F (UFSAR Section 14.3.5.1.1). For EQ purposes, maximum pressure and temperature of 45.81 psig and 266.97 ^{o}F are used.

Post-accident chemistry changes due to the elimination of the spray additive tank and installation of Trisodium Phosphate Baskets were evaluated and it was determined that these changes did not affect the environmental qualification of equipment located within the containment required to mitigate the consequences of design basis accidents.

For the purpose of responding to Generic Letter 2004-02 (Generic Safety Issue 191), the Trisodium Phosphate pH buffer was replaced by Sodium Tetraborate. The new buffer material was evaluated for its effect on equipment located in the containment required to mitigate the consequences of design basis accidents (Reference 31). The evaluation concluded that due to the similarities between post-LOCA Trisodium Phosphate and Sodium Tetraborate buffered sump solutions; the equipment qualified for Trisodium Phosphate remains qualified for the new Sodium Tetraborate buffered solution. Therefore, there would be no impact on existing IP2 EQ equipment as a result of the subject post-LOCA buffered sump chemistry change.

Complete and auditable records are available and will be maintained at a central location. These records describe the environmental qualification methods used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines, NUREG-0588, and/or 10 CFR 50.49. Such records were updated and maintained current as equipment was replaced, further tested, or otherwise further qualified.

7.1.5 Regulatory Guide 1.97 Compliance

Compliance of the instrumentation at Indian Point Unit 2 with the intent of Regulatory Guide 1.97, Revision 2, as required by NUREG-0737, Supplement 1 (Generic Letter 82-33), has been addressed in submittals to the Nuclear Regulatory Commission, dated August 30, 1985[15], September 12, 1986[16], October 26, 1988[17], October 27, 1989[18], August 7, 1991[19,] November 19 1992[20], November 26,1986[22], November 15, 1994[21], April 7, 1995[23], and September 18, 1995[24]. The submittals included the degree of compliance and intended upgrades or justifications where deviations of present instrumentation were identified. The NRC determined that the plant design is acceptable with respect to conformance to Regulatory Guide 1.97 Rev 2 In SERs dated September 27, 1990[25], August 31, 1992[26], August 27, 1993[27], February 2, 1995[28], and November 27, 1995[29]. Control room indicators and recorders for instrumentation designated as Types A, B, and C and Categories 1 and 2 of Regulatory Guide 1.97 are specifically identified on the control room panels.

REFERENCES FOR SECTION 7.1

1. Letter from J. D. O'Toole, Con Edison, to S. A. Varga, NRC, Subject: Environmental Qualification Rule, dated May 20, 1983.

2. J. Nay, Topical Report Supplier Post Accident Testing of Process Instrumentation, WCAP-7354-L (Proprietary), Westinghouse Electric Corporation, July 1969.

3. J. Locante, Topical Report Environmental Testing of Engineered Safety Features Related Equipment (NSSS-Standard Scope), WCAP-7410-L (Volumes I and II, Proprietary), Westinghouse Electric Corporation, December 1970

4. C. V. Fields, Fan Cooler Motor Unit Development and Test, WCAP-9003 (Proprietary), Westinghouse Electric Corporation, January 1969.

5. Westinghouse Electric Corporation, Reactor Containment Fan Cooler Motor Insulation Irradiation Testing, WCAP-7343-L (Proprietary), WCAP-7829 (Nonproprietary), July 1969.

6. Schulz Electric Report No. N4446EQFWCD, “Environmental Qualification Report Number N4446EQFWCD for Schulz Electric Company’s Form Wound, Continuous Duty Insulation System.”

7. Schulz Electric Report No. 45925-1, “Schulz Electric Company’s Environmentally Qualified Insulation System Supplement 1.”

8. Letter from Con Edison response to NRC, Subject: Response to NRC request of March 5, 1980, on Environmental Qualification Program, dated May 9, 1980.

9. Letter from Con Edison to NRC, Subject: Additional Material on Environmental Qualification Program, dated October 31, 1980.

10. Letter from Con Edison to NRC, Subject: Additional Material on Environmental Qualification Program, dated January 14, 1981.

