June 2011

This version of the Indian Point Unit 3 Updated Final Safety Analysis Report (UFSAR) is the licensee's version submitted to the NRC on October 13, 2009, with information current through April 2009, 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:
Figure 1.2-3, which was supplied as plant drawing 9321-F-64513.
Figure 2.2-3, supplied as drawing 9321-F-64513.
Figure 7.7-1, 7.7-2, supplied as drawings 9321-F-30523, 33833.
Figures 11.2-1, 11.2-2A, 11.2-2B, 11.2-3, 11.2-4, supplied as drawings 9321-F-25083, 25023, 25033, 25153, 25243.

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 3 UFSAR for Revision 03

Current Changes - Revision 03:
The 2009 revision to the Unit 3 UFSAR is Revision 03. 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 the 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 IP3:
Information contained in the Unit 3 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.

UFSAR Figures and Plant Drawings:
In an effort to improve the accuracy of the figures in the UFSAR and to reduce redundant work, Revision 01 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 Indian Point 3 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 3
UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR)

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CHAPTER 1

INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

The Final Safety Analysis Report for Indian Point 3 (as supplemented and amended – Supplements 1 through 32) was submitted by Consolidated Edison Company of New York (Con Ed) as part of an application for a license to operate this facility (Docket No. 50-286, Construction Permit #CPPR-62). This report provided pertinent technical information in accordance with the requirements of Section 50.34 of 10 CFR Part 50, in effect at the time of the application for operating license. Facility Operating License No. DPR-64 which authorized fuel loading and subcritical testing of Indian Point 3 was issued to Con Ed on December 12, 1975.

Westinghouse Electric Corp. was the primary contractor and had turnkey responsibility for the design, construction, testing and initial start-up of the facility. Westinghouse contracted the services of United Engineers and Constructors Inc. as architect-engineer to provide engineering assistance in the design and construction of the plant.

After completion of construction, preoperational tests, and initial start-up, Con Ed assumed responsibility for the operation of the plant on a commercial basis.

Concurrent with NRC review of the Con Ed Application for Operating License, Con Ed and the Power Authority of the State of New York (the Authority) filed an application on April 25, 1975 which requested that any operating license issued to Con Ed for operation of Indian Point 3 be amended to make the Authority a co-holder of the license. The amendment sought to permit the Authority to purchase and acquire title to Indian Point 3. However, Con Ed would retain complete responsibility for the operation of the facility in a safe manner and in accordance with Nuclear Regulatory Commission (NRC) requirements. In response to this application, NRC issued, on December 24, 1975, Amendment No. 1 to Facility Operating License No. DPR-64, which authorized the Authority to own but not operate Indian Point 3. On December 31, 1975, the Authority purchased the facility from Con Ed, while Con Ed retained complete responsibility for operation of Indian Point 3 under contract with the Authority. Services contracted for included: operation (including quality assurance), engineering, maintenance and training services, health physics, water chemistry, environmental monitoring, plant and site security and construction management.

On March 11, 1977, the Authority and Con Ed filed an application (supplemented by letters dated August 9, 1977, October 27, 1977 and December 14 and 20, 1977) which sought to provide the Authority with the sole responsibility for operation of Indian Point 3. Amendment No. 12 to Facility Operating License DPR-64, issued by NRC on March 8, 1978 and effective at 12:01 A.M., March 10, 1978, licensed the Authority to own, use and operate the facility. Since the date of effectiveness of this amendment, the Authority has had the sole responsibility for operation of Indian Point 3.

On June 11, 1990, the Authority filed an application (supplemented by letters dated June 18, 1991, February 11, 1992, and May 13, 1992) to extend the expiration date of the Indian Point 3 Operating License, from August 13, 2009 to December 12, 2015. Amendment No.

124 to Facility Operating License (FOL) DPR-64, issued by the NRC on July 15, 1992, extended the expiration date of the FOL to midnight, December 12, 2015.

The Authority and Entergy Nuclear Operations, Inc., filed an application under cover letters dated May 11 and May 12, 2000, as supplemented on June 13 & 16, July 14, September 21, October 26, and November 3, 2000, which sought to provide Entergy Nuclear Incorporated with the sole responsibility for operation of Indian Point 3. Amendment No. 203 to Facility Operating License DPR-64, issued by NRC and effective on November 21, 2000, licensed Entergy to own, use and operate the facility. Since the date of effectiveness of this amendment, Entergy has had the sole responsibility for operation of Indian Point 3.

In accordance with the requirements of Section 50.71 (e) of 10 CFR Part 50, Entergy Nuclear Northeast, as licensee for the Indian Point 3 facility, submits this updated Final Safety Analysis Report. The information presented herein describes the as-built condition of the facility and reflects the current operating conditions of Indian Point 3, effective April 15, 2009.

In accordance with NEI 98-03, Rev. 1, *Guidelines for Updating Final Safety Analysis Reports*, which is endorsed by Regulatory Guide 1.181, *Contents of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)*, dated September 1999, and as stated in the NRC Safety Evaluation for License Amendment No. 205, dated February 27, 2001, the Technical Requirements Manual (TRM) for Unit 3 is incorporated by reference into the UFSAR.

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

The facility employs a pressurized water reactor (PWR) and nuclear steam supply system (NSSS) which was furnished by Westinghouse Electric Corporation. It is similar in design to several plants now in operation as discussed in Section 1.4.

As prescribed in Amendment No. 17 to Facility Operating License DPR-64, issued by NRC on August 18, 1978, the Authority was authorized to operate the facility at reactor core power levels not in excess of 3025 megawatts thermal (100 % of rated power). This thermal power level corresponds to a turbine – generator output of 965 MWe net.

By Amendment 225 to Facility Operating License DPR-64, issued by the NRC on March 24, 2005, Entergy is authorized to operate the facility at reactor core power levels not in excess of 3216 megawatts thermal (100% of rated power). This thermal power level corresponds to a turbine-generator output of approximately 1080 MWe.

The plant heat removal systems were designed for the turbine guaranteed rating of 3083 MWt and the portions of the safety analysis dependent on heat removal capacity of plant and safeguards systems assumed the higher, maximum calculated power of 3216 MWt, as did the evaluations of activity release and radiation exposure for the unlikely event of a design basis accident (DBA).

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 Reactor Coolant System; Chapter 5, the Containment System; Chapter 6, the Engineered Safety Features; Chapter 7, Plant Instrumentation and Control; Chapter 8, the Electrical Systems; Chapter 9, the Auxiliary and Emergency Systems; Chapter 10, the Steam and Power Conversion System; Chapter 11, Radioactive Wastes and Radiation Protection. Chapter 12 and 13 are Conduct of Operations and Initial Tests and Operation, respectively. They describe plant organization, training programs, and start-up administrative procedures. 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 show that the consequences of various postulated accidents are within the guidelines set forth in 10 CFR 100. Chapter 15 references the facility's Technical Specifications. The design criteria for structures and equipment are summarized in Chapter 16. The Entergy Quality Assurance Program Manual provides a description of Entergy's Quality Assurance organization for operation of the facility.