11. Letter from Con Edison to NRC, Subject: Additional Material on Environmental Qualification Program, dated April 13, 1981.

12. Letter from Con Edison, to NRC, Subject: Response to NRC Safety Evaluation Report issued by NRC on May 21, 1981, dated September 4, 1981.

13. Letter from Con Edison, to NRC, Subject: Additional Information on Environmental Qualification of Safety Related Electrical Equipment, dated May 4, 1982.

14. Letter from J. D. O'Toole, Con Edison, to S. A. Varga, NRC, Subject: Environmental Qualification of Safety Related Electrical Equipment, dated October 5, 1984.

15. Letter from S. A. Varga, NRC, to J. D. O'Toole, Con Edison, Subject: Final Resolution of Environmental Qualification of Electric Equipment Important to Safety, dated December 7, 1984.

16. Letter from J.D. O'Toole, Con Edison, to H.L. Thompson, Jr., NRC, Subject: Compliance with Guidance of Regulatory Guide 1.97, Revision 2, dated August 30, 1985.

17. Letter from M. Selman, Con Edison, to H. L. Thompson, Jr., NRC, dated September 12, 1986, with attachment "Response to Preliminary Technical Evaluation Report."

18. Letter from John Basile, Con Edison, to Document Control Desk, NRC, Subject: Additional Information Regarding NUREG-0737, Supplement 1 (Regulatory Guide 1.97 Revision 2), dated October 26, 1988.

19. Letter from Stephen Bram, Con Edison, to Document Control Desk, NRC, Subject: Clarification of Information Regarding NUREG-0737 Supplement 1 (Regulatory Guide 1.97 Revision 2) dated October 27, 1989.

20. Letter from Stephen Bram, Con Edison, to Document Control Desk, NRC, Subject: Supplemental Information Regarding NUREG-0737, Supplement 1 (Regulatory Guide 1.97, Revision 2) dated August 7, 1991.

21. Letter from Stephen Bram, Con Edison, to Document Control Desk, NRC, Subject: Supplemental Information Regarding NUREG-0737, Supplement 1 (Regulatory Guide 1.97, Revision 2) dated November 19, 1992.

22. Letter from Stephen Quinn, Con Edison, to Document Control Desk, NRC, Subject: Steam Generator Wide Range Level Indication Upgrade (Regulatory Guide 1.97, Revision 2) dated November 15, 1994

23. Letter from J. Basile, Con Edison, to M. Slosson, NRC, Subject: Indian Point Unit No. 2, Docket No. 50-247, dated November 26, 1986.

24. Letter from Stephen E. Quinn, Con Edison, to Document Control Desk, NRC, Subject: Response to Request for Additional Information Regarding Neutron Flux Instrumentation (TAC No. M81727) dated April 7, 1995.

25. Letter from Stephen E. Quinn, Con Edison, to Document Control Desk, NRC, Subject: Supplemental Submittal Regulatory Guide 1.97 Qualification of Neutron Flux Instrumentation (TAC No. M81727) dated September 18, 1995.

26. Letter from Donald S. Brinkman, NRC, to Stephen B. Bram, Con Edison, Subject: Conformance to Regulatory Guide 1.97, Revision 2 (TAC No. M51098) dated September 27, 1990.

27. Letter from F. J. Williams, Jr., NRC, Subject: Regulatory Guide 1.97 - Instrumentation to Follow the Course of an Accident, Indian Point Nuclear Generating Unit No. 2 (TAC No. M51098) dated August 31, 1992.

28. Letter from F. J. Williams, NRC, to Stephen B. Bram, Con Edison Subject: Supplemental Safety Evaluation - Regulatory Guide 1.97 Instrumentation to Follow the Course of an Accident, Indian Point Nuclear Generating Unit NO.2 (TAC No. M81727) dated August 27, 1993.

29. Letter from F. J. Williams, Jr., to Stephen E. Quinn, Con Edison, NRC, Subject: Steam Generator Wide Range Level Indication Upgrade, Indian Point Nuclear Generating Unit No. 2 (TAC No. M81727) dated February 2, 1995.

30. Letter from F. J. Williams, Jr., NRC, to Stephen E. Quinn, Con Edison, Subject: Conformance to Regulatory Guide 1.97, Revision 2, Post-Accident Neutron Flux Monitoring Instrumentation for Indian Point Nuclear Generating Unit NO. 2 (TAC No. M81727), dated November 27, 1995.

31. "Evaluation of IP2 and IP3 Post-LOCA Buffered Borate Sump Chemistry for Equipment Qualification," IP-RPT-08-00025, Revision 0.

TABLE 7.1-1
Postaccident Equipment (Inside Containment)
Operational and Testing Requirements

Equipment Name and Tag Number	Operating Mode	Environmental Testing
CATEGORY 1 - INSTRUMENTATION		
Pressurizer pressure channels: PT-455, 456, 457,474	Continuous	Required
Pressurizer level channels: LT-460, 459, 461	Continuous	Required
High-head flow channels: FT-924, 925, 926, 927	Continuous	Required
Steam generator level channels: LT-417A-C, 427A-C,437A-C, 447A-C	Continuous	Required
Recirculation spray flow channels: FT-945A, B	Intermittent	Required
Recirculation sump level channel: LT-3301	Continuous	Required
Containment sump level channel: LT-3300	Continuous	Required
Residual heat loop flow channels: FT-640, 946A, B, C, D	Continuous	Required

Equipment Name and Tag Number	Operating Mode	Environmental Testing

TABLE 7.1-1 (Cont.)
Postaccident Equipment (Inside Containment)
Operational and Testing Requirements

CATEGORY 2 - VALVES

Safety injection line valves:		
MOV-856A, C, D, E	Open or close on demand$_1$	Required
MOV-856B, F	Open or close on demand$_2$	Required
Recirculation spray valves: MOV-889A, B	Open or close on demand	Required
Recirculation pump discharge valves: MOV-1802A, B	Open after injection phase, close on demand	Required
Containment sump isolation valve: MOV-1805	Open or close on demand	Required
Residual heat exchanger cooling water supply valves: MOV-822A, B	Open on demand$_1$	Required
Residual heat exchanger isolation valves: MOV-745A$_4$, B$_4$, 746$_1$, 747$_1$	Open or close on demand	Required
Residual heat loop flow control valves: HCV-638, 640	Open or close on demand	Required
Air-operated isolation valves	Close on demand	Not required$_3$

CATEGORY 3 - MISCELLANEOUS ITEMS

Fan cooler motors: 21, 22, 23, 24, 25	Continuous	Required
Internal recirculation pump motors	Start after injection phase and continue operating	Required
Hydrogen recombiner system	Operate on demand	Required
Safeguard equipment power, control and instrument cable	Continuous	Required

Notes:
1. Also open on SI signal.
2. Deenergized closed per Technical Specifications.
3. All air-operated valves fail in closed position.
4. Deenergized open per Technical Specifications.

TABLE 7.1-2
DELETED

TABLE 7.1-3
DELETED

TABLE 7.1-4
DELETED

TABLE 7.1-5
DELETED

7.1 FIGURES

Figure No.	**Title**
Figure 7.1-1	ENVIRONMENTAL CONDITIONS FOR EQUIPMENT TESTING - PRESSURE VS TIME
Figure 7.1-2	ENVIRONMENTAL CONDITIONS FOR EQUIPMENT TEMPERATURE VS TIME
Figure 7.1-3	INSTANTANEOUS GAMMA DOSE RATE INSIDE THE CONTAINMENT AS A FUNCTION OF TIME AFTER RELEASE - TID - 14844 MODEL
Figure 7.1-4	INTEGRATED GAMMA DOSE LEVEL INSIDE THE CONTAINMENT AS A FUNCTION OF TIME AFTER RELEASE - TID - 14844 MODEL
Figure 7.1-5	Deleted
Figure 7.1-6	Deleted
Figure 7.1-7	Deleted
Figure 7.1-8	Deleted

7.2 PROTECTION SYSTEMS

The protection systems consist of both the reactor protection system and the engineered safety features. Equipment supplying signals to any of these protection systems is considered a part of that protection system.