The design of Indian Point 3 is essentially the same as Indian Point 2 which was licensed for operation by NRC (FOL DPR-26) and is located at the same site as Indian Point 3. All functional and safety systems for Indian Point 3 are independent of the other units on the site, except for the following:

1) The common discharge canal, outfall structure and associated instrumentation and sampling systems,

2) Electrical supplies and interties,

3) Station air intertie,

4) Demineralized water, condensate makeup and hydrogen interties,

5) City water and fire protection interties,

6) Diesel fuel oil supply (dedicated service of No. 2 fuel oil) system,

7) Sewage treatment facility,

8) Chlorine supply system,

9) Carbon dioxide supply system,

10) Auxiliary steam system intertie,

11) Service boiler fuel oil (No. 6) supply system,

12) Liquid SGBD radwaste processing and discharge (to Indian Point 1) facilities,

13) Chemistry facilities and equipment,

14) Instrumentation and control facilities and equipment, and

15) Environmental monitoring services.

Rules governing the use of these shared facilities have been mutually agreed on by the Authority and Consolidated Edison, which are now Indian Point Energy Center (IPEC) of Entergy Nuclear Northeast.

1.2 SUMMARY PLANT DESCRIPTION

1.2.1 Site

Indian Point 3 is located adjacent to and south of Indian Point 1 on a site of approximately 235 acres of land on the east bank of the Hudson River at Indian Point, Village of Buchanan in upper Westchester County, New York. The Indian Point 2 plant is adjacent to and north of Indian Point 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, Figure 1.2-1, shows the site and about 58 square miles of the surrounding area.

Meteorology

Meteorological conditions in the area of the site were determined during a two-year test program (1955 to 1957). The validity of these conclusions was verified by a test program completed in October 1970. The meteorological analysis also includes data from periods of November 26, 1969 through October 1, 1970, and January 1, 1970 through December 31, 1971. 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 indicated that the site meteorology provides adequate diffusion and dilution of any released gases.

Geology and Hydrology

Geologically, the site consists of a hard limestone formation 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 square foot, 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 and examination of soil borings have confirmed the above conclusion. (See Section 2.7)

In the Hudson River, about 80,000,000 gallons of water flow past the plant each minute during peak tidal flow. This flow provides additional mixing and dilution for liquid discharges from the facility. In fact, 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 to reduce concentrations of radioactive discharges by dilution with ordinary plant effluents to concentrations that would be tolerable for drinking water. There is no danger of flooding at the site as discussed in Section 2.5.

Seismology

Seismic activity in the Indian Point area is rare and no damage has resulted therefrom. As stated in Section 2.8, the site is “practically non-seismic” and is “as safe as any area at present known.” Not withstanding such assurance, the safety related equipment, components and structures of the plant were designed to withstand an earthquake of the highest intensity which can reasonably be predicted from geologic and seismic evidence developed for the site.

Environmental Radiation Monitoring

Environmental radioactivity has been measured at the site and surrounding area in association with the operation of Indian Point 1, 2 and 3. These measurements are continued and reported as required by the Technical Specifications. The radiation measurements of fallout, water samples, vegetation, marine life, etc. have shown no significant 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 two-year post-operative study, see Section 2.9, found that environmental radioactivity in the vicinity of the site is no higher than anywhere else in the State of New York.

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 3 nuclear power plant. Accident analyses presented in Chapter 14 verify that the maximum expected doses at or beyond the site boundary are well within the 10 CFR 100 suggested guidelines.

1.2.2 Plant Description

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

The general layout and profile of the plant are shown on Figures 1.2-1, 1.2-2, 1.2-4 and plant drawing 9321-F-64513 [Formerly Figure 1.2-3].

Nuclear Steam Supply System

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. An electrically heated pressurizer is connected to the loop associated with Steam Generator 34.

The reactor core is composed of uranium dioxide pellets enclosed in Zircaloy or Zirlo™ 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 Zircaloy or Zirlo and guide tubes located within the fuel assembly. The core was initially loaded in three regions of different enrichments. At successive refuelings fuel not designated for the next core is removed from the core and discharged to the spent fuel storage facility. The fuel elements that remain are arranged in accordance with the new core design, and new fuel is introduced.

The steam generators, which were installed during the cycle 6/7 refueling outage are vertical U-tube units employing Inconel 690 tubes. Integral separating equipment reduces the moisture content of the steam leaving the steam generators to 0.10 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) Change 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 shutdown
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
12) Dispose of gaseous and solid wastes.

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 ± 10 % and ramp load changes of ± 5% per minute over the load range of 15% to, but not exceeding, 100% power under normal operating conditions subject to xenon limitations.

Turbine and Auxiliaries

The turbine is a tandem-compound-unit, comprising one high pressure and three low pressure cylinders which rotate at 1800 rpm. The unit is equipped with 44 inch exhaust blading in the low pressure cylinders. Six combination moisture separator-reheater units are employed to dry and superheat the steam between high and low pressure turbine cylinders.

The turbine-generator is capable of sustaining a 50% loss of external electrical load without turbine or reactor trip. The turbine auxiliaries include surface condensers, steam jet air ejector, turbine driven main feed pumps, motor driven condensate pumps, six stages of feedwater heating, and moisture preseparators.

The turbine generator was designed for a guaranteed capability of 1,021,793 Kw at 1.5 in. HG absolute exhaust pressure with zero percent make up and six stages of feedwater heating.

Electrical System

The main generator feeds electrical power through an isolated phase bus to half-sized main power transformers.

The Auxiliary Electrical System provides power to those auxiliary components which are required to operate during any of the plant's normal emergency conditions of operation.

Standby power required for plant start-up and for plant operation following a reactor trip, and auxiliary power are supplied by a 138 kV system from the Buchanan substation through an auxiliary transformer.

Station auxiliaries receive power during normal operation from the auxiliary transformer connected to the isolated phase bus and from the auxiliary transformer connected to the 138kV system.

Emergency power supply for vital instruments and controls is from four 125 volt DC station batteries.

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

Control Room (CR)

The plant is provided with a reactor and turbine-generator control room, designed under seismic and tornado criteria, and which contains all the necessary instrumentation for the plant's operation under normal and accident conditions.

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

Diesel Generators

Three diesel generator sets supply emergency power for plant shutdown and essential safeguards operation in the event of the loss of all other AC auxiliary power.

Waste Disposal System

The Waste Disposal System (WDS) collects liquid, gaseous and solid wastes from plant operation and processes gaseous and solid wastes for discharge or removal from the plant site.

The WDS also provides the hydrogen and nitrogen required by the primary system during normal plant operation. All shipments of solid waste from the site are made in accordance with government guidelines.

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 minimize 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 for preventing fuel storage criticality.
3) Visual monitoring of the refueling process at all times.

Engineered Safety Features

The Engineering Safety Features for this plant have sufficient redundancy in component and power sources such that under the conditions of a hypothetical loss-of-coolant accident, the system can, even when operating with partial effectiveness, maintain the integrity of the containment and keep the exposure of the public below the limits of 10 CFR 100.

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 leak-tight 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) which provides borated water to cool the core in the event of a loss of coolant accident.

3) The Containment Air Recirculation Cooling and Filtration System which provides a heat sink to cool the containment atmosphere and provides filtration of the containment atmosphere to remove airborne particulate and halogen fission products which form the source for potential public exposure.