7.2.1 Design Bases

7.2.1.1 Control Room

Criterion: The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit continuous occupancy of the control room under any credible post-accident condition or as an alternative, access to other areas of the facility as necessary to shut down and maintain safe control of the facility without excessive radiation exposures of personnel. (GDC 11)

The plant is equipped with a control room that contains those controls and instrumentation necessary for the operation of the reactor and turbine generator under normal and accident conditions.

The control room is continuously occupied by the qualified operating personnel under all operating and maximum credible accident conditions.

As part of the initial plant design, sufficient shielding, distance, and containment integrity were provided to ensure that control room personnel would not be subjected to doses under postulated accident conditions during occupancy of, ingress to, and egress from the control room, which in the aggregate, would exceed limits in 10 CFR 100. The control room ventilation consists of a system having a large percentage of recirculated air. The fresh air intake is automatically diverted to charcoal filters to remove airborne activity if monitors indicate that such action is appropriate.

An earlier study was performed to evaluate the system and structures pertaining to the habitability of the control room in accordance with NUREG-0737, TMI Action Plan Requirements, item III.D.3.4. This TMI action item was to ensure that control room operators will be adequately protected against the effects of accidental releases of toxic and radioactive gases and that the plant can be safely operated or shut down under design-basis accident conditions. The results of an evaluation for specifically measured central control room conditions[1] indicated that the radiation doses in the control room integrated over 30 days post-LOCA were below the dose guidelines. Additional radiation detectors were installed in the central control room outside air intake. As a result of an analysis of the toxic chemical data within a 5-mile radius of Indian Point, redundant toxic chemical monitors were also installed. High concentrations of the specific toxic gases of concern result in an alarm and automatic isolation of the control room.

Two independent toxic gas detection systems, each capable of detecting chlorine and anhydrous ammonia, shall be operable at all times except as specified in the Unit 2 Technical Requirements Manual (TRM). ***[Note:]*** This was moved out of Technical Specifications by NRC SER for TS Amendment 208. The requirements of the Tech Spec per the NRC Order were relocated to the UFSAR and in Rev. 21 to the UFSAR were relocated to the TRM and need to remain as a License Condition.

Smoke detection capability has also been provided at the central control room outside air intake. The detection of smoke results in an alarm and automatic isolation of the control room. For further information, see NRC Safety Evaluation Report (SER) dated January 27, 1982 (Reference 6).

More recently the application of the NUREG-1465 alternative source term methodology for Indian Point Unit 2 includes verification that the radiological dose to control room personnel following postulated accidents remains within the limits specified in 10 CFR 50.67 as presented in Section 14.3.6.5.

7.2.1.2 Reactor Protection System

Criterion: Core protection systems, together with associated equipment, shall be designed to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits. (GDC 14)

The basic reactor tripping philosophy is to define a region of power and coolant temperature conditions allowed by the primary tripping functions, the overpower high ΔT trip, the overtemperature high ΔT trip, and the nuclear overpower trip. The allowable operating region within these trip settings is provided to prevent any combination of power, temperature, and pressure, which would result in departure from nucleate boiling with all reactor coolant pumps in operation. Additional tripping functions such as a high pressurizer pressure trip, low pressurizer

pressure trip, high pressurizer water level trip, loss of flow trip, steam and feedwater flow mismatch trip, steam generator low-low water level trip, turbine trip, safety injection trip, nuclear source and intermediate range trips, fuel cooldown trip, and manual trip are provided as backup to the primary tripping functions for specific accident conditions and mechanical failures.

A dropped rod signal provides a turbine load runback if above a given power level. The dropped rod is indicated from individual rod position indicators or by a rapid neutron flux decrease on any of the power range nuclear channels (as discussed in Section 14.1.4).

Intermediate Range and Power Range rod stops, Overtemperature ΔT and Overpower ΔT rod stops, are provided to prevent abnormal power conditions which could result from excessive control rod withdrawal initiated by operator violation of administrative procedures.
The core protection system is shown schematically on the plant logic diagrams, Plant Drawings 225094 through 225107 [Formerly UFSAR Figures 7.2-1 through 7.2-14].

7.2.1.3 Engineered Safety Features Protection System

Criterion: Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features. (GDC 15)

Instrumentation and controls provided for the protection systems are designed to trip the reactor, to prevent or limit fission product release from the core and to limit energy release, to signal containment isolation, and to control the operation of engineered safety features equipment.

The engineered safety features systems are actuated by the engineered safety features actuation channels. Each coincidence network energizes an engineered safety features actuation device that operates the associated engineered safety features equipment, motor starters, and valve operators. The channels are designed to combine redundant sensors and independent channel circuitry, coincident trip logic, and different parameter measurements so that a safe and reliable system is provided in which a single failure will not defeat the channel function. The action initiating sensors, bistables, and logic are shown in the figures included in the detailed engineered safety features instrumentation description given in the system design section. The engineered safety features instrumentation system actuates (depending on the severity of the condition) the safety injection system, the containment isolation system, the containment air recirculation system, and the containment spray system.

The passive accumulators of the safety injection system do not require signal or power sources to perform their function. The actuation of the active portion of the safety injection system is described in Section 7.2.3.

The containment air recirculation coolers are normally in use during plant operation. These units are, however, in the automatic sequence, which actuates the engineered safety features upon receiving the necessary actuating signals indicating an accident condition. The fan cooler bypass valves open on safety injection to provide maximum service water flow.

Containment spray is actuated by coincident and redundant high containment pressure signals.

The containment isolation system provides the means of isolating the various pipes passing through the containment walls as required to prevent the release of radioactivity to the outside environment in the event of a loss-of-coolant accident.

The engineered safety features protection systems are shown schematically on the plant logic diagrams, Plant Drawings 225094 through 225107 [Formerly UFSAR Figures 7.2-1 through 7.2-14].

7.2.1.4 Protection Systems Reliability

Criterion: Protection systems shall be designed for high functional reliability and in-service testability necessary to avoid undue risk to the health and safety of the public. (GDC 19)

The reactor uses the Westinghouse magnetic-type control rod drive mechanisms. Upon a loss of power to the coils, the rod cluster control (RCC) assemblies with full-length absorber rods are released and fall by gravity into the core. The undervoltage trip coils and the shunt trip coils of the reactor trip breakers are used as the primary and backup devices, respectively, for the automatically initiated reactor trip signals.

The reactor internals, fuel assemblies, RCC assemblies, and drive system components are designed as seismic Class I equipment. The RCC assemblies are fully guided through the fuel assembly and for the maximum travel of the control rod into the guide tube. Furthermore, the RCC assemblies are never fully withdrawn from their guide thimbles in the fuel assembly. Because of this and the flexibility designed into the RCC assemblies, abnormal loadings and misalignments can be sustained without impairing operation of the RCC assemblies.

The RCC assembly guide system is locked together with pins throughout its length to ensure against misalignments that might impair control rod movement under normal operating conditions and credible accident conditions. An analogous system has successfully undergone 4132 hr of testing in the Westinghouse reactor evaluation channel, during which about 27,200-ft of step-driven travel and 1461 trips were accomplished with test misalignments in excess of the maximum possible misalignment that may be experienced when installed in the plant.

All reactor trip protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power.

Reliability and independence are obtained by redundancy within each tripping function. In a two-out-of-three circuit, for example, the three channels are equipped with separate primary sensors. Each channel is continuously fed from its own independent electrical source. Failure to deenergize a channel when required would be a mode of malfunction that would affect only that channel. The trip signal furnished by the two remaining channels would be unimpaired in this event.

7.2.1.5 Protection Systems Redundancy and Independence

Criterion: Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of such a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection function to be served. (GDC 20)

The reactor protection systems are designed so that the most probable modes of failure in each protection channel result in a signal calling for the protective trip. Each protection system design combines redundant sensors and channel independence with coincident trip philosophy so that a safe and reliable system is provided in which a single failure will not defeat the channel function, cause a spurious plant trip, or violate reactor protection criteria.

Channel independence is carried throughout the system extending from the sensor to the relay actuating the protective function. The protective and control functions when combined are combined only at the sensor. Both of these functions are fully isolated in the remaining part of the channel, control being derived from the primary protection signal path through an isolation amplifier. As such, a failure in the control circuitry does not affect the protection channel. This approach is used for pressurizer pressure and water level channels, steam generator water level, T_{avg} and delta T channels, steam flow-feedwater flow, and nuclear source and power range channels.