4) The Containment Spray System which provides a spray of cool, chemically treated, borated water to the containment atmosphere as a backup heat sink, and iodine removal capability for the Containment Air Recirculation Cooling and Filtration System.

Structures

The major structures are the reactor containment building, the auxiliary building, the control building, the turbine building, the administration building, the outage support building, the training building and the Condensate Polisher Building. General layouts and interior component arrangement of the reactor containment structure are shown in Section 5.1. General layouts and interior components arrangement of the auxiliary building, spent fuel pit building, and holdup tank building are shown in Plant Drawings 9321-F-25153, -25113, -25143, and -25173 [Formerly Figures 1.2-5 through 1.2-8]. The general arrangement of the turbine building is shown in Section 10.2. General layout and component arrangement of the Control Room is presented in Section 7.7.1.

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. It is virtually leaktight and provides adequate radiation shielding for both normal operation and accident conditions.

Continuously pressurized double containment is provided at nearly all liner seams and penetrations including access openings and ventilation ducts.

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, there will be no leakage from containment to the environment.

Ground accelerations for the operational basis earthquake used for containment design purposes and for all Seismic Class I structures (Chapter 16) are 0.1g applied horizontally and 0.05g applied vertically. In addition, ground accelerations for the design basis earthquake of 0.15g horizontal and 0.10g vertical were used to analyze the no loss-of-function concept.

The primary auxiliary building, the spent fuel pit and control building are Seismic Class I structures. The turbine building is a Seismic Class III structure modified in accordance with the design basis earthquake criteria to preclude collapse or other damage to nearby Class I structures.

1.3 GENERAL DESIGN CRITERIA

The General Design Criteria establish the necessary design, fabrication, construction, testing and performance requirements for structures, systems, and components important to safety; that is structures, systems and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants.

The General Design Criteria presented and discussed in specific sections of the FSAR (which describe systems, structures, equipment and components important to safety) are those which were in effect at the time when Indian Point 3 was designed and constructed. The General Design Criteria which formed the bases for the Indian Point 3 design were published by the Atomic Energy Commission in the Federal Register of July 11, 1967 and subsequently made part of 10 CFR 50.

The Authority completed a study of the method by which the Indian Point 3 facility complied with the safety rules and regulations, in particular those contained in 10 CFR Parts 20 and 50, that were in effect at the time of the study. The study was conducted in accordance with the provisions of NRC Confirmatory Order of February 11, 1980 and were submitted to the NRC on August 11, 1980. The NRC audit of submittal indicated that the Indian Point 3 design and operation meet the applicable regulations. The following sections provide the results of the compliance study, updated to reflect changes made to the configuration since the study was completed.

1.3.1 Overall Requirements (Criteria 1 to 5)

Quality Standards and Records (Criterion 1)

Criterion: Structures, systems and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems and components important to safety shall be maintained by or under the control of the nuclear power plant licensee throughout the life of the unit.

The following list of Indian Point 3 FSAR Sections detail the quality standards used in the design, fabrication, erection and testing of structures, systems and components important to safety.

Each section identifies those specific codes and standards used, including the scope of each, and any supplements or modifications.

Section		Title
3.3.3.1	-	Fuel Quality Assurance
4.1.2	-	Reactor Coolant System, General Design Criteria - Quality Standards, and Records Requirements
4.1.7	-	Reactor Coolant System, Codes and Classifications
4.3.1	-	Reactor Coolant System, Safety Factors – Reactor Coolant Pump Flywheel, Methods of Quality Assurance
4.5	-	Reactor Coolant System, Inspections and Tests
5.1.1.1	-	Containment System Structures, Principal Design Criteria – Quality Standards and Records Requirements
5.1.1.5	-	Containment System Structures, Codes and Classifications
6.2.1	-	Safety Injection System, Design Basis – Codes and Classifications

6.3.1	-	Containment Spray System, Design Basis – Codes and Classifications
6.3.5	-	Containment Spray System, Inspections and Tests
6.8.4	-	Hydrogen Recombination System, Inspections and Tests
9.2.1	-	Chemical and Volume Control System, Design Basis - Codes and Classification
9.2.5	-	Chemical and Volume Control System, Tests and Inspections
9.3.1	-	Auxiliary Coolant System, Design Basis – Codes and Classifications
9.3.5	-	Auxiliary Coolant System, Tests and Inspections
9.5.5	-	Fuel Handling System, Tests and Inspections
13	-	Initial Tests and Operation
16.1.1	-	Definition of Seismic Design Classifications
16.1.4	-	Class I Design Criteria for Vessels and Piping

IP3's Quality Assurance Program is described in the Entergy Quality Assurance Program Manual.

<u>The Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Plant (Regulatory Guide 1.54, June 1973)</u>

The Quality Assurance requirements for protective coatings comply with Regulatory Guide 1.54, with the following exceptions:

> In lieu of the inspection defined in Section 6.2.4 of ANSI N101.4-1972, inspection is in accordance with ANSI N5.12-1974 Section 10, Inspection for Shop and Field Work.

Regarding the extent of coverage, the following offers clarification of paragraph 1.2.4 of ANSI N101.4-72:

Regulatory Guide 1.54 will be applied as follows:

a) Surfaces within the primary containment liner boundary:

i) For large surface area components, the documents shall be retained by Entergy as required by ANSI N101.4-72. These components include such items as the reactor building crane, containment, structural steel (including miscellaneous steel and handrails), concrete, ductwork, uninsulated pipe, exterior of uninsulated tanks and vessels, and major equipment supports.

ii) For manufactured equipment such as pumps, motors, pipe hanger and supports, the documentation required by ANSI N101.4-72 shall be maintained in the Seller's files for the complete duration of the contract warranty/guarantee period. A certificate of compliance signed by responsible management personnel shall be furnished by the Seller.

b) Other surfaces where coating failure could compromise the design function of equipment or components intended to prevent or mitigate the consequences of postulate accidents which could affect the public health or safety.

Because of the impracticability of imposing the Regulatory Guide requirements on the standard shop process used in painting valve bodies, handwheels, electrical cabinetry and control panels, loud speakers, emergency light cases and like components, the Regulatory Guide will not be invoked for these items since the total surface are for which the requirements will be imposed.

The reference substitution of ANSI N5.12 as the basis for inspection, rather than ANSI N5.9 reflects a revision to a standard referenced in the basis document, ANSI N101.4.

Design Bases for Protection Against Natural Phenomena (Criterion 2)

Criterion: Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems and components shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

Seismic Class I structures were designed to maintain within the allowable stress limit a combination of primary steady state stresses with seismic stresses resulting from the application of seismic motion with a maximum ground acceleration of 0.05 g acting in the vertical and 0.1 g acting in the horizontal planes simultaneously.

Also, primary steady state stresses when combined with seismic stresses resulting from the application of seismic motion with a maximum ground acceleration of 0.10 g acting in the vertical and 0.15 g acting in the horizontal planes simultaneously, were 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.

The plant is safeguarded from tornadoes by the combined use of buildings and structures, designed to withstand tornadoes, and redundancy of components. All seismic Class I buildings and structures were designed to withstand tornado winds corresponding to 300 mph tangential velocities, traverse velocities of 60 mph and a differential pressure drop of 3 psi in 3 seconds with no loss of function. The exceptions to this include areas without safety related equipment or redundant equipment as discussed in FSAR Section 16.2-2.

Furthermore, using a Probable Maximum Hurricane at the Battery of 130 mph and an inland reduction factor of 0.7, a wind speed of 90 mph due to hurricane was derived for use in the design at Indian Point.

From the evaluation of flooding conditions at Indian Point, done by Environmental Science and Engineering Consultants (Section 2.5), it was concluded that the maximum elevation of water at Indian Point due to flooding and wave runup is 15 feet or less. The consultants arrived at the above conclusion after assuming a critical set of simultaneous occurrences of the following three severe conditions:

1) Probable maximum precipitation over the Ashokan Reservoir resulting in a dam failure

2) Runoff generated by standard project precipitation over the Hudson Basin

3) Peak storm surge resulting from standard project hurricane for the New York Harbor area.

Although the simultaneous occurrence of the above three conditions is extremely remote, using the above conditions or various other conditions described in Chapter 2, the flood effects were not governing in the design of the plant. The combination of the elevation of the plant structures, the load-bearing capacity of the intake structure and the Technical Requirements Manual (TRM) requirements on plant operation and service water pump protection, result in acceptable conditions to protect the plant against flooding.

From the total group of meteorological conditions using a wind speed of 30 mph, an effective fetch length of 5.2 miles and an average water depth of 25 feet, the significant wave height at the site will be 2.8 feet with a significant wave period of 5.9 sec. Maximum wave height will be 5.0 feet. (Section 2.6)

During a probable maximum hurricane condition with a wind speed of 90 mph, a water depth of 35 feet, and a fetch of 5.2 miles the wave height at the site will be 9.1 feet. (Section 2.6)

From the above, it is evident that design loads due to the effects of tornadoes, hurricanes and flood were determined after considering the most severe of the natural phenomena that have been historically reported for the site and surrounding area. It is also evident that sufficient margin for limited accuracy, quantity, and period of time in which historical data have been accumulated was appropriate for each phenomenon.

The earthquake response spectra were developed from the average acceleration velocity displacement curves presented in TID-7024, Nuclear Reactors and Earthquake, for large-magnitude earthquakes at moderate distances from the epicenter. As such, the curves are made up of the combined normalized response spectrum determined from components of four strong-motion ground accelerations: El Centro, California, December 30, 1934; El Centro, California, May 18, 1940; Olympia, Washington, April 13, 1949; and Taft, California, July 21, 1952.

In investigating the overall and local structural effects, the following appropriate combinations of the effects of normal and accident conditions with effects of the natural phenomena were made:

The maximum tornado wind load was combined with missile load, dead load and live load, or with 3 psi negative pressure, and missile loads yielding the most conservative load combination and/or the highest stress condition.
Tornado loads act independently of other severe loads, and were found to be small by comparison to seismic loading.

With the exception of the Containment, all other seismic Class I structures used the following load combinations:

1) Primary steady state stresses were combined with the seismic stress resulting from the application of seismic motion with a maximum ground acceleration of 0.05g acting in the vertical and 0.1 g acting in the horizontal planes simultaneously. Under this combination the stresses were 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 were combined with the seismic stress resulting from the application of seismic motion with a maximum ground acceleration of 0.10g acting in the vertical and 0.15g acting in the horizontal planes simultaneously. Under this combination, the stresses were 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.

In the case of the Containment the following load combinations were made:

a) The dead load of structure was increased or decreased by 5 percent and each separate result was added to the accident pressure which was increased by 50 percent and to load due to maximum temperature and load exerted by the liner based upon temperatures associated with 1.5 times accident pressure.

   Or,
   $C = 1.0\ D \pm 0.05\ D + 1.5\ P + 1.0\ (T + TL)$

b) As in 'a)', the dead load was added to the accident pressure which was increased by 25 percent and to the load due to maximum temperature gradient through the concrete shell and mat based upon temperatures associated with 1.25 times the accident pressure and to the load exerted by the liner based upon temperatures associated with 1.5 times the accident pressure, and to the load resulting from operational basis earthquake increased by 25 percent.

   Or,
   $C = 1.0\ D \pm 0.05\ D + 1.25\ P + 1.0\ (T' + TL') + 1.25\ E$

c) As in ‘a)’, the dead load was added to the accident pressure load, to the load due to maximum temperature gradient through the concrete shell and mat based upon temperature associated with the accident pressure, to the load exerted by the liner based upon temperatures associated with the accident pressure, to the load exerted by the liner based upon temperatures associated with the accident pressure and to the load resulting from design basis earthquake.

Or,
$C = 1.0\ D \pm 0.05\ D + 1.0\ P + 1.0\ (T'' + TL'') + 1.0\ E'$

d) As in ‘a)’, the dead load was added to tornado wind load and to pressure drop effect.

Or,
$C = 1.0\ D \pm 0.05\ D + 1.0\ W'$

Symbols used in the above formula are defined as follows:

C	=	Required load capacity 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 operational basis earthquake
E ’	=	Load resulting from design basis earthquake
T”	=	Load due to maximum temperature gradient through the concrete shell, and mat based upon temperature associated with the accident pressure
TL”	=	Load exerted by the liner based upon temperatures associated with the accident pressure
W’	=	Tornado wind load and the pressure drop effect

Electrical

All electrical systems and components vital to plant safety, including the emergency diesel generators were designed as seismic Class I so that their integrity is not impaired by the maximum potential earthquake, windstorms, floods or disturbances on the external electrical systems. Power, control and instrument cabling, motors and other electrical equipment required for operation of the engineered safety features are suitably protected against the effects of either a nuclear system accident or of external environment phenomena order to assure a high degree if confidence in the operability of such components in the event that their use is required. (Section 8.1)

The physical locations of electrical distribution system equipment is such as to minimize vulnerability of vital circuits to physical damage as a result of accidents. The 6990 volt switchgear and 480 volt load centers are located in areas which minimize their exposure to mechanical, fire and water damage. The 480 volt motor control centers associated with the nuclear steam supply system are located in the Primary Auxiliary Building. (Section 8.2)

The diesel generator units are located in a seismic Class I structure located near the Control Building.

Mechanical

Systems and components important to safety either were designed to withstand the effects of, or measures were taken in the plant design to protect against, earthquakes, high winds, sudden barometric pressure changes, flooding and other natural phenomena. This is applicable to all components except those specifically exempted (in accordance with the requirements of NUREG-0800) which serve exclusively as redundant components.

Chapter 16 discusses the systems and components important to safety which were designed to withstand the effects of earthquakes without loss of function. Seismic design criteria are provided in Sections 16.1 and 16.3.

Appropriate load combinations of the effect of normal and accident conditions with the effects of an earthquake are provided in Section 16.1.

System and components important to safety are protected from the effects of tornadoes by either being housed in structures designed to withstand such loadings or by providing sufficient systems and equipment redundancy. Section 16.2 describes the method of protection for these systems and components.

Flooding at the site would have to reach 15’-3” above mean sea level before it would seep into the lowest elevation of any of the buildings and thereby possibly affect safety related systems and components. Different flooding conditions governing the maximum flooding elevation at the site were investigated. Results confirmed the maximum water elevation to be lower than the critical elevation noted above (Section 2.5).

Fire Protection (Criterion 3)

Criterion: Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

The Authority conducted a review of the Indian Point 3 Fire Protection Program in response to a NRC request dated June 10, 1976. This review compared the existing fire protection provisions at Indian Point 3 with the guidelines set forth in Standard Review Plan 9.5.1, “Fire Protection,” dated May 1, 1976. This review further described the implementation of modifications or changes underway to meet the guidelines, and deviations from these guidelines and the basis thereof. As a result of this report and associated correspondence, the Commission issued Amendment No. 24 on March 6, 1979 to the Indian Point 3 Operating License which included a Fire Protection Safety Evaluation Report. The SER was later supplemented to resolve open items in the original issuance.

Section 2.0 of the March 6, 1979 SER entitled “Fire Protection Guidelines of the IP3 Fire Protection SER” cites GDC-3 as the basic criterion for fire protection and “Appendix A” of Branch Technical Position 9.5-1 as guidance for the implementation of GDC-3. The conclusion of the Commission’s SER, as amended on May 2, 1980, states, in part, that the provisions of section 2.0 are satisfied.

On February 17, 1981, 10 CFR 50.48 and Appendix R became effective. Appendix R to 10 CFR 50 established fire protection features required to satisfy Criterion 3 of Appendix A to 10 CFR 50 with respect to certain generic issues related to nuclear power plants licensed to operate prior to January 1, 1979. As a minimum, 10 CFR 50.48 required all licensees to conform to the requirements of Section III.G, III.J, and III.O, of Appendix R which address fire protection of safe shutdown capability, emergency lighting, and reactor coolant pump oil collection systems, respectively. Other sections of Appendix R apply to those licensees who had open items remaining from the BTP 9.5-1, Appendix A review. The review of Indian Point 3 to BTP 9.5-1, Appendix A, was completed, as documented in the NRC Safety Evaluation Reports dated March 6, 1979 and May 2, 1980.

A re-evaluation of Indian Point 3 against the requirements of Section III.G of Appendix R to 10 CFR 50 was completed in August 1984. The report submitted to the NRC on August 16, 1984 described the bases on which Indian Point 3 conformed to Section III.G of Appendix R. The report provided a historical chronology of correspondence between the NRC and the Authority on Appendix R compliance by summarizing all pertinent documentation submitted to the NRC in response to 10 CFR 50.48 and Appendix R through August 1984. A new report was issued in May 1995, which supersedes the August 1984 report. The new report will be maintained by periodic updates.

Environmental and Missile Design Bases (Criterion 4)

Criterion: Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing and postulated accidents including loss-of-coolant accidents. These structures, systems and components shall be appropriately protected against dynamic effects including the effects of missiles, pipe whipping and discharging fluids that may result from equipment failures and from events and conditions outside the nuclear power unit.

All piping, components and supporting structures of the Reactor Coolant System were designed as seismic Class I equipment and therefore are capable of withstanding:

1) The Operational Basis Earthquake acceleration within code allowable working stresses.

2) The Design Basis Earthquake acceleration acting in the horizontal and vertical direction simultaneously with no loss of function.

The Reactor Coolant System is located in the Containment Building. In addition to being a seismic Class I structure, the design of the containment structure also considered accidents and other applicable natural phenomena. Details of the containment design are given in Chapter 5.

The dynamic effects during blowdown following a Loss-of-Coolant Accident were evaluated during the detailed layout and design of the high pressure equipment and barriers which afford missile protection. Support structures were designed with consideration given to fluid and mechanical thrust loadings.

The Indian Point 3 steam generators are supported, guided and restrained in a manner which prevents rupture of the steam side of a generator, the steam lines or 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.

The mechanical consequences of a pipe rupture are restricted by design features so that the functional capability of the engineered safety features is not impaired. (Section 4.1)

Reactor Coolant System components are surrounded by a 3-foot thick concrete wall which serves as a missile and partial radiation shield. A 2-foot thick reinforced concrete floor covers the Reactor Coolant System.

There is a reinforced concrete missile shield wall around the pressurizer above the operating floor. The original design is to protect the containment steel liner from postulated valve piece or instrument missiles connected to the pressurizer. Currently these missiles have been shown not to be credible.

In 1986, a NRC SER stated, “The dynamic effects associated with postulated pipe breaks in the primary coolant system (hot legs, cold legs, cross over legs) need not be a design basis.” (NRC SER dated March 10, 1986).

A structure is provided over the control rod drive mechanism to block any missiles generated from fracture of the mechanism.

Systems and components containing hot pressurized fluids which might affect the engineered safeguards components were carefully checked against the possibility of being missile sources. Provisions were taken, when necessary, against the generation of missiles rather than allow missile formation and try to contain their effects.

Once the requirement that the above systems are not sources of missiles was set forth, the identification of potential deficiencies and the generation of corrective design modifications was initiated through the Quality Assurance Program. (Section 5.1)

Incoming and outgoing lines which penetrate the Reactor Containment are normally or intermittently open during reactor operation, and are connected to closed systems inside the Containment and protected from missiles throughout their length. (Section 5.2)

Sharing of Structures, Systems and Components (Criterion 5)

Criterion: Structures, systems and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

The only structure important to safety that is shared by the nuclear units at the site is the cooling water discharge canal which carries the safety related Service Water System discharge to the river. Since this channel is of sufficient capacity to handle the discharge flow from both operating units, sharing of this structure will in no way impair the ability of safety related systems in either of the nuclear units to perform their safety functions.

There are no safety related systems shared by the nuclear units at the site. [Deleted]

The city water supply system provides a backup source of water to the Condensate Water Storage Tank for the Auxiliary Feedwater System of Indian Point 3.

The Fire Protection Systems formerly shared between Indian Point 1, 2 and 3 have been separated to provide independent fire protection capability. Details of the system modification are addressed in Section 9.6.

The only components important to safety that are shared by the two operating nuclear units (Indian Point 2 and 3) are the backup fuel oil storage tanks for the emergency diesel generators. The fuel oil storage tanks dedicated to Indian Point 3 have a capacity sufficient to assure continuous operation of two of the three Indian Point 3 diesels at minimum safeguards load for a total of 48 hours. The additional fuel oil required for continuous operation for a minimum of seven days can be transported by truck from the 200,000 gallon fuel oil storage tank at the Buchanan Substation located immediately across Broadway and/or from other local oil supplies (Section 8.2).

1.3.2 Protection by Multiple Fission Product Barriers (Criteria 10 to 19)

Reactor Design (Criterion 10)

Criterion: The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

The reactor core, with its related control and protection system, was 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, trip of the turbine generator, loss of normal feedwater and loss of all offsite power.

The Reactor Control and Protection System was 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 limit for the analyzed accidents.

The integrity of fuel cladding is ensured by preventing excessive clad heating, excessive cladding stress and strain. This was achieved by designing the fuel rods so that the following conservative limits are not exceeded during normal operation or any anticipated transient condition:

1) Minimum DNB ratio equal to or greater than the applicable limit;

2) Fuel center temperature below melting point of UO_2 ;

3) Internal gas pressure less than the nominal external pressure (2250 psia) even at the end of life;

4) Clad stresses less than the Zircaloy and ZIRLO™ yield strengths;

5) Clad strain less than 1%;

6) Cumulative strain fatigue cycles less than 80% of design strain fatigue life.

The ability of the fuel, when designed and operated according to these criteria, to withstand postulated normal and abnormal service conditions as shown by analyses described in Chapter 14 has been ensured. These analyses have been and will be amended as necessary for each cycle by the corresponding reload safety evaluations.

The Indian Point 3 Technical Specifications establish reactor criticality limits on moderator temperature coefficient and minimum temperature.

Doppler and Power Coefficients

The Doppler coefficient is defined as the change in neutron multiplication per degree change in fuel temperature. The coefficient is obtained by calculating neutron multiplication as a function of effective fuel temperature.

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 was taken to calculate the power coefficient, based on operating experience of existing Westinghouse cores. Section 3.2 provides the power coefficient as a function of power obtained by this method. The results presented do not include any moderator coefficient even though the moderator temperature changes with power level.

<u>Reactor Inherent Protection (Criterion 11)</u>

Criterion: The reactor core and associated coolant system shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

Prompt compensatory reactivity feedback effects are assured when the reactor is critical by the negative fuel temperature effect (Doppler effect) and by the non-positive operational limit on the moderator temperature coefficient of reactivity. The negative Doppler coefficient of reactivity is assured by the use of low enrichment fuel, the non-positive moderator temperature coefficient of reactivity is assured by keeping the dissolved absorber concentration below a certain limit through the use of burnable poison.

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, was evaluated by means of a detailed plant simulation. In these calculations, reactivity coefficients were 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 was established to determine the response of the plant throughout life and to establish the design of the Reactor Control and Protection Systems.

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 is directly proportional to the amount of reactivity controlled by the dissolved poison.

In order to insure a negative moderator temperature coefficient, burnable poison is incorporated in the core design. The result is a reduction in dissolved poison requirements so that changes in the coolant density will have less effect on the density of poison and the moderator temperature coefficient will remain negative.

The burnable poison is in one of three forms: 1. borated pyrex glass rods clad in stainless steel; 2. annular pellets of boron carbide aluminum oxides contained within two concentric Zircaloy tubings; or 3. integral boron coatings ZrB2 applied directly to the fuel pellets themselves. The moderator temperature coefficient is negative at operating conditions with burnable poison rods installed.

The effect of burnup on the moderator temperature coefficient was 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.

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.

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 results in a high pressurizer pressure trip and thereby prevents fuel damage for this transient. A loss of external electrical load of 50% 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 limit 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. Protection circuits, actuated directly by the coolant conditions identified with core limits, are therefore effective in preventing core damage.

Suppression of Reactor Power Oscillations (Criterion 12)

Criterion: The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

The potential for possible spatial oscillations of power distribution for this core has been reviewed. It was concluded that low frequency xenon oscillations may occur in the axial dimension. Part length rods, originally supplied, were removed from Indian Point 3; axial xenon oscillations can be controlled with full length rods. The core is stable to xenon oscillations in the X-Y dimension. Out-of-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. Extensive analysis, with confirmation of methods by spatial transient experiments at Haddam Neck, has shown that any induced radial or diametral xenon transients would die away naturally.[*] (Incore instrumentation is used to periodically calibrate and verify the information provided by the out-of-core instrumentation.)

Instrumentation and Control (Criterion 13)

Criterion: Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can effect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

The plant is equipped with Nuclear, Process, and Incore Instrumentation Systems to monitor variables and systems for normal operation and accident conditions to assure safety. Controls are provided to maintain these variables and systems within prescribed operating ranges.

Additionally, two independent systems provide Reactor Coolant System (RCS) water level indication during reduced RCS inventory conditions. These systems have been installed in response to NRC Generic Letter 88-17, Loss of Decay Heat Removal. Two additional independent indicators have been added to provide RCS water level during reduced RCS inventory conditions that are also qualified to function if the RCS is in a vacuum condition, such as RCS vacuum refill. These twp independent indicators are used with hand held UT devices to comply with the requirements of Generic Letter 88-17.

The Nuclear Instrumentation System is provided to monitor the reactor power from source range, through the intermediate range and power range, up to 120 percent of full power. The System consists of eight independent detectors in six instrument wells located around the reactor with associated equipment designed to provide indication and control in the Control Room for reactor operation and protection. (Section 7.4)

The source range neutron detectors (two) are proportional counters with a nominal sensitivity of 10 counts per sec per neutron per sq cm per second.

The detectors sense thermal neutrons in the range from 10^{-1} to $5x\ 10^{4}$ neutrons per sq cm per sec.

The intermediate range neutron detectors (two) are compensated ionization chambers that sense thermal neutrons in the range from 2.5×10^{2} to 2.5×10^{10} neutrons per sq cm per sec and have a nominal sensitivity of 4×10^{-14} amperes per neutron per sq cm per second. These detectors are located in the same detector assemblies as the proportional counters for the source range channels.

The power range consists of four independent long uncompensated ionization chamber assemblies. Each assembly is made up of two sensitive lengths. One sensitive length covers the upper half of the core, and the other length covers the lower half of the core. The arrangement provides in effect a total of eight separate ionization chambers approximately one-half the core height. The eight uncompensated (guard-ring) ionization chambers sense thermal neutrons in the range from 5×10^{2} to 1×10^{11} neutrons per sq cm per sec. Each has a nominal sensitivity of 3.1×10^{-13} amperes per neutron per sq cm per sec. The four long ionization chamber assemblies are located in vertical instrument wells adjacent to the four “corners” of the core. The assembly is manually positioned in the assembly holders and is electrically isolated from the holder by means of insulated stand-off rings.

The electronic equipment for each of the source, intermediate and power range channels is contained in draw-out panels mounted in racks in the Control Room.

Additionally, an Excore Neutron Flux Detection System has been installed per Regulatory Guide 1.97 requirements. The system consists of two redundant trains providing full scale (10^{-6} to 100% full power) flux indication only in the Control Room via the Qualified Safety Parameters Display System (QSPDS). The detectors are high-sensitivity fission chambers sensing thermal neutrons in the range from 10^{-2} to 10^{10} neutrons per sq cm per sec and are located at the 90° and 270° instrument wells. (Refer to section 7.4.2).

* WCAP-7407-LK, “Power Maldistribution Investigations,” R. F. Barry (1970)

The Process Instrumentation System measures temperature, pressure, flow, and level in the Reactor Coolant System, steam system, Reactor Containment, and auxiliary systems. Process variables required on a continuous basis for the startup, operation, and shutdown of the unit are indicated, recorded, and controlled from the Control Room. 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.

Certain controls which require a minimum of operator attention, or are only in use intermittently, are located on local control panels near the equipment to be controlled. Monitoring of the alarms of such control systems is provided for the Control Room.

Redundant instrument channels are provided for all safety systems. Where wide process variable ranges and precise control are required, both wide range and narrow range instrumentation are utilized. Visual and audible alarms are provided in the Control Room to annunciate off-normal and abnormal conditions.

The incore instrumentation yields information on the neutron flux distribution and fuel assembly coolant outlet temperatures at selected core locations. The system provides a means for acquiring data and performs no operational plant control, although it is used for excore detector calibration and to insure that power peaking factors are within limits.

The Incore Instrumentation System consists of thermocouples, positioned to measure fuel assembly coolant outlet temperature at preselected locations; and flux thimbles, which run the length of selected fuel assemblies to measure the neutron flux distribution within the reactor core.

The data obtained from the incore temperature and flux distribution instrumentation system, in conjunction with previously determined analytical information, can be used to determine the fission power distribution in the core at any time throughout core life. This method is more accurate than using calculational techniques alone. Once the fission power distribution has been established, the maximum power output is primarily determined by thermal power distribution and the thermal and hydraulic limitations determine the maximum core capability.

The incore instrumentation provides information which may be used to calculate the coolant enthalpy distribution, the fuel burnup distribution, and an estimate of the coolant flow distribution.

Both radial and azimuthal symmetry of power may be evaluated by combining the detector and thermocouple information from the one quadrant with similar data obtained from the other three quadrants.

During reduced inventory conditions, RCS level indication is provided by two independent indications. One system, the mid-loop level monitoring system, upgrades a previously existing tygon tube level indication system. The upgraded system consists primarily of two parallel level columns which are vented to the pressurizer through a common line, a video camera located inside Containment, and a video monitor located in the Control Room. The system provides Operators in the Control Room with continuous indication of RCS water level.

The second system, the mid loop Ultrasonic Level Measuring System (ULMS), consists of a sensor, preamp/pulsar module and signal processing module. When used, the ULMS output will be connected to a recorder located in the Control Room panel SFF, and an audible alarm will be installed prior to the start of the Cycle 8/9 refueling outage.

A third system, the Mansell Level Monitoring System (MLMS) provides two independent differential pressure sensing level indication loops which provides RCS level indication on computer monitors as well as LCD displays. Each level indication loop uses hand held UT devices during RCS mid-loop operation to increase the range of level indication to the bottom of the hot leg pipe and to comply with Generic Letter 88-17 requirements.

These systems can be used to monitor RCS water level during mid-loop operation. All systems provide indication. The ULMS and MLMS system also provide audible alarms.

Reactor Coolant Pressure Boundary (Criterion 14)

Criterion: The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

The Reactor Coolant System in conjunction with its control and protection provisions was designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions, and to maintain stresses within applicable code stress limits.

Fabrication of the components which constitute the pressure retaining boundary of the Reactor Coolant System was carried out in strict accordance with applicable codes. In addition, there are areas where equipment specifications for Reactor Coolant System components go beyond the applicable codes in order to assure design conservatism. 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 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 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.

Isolatable 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. (Section 4.1)

The Reactor Coolant Pressure Boundary was designed to reduce to an acceptable level the probability of a rapidly propagating type failure.

Assurance of adequate fracture toughness in the reactor vessel material was provided by compliance insofar as possible with the requirements for fracture toughness testing included in the Summer 1972 Addenda to Section III of the ASME Pressure Vessel and Boiler Code. In cases where it was not possible to perform all tests in accordance with these requirements, conservative estimates of material fracture toughness were made using information available.

Assurance that the fracture toughness properties remain adequate throughout the service life of the plant, is provided by a radiation surveillance program.

Safe operating heatup and cooldown limits are established using methodology similar to 1996 Section III, ASME Pressure Vessel and Boiler Code, Appendix G.

Changes in fracture toughness of the core region plates, weldments, and associated weld heat affected zone due to radiation damage are monitored by a surveillance program based on ASTM E-185, Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels. The evaluation of the radiation damage in this surveillance program is based on pre-irradiation testing by Charpy V-notch, dropweight test, and tensile specimens and post-irradiation testing of Charpy V-notch, tensile, and wedge opening loading specimens carried out during the lifetime of the reactor vessel. Specimens are irradiated in capsules located near the core mid-height and removed from the vessel at specified intervals.

All pressure-containing components of the Reactor Coolant System were designed, fabricated, inspected and tested in conformance with applicable codes.

The NRC has concluded that the current IP3 leakage detection system capability is adequate to continue to support the technical bases cited in the NRC's March 10, 1986, SE approving Leak Before Break (LBB) for the IP3 Primary Coolant Loop piping. This position was further clarified in the IP3 Supplement to Safety Evaluation re: Leakage Detection Systems (TAC No. MB3328, dated 01/30/02).

Reactor Coolant System Design (Criterion 15)

Criterion: The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

The Reactor Coolant System design and operating pressure, together with the safety, power relief and pressurizer spray valves set points and the protection system set point 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.

The design temperature for each component was 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.

All components in the Reactor Coolant System were 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 operation.

The number of thermal and loading cycles used for design purposes and the bases thereof are given in Table 4.1-8. During unit startup and shutdown, the rates of temperature and pressure changes are limited as indicated in Section 4.4.1. The cycles were estimated for equipment design purposes (40 year life) and are not intended to be an accurate representation of actual transients of actual operating experience.

To provide the necessary high degree of integrity for the equipment in the Reactor Coolant System, the transient conditions selected for equipment fatigue evaluation were based on a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from normal operation, normal and abnormal load transients. To a large extent, the specific transient operating conditions considered for equipment fatigue analyses were based upon engineering judgment and experience. Those transients were chosen which are representative of transients to be expected during plant operation and which are sufficiently severe or frequent to be of possible significance to component cyclic behavior.

The following cyclic loads were considered in the design of reactor coolant system components:

1) Heatup and Cooldown
2) Unit Loading and Unloading
3) 10% Step Load Increase and Decrease
4) Large Step Decrease in Load
5) Reactor Trip from Full Power
6) Hydrostatic Test Conditions
7) Loss of Load Without Immediate Turbine Reactor Trip
8) Loss of Flow

Over the range from 15% of full power to and including but not exceeding 100% of full power, the Reactor Coolant System and its components were designed to accommodate +10% of full power step changes in plant load and + 5% of full power per minute ramp changes without reactor trip. The Reactor Coolant System can 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% of full power without reactor trip.

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 only component of the Reactor Coolant System which is exposed to a significant level of neutron irradiation and it is therefore the only component which is subject to material radiation damage effects.

The NDTT shift of the vessel material and welds, due to radiation damage effects, is monitored by a radiation damage surveillance program which conforms with ASTM E-185 standards.

Reactor vessel design was 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 Section III of the ASME Boiler and Pressure Vessel Code for Class "A" vessels, the unit operating conditions were 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.

All pressure containing components of the Reactor Coolant System were designed, fabricated, inspected and tested in conformance with the applicable codes listed in Table 4.1-9.

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.

Containment Design (Criterion 16)

Criterion: Reactor containment and associated systems shall be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

The Reactor Containment completely encloses the entire reactor and the Reactor Coolant System and ensures that essentially no leakage or radioactive materials to the environment would result even if a design basis Loss-of-Coolant Accident were to occur. The liner and penetrations were designed to prevent any leakage through the Containment. The structure provides biological shielding for both normal and accident situations.

The Reactor Containment was designed to safely withstand several conditions and their credible combinations. The major loading conditions were: (Section 5.1)

a) Occurrence of a gross failure of the Reactor Coolant System which creates a high pressure and temperature condition within the Containment.
b) Coincident failure of Reactor Coolant System, an earthquake or wind.

The design basis accident pressure load is discussed in Section 5.1. The design value is at least 5 percent in excess of the maximum calculated containment pressure as discussed in Section 14.3.6.

The design basis accident containment temperature induces loads in the concrete shell as the concrete acts to restrain liner thermal expansion. This thermal load effect on the liner is combined with pressure load effects to develop design basis accident design load requirements as a function of time. Accident temperature induced thermal gradients through the wall were not a factor in concrete shell design since the accident temperature effect penetrates approximately 10 percent of the containment wall thickness during the significant overpressure phase of the accident and the cracking of the concrete shell due to containment pressurization acts to relieve secondary stresses induced by thermal gradient effect. (Appendix 5A)

Quality standards of material selection, design, fabrication, and inspection of features (which are essential to the prevention, or mitigation of the consequences, of nuclear accidents) conform to the applicable provisions of recognized codes and good nuclear practice. The concrete structure of the Reactor Containment conforms to the applicable portions of ACI-318-63. These components are considered ASME Section XI Class or CC components and any repair or replacement activities shall be performed in accordance with ASME Section XI Subsections IWE and IWL of the ASME Code, with certain exceptions whenever specific relief is granted by the NRC. Further elaboration on quality standards of the Reactor Containment is given in Section 5.1.

All components and supporting structures of the Reactor Containment were designed so that there is no loss of function of such equipment in the event of design basis 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, was included in the design analysis.

Penetrations are equipped with double seals which are continuously pressurized above accident pressure. 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 utilizes a supply of clean, dry, compressed air which places 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 at any time. Capability is provided for testing the functional operability of valves and associated apparatus during periods of reactor shutdown.

Section 14.3.6 provides details of the analysis conducted to assure the integrity of the Reactor Containment Building for the duration of a postulated major Reactor Coolant System pipe rupture.

The OEH is designed to accommodate transportation of large equipment in and out of the Containment Building expeditiously during outage, and shall provide an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment due to events postulated during refueling accident conditions.

The roll-up door is an alternate device that is capable of rapid closure. It is effectively an airtight, but not pressure-resistant, door that when closed prevents direct communication between the containment atmosphere and the outside atmosphere.

Subsequent to a loss RHR cooling as defined in ITS 3.9.4 and 3.9.5, the roll-up door provides rapid containment closure until either cooling is restored, or the main equipment hatch (or OEH) may be installed within four hours.

Electrical Power Systems (Criterion 17)

Criterion: An onsite electric power system and an offsite electrical power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that:

1) Specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences.

2) The core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electrical power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be

> designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limit and design limit and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a Loss-of-Coolant Accident to assure that the core cooling, containment integrity, and other vital safety functions are maintained.
>
> Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

Independent alternate power systems are provided with adequate capacity and testability to supply the required Engineered Safety Features and protection systems. The plant is supplied with normal, standby (offsite) and emergency (onsite) power sources as follows:

1. The normal source of auxiliary power during plant operation is supplied from both the plant's generator and offsite power.

2. Offsite power is supplied from Buchanan Substation (approximately ¾ mile from the plant) by 138kV and 345 kV feeders, and two underground 13.8 kV feeders. The Buchanan Substation has two 345kV and two 138 kV circuits to Millwood Substation and a 345kV circuit to Ladentown Substation which interconnects with the PJM system. Millwood Substation has ties to Pleasant Valley Substation which is the interconnection point between Consolidated Edison Co. and Niagara Mohawk and Connecticut Light and Power system. [Deleted] The 138kV feeders are connected to the 6.9 kV buses through the station auxiliary transformer, the 13.8 kV feeders and combustion turbines are connected to the 6.9 kV buses through autotransformers. The 480 volt engineered safety features buses are connected to the 6.9 kV buses through station auxiliary transformers.

3. Three diesel generators are each connected to their respective engineered safety features buses to supply emergency shutdown power in the event of loss of all other AC auxiliary power. There are no automatic ties between the buses associated with each diesel generator. Each diesel will be started automatically on a safety injection signal or upon the occurrence of under voltage on its associated 480 volt bus. Any two diesels have adequate capacity to supply the engineered safety features for the hypothetical accident concurrent with loss of outside power. This capacity is adequate to provide a safe and orderly plant shutdown in the event of loss of outside electrical power. The diesel generator units are capable of being started and sequence load begun within 10 seconds after the initial signal. Interlocks are provided so that a fault on any bus will lock out all possible sources of power to that bus. Interlocks are provided to prevent circuit breakers connecting emergency diesel generators No. 31, 32, and 33 to buses No. 2A, 6A, and 5A from automatically closing if there is a voltage on the bus. (Chapter 8)

Each of the four 480 volt switchgear bus sections which supply power to the safeguards equipment receives DC control power from its associated battery source. Batteries No. 31, 32, and 33 supply DC control power to 480 volt buses No. 5A, 6A and 2A/3A respectively. The 125 volt DC system is divided into five buses with one battery and battery charger (supplied from the 480 volt system) serving each. The battery chargers supply the normal DC loads as well as maintaining proper charges on batteries. Battery chargers 31, 32 and 33 are also relied upon to support the continued operation of systems and components required to either mitigate the consequences of a Design Basis Accident or provide post-accident monitoring subsequent to depletion of batteries No. 31, 32 and 33. The DC system is redundant from battery source to actuation devices which are powered from the batteries. There are five station batteries which feed five DC power panels, two of which (PP-31 and 32) in turn feed six DC distribution panels. Four of the main distribution power panels feed four relay and instrument DC buses. Redundant safeguards relays which use DC as a power source receive their power from one of the three buses. The 120 volt AC instrument supply is split into four buses, all are fed by inverters which are in turn supplied from separate 125 volt DC buses.

The Electrical Systems, under single failure conditions with necessary automatic switching, can still provide necessary power for the safeguard equipment in order to maintain plant safety functions. The plant auxiliary equipment is arranged electrically so that multiple items receive their power from different sources. (Section 8.2)

<u>Inspection and Testing of Electrical Power System (Criterion 18)</u>

Criterion: Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important area and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically:

a) The operability and functional performance of the components of the systems such as onsite power sources, relays, switches, and buses.

b) The operability of the systems as a whole and, under conditions as close to design as practical, the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system and the onsite power system.

Electric power systems important to safety were designed as described below in order that the above mentioned aspects of the system can be periodically tested.

The Electrical Systems, under single failure conditions with necessary automatic switching, can still provide necessary power for the safeguard equipment to maintain plant safety functions. The 480 volt equipment is arranged on 4 buses. The 6900 volt equipment is supplied from 6 buses. The bus arrangements specified for operations ensure that power is available to an adequate number of safeguards auxiliaries.

Any two of the three diesel generators, the