A nonelectrical backup to the pressurizer pressure and level indication exists that uses pneumatic transmitters with indicators located inside and outside the containment.

The transmitters are supplied with air from the instrument air headers inside the containment with a backup supply from the nitrogen system also located inside the containment.

The same nonelectrical backup is used to provide indication of steam-generator levels inside and outside the containment.

The engineered safety features equipment is actuated by either one or both of the engineered safety features actuation channels. Each coincidence network actuates an engineered safety actuation device that operates the associated engineered safety features equipment, motor starters, and valve operators. As an example, the control circuit of a safety injection pump is typical of the control circuit for a large pump operated from switchgear. The actuation relay, energized by the engineered safety features instrumentation system, has normally open contacts. These contacts energize the circuit breaker closing coil to start the pump when the control relay is energized.

In the reactor protection system, two reactor trip breakers are provided to interrupt power to the full-length rod drive mechanisms. The breakers main contacts are connected in series (with power supply) so that opening either breaker interrupts power to all full-length rod mechanisms, permitting them to fall by gravity into the core. In the event of a loss of rod control power, the reactor trip breakers are deenergized and trip to an open mode.

Further information on redundancy is provided through the detailed descriptions of the respective systems covered by the various sections in this chapter. In summary, reactor protection is designed to meet all presently defined reactor protection criteria and is in accordance with the proposed IEEE criteria for nuclear power plant protection system (IEEE-279 Code), dated August 28, 1968. Redundancy and independence are more than achieved by protection channel designs, which combine more than one sensor and Parameter measurement with coincident trip circuitry (e.g., pressure coincident with level and interlocked with flow or nuclear flux).

Required continuous electrical supply is discussed in Chapter 8.

7.2.1.6 Protection Against Multiple Disability for Protection Systems

Criterion: The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident, shall not result in loss of the protection function or shall be tolerable on some other basis. (GDC 23)

The components of the protection system are designed and laid out so that the mechanical and thermal environments accompanying any emergency situation in which the components are required to function do not interfere with that function.

The separation of redundant analog protection channels originates at the process sensors and continues back through the field wiring and containment penetrations to the analog protection racks. Physical separation is used to the maximum practical extent to achieve the separation of redundant transmitters. The separation of field wiring is achieved using separate wireways, cable trays, conduit runs, and containment penetrations for each redundant channel. Redundant analog equipment is separated by locating redundant components in different protection racks. Each channel is energized from a separate vital instrument bus.

7.2.1.7 Demonstration of Functional Operability of Protection Systems

Criterion: Means shall be included for suitable testing of the active components of protection systems while the reactor is in operation to determine if failure or loss of redundancy has occurred. (GDC 25)

The signal conditioning equipment of each protection channel in service at power is capable of being tested and tripped independently by simulated analog input signals to verify its operation. This includes checking through to the trip breakers, which necessarily involves the trip logic. Thus, the operability of each trip channel can be determined conveniently and without ambiguity.

The testing of the diesel-generator starting is performed from the diesel-generator control board. The generator breaker is not closed automatically after starting during this testing. The generator may be manually synchronized to the 480-V bus for loading. Complete testing of the starting of diesel generators can be accomplished by tripping the associated 480-V undervoltage relays. The ability of the units to start within the prescribed time and to carry load is periodically checked. (The electrical system is discussed in more detail in Section 8.2.3.)

7.2.1.8 Protection Systems Failure Analysis Design

Criterion: The protection systems shall be designed to fail into a safe state or into a state established as tolerable on a defined basis if conditions such as disconnection of the system, loss of energy (e.g., electrical power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced. (GDC 26)

Each reactor trip circuit is designed so that circuit trip occurs when the circuit is deenergized; therefore, loss of channel power causes the system to go into its trip mode. In a two-out-of-three circuit, the three channels are equipped with separate primary sensors and each channel is energized from an independent electrical bus. Failure to deenergize when required is a mode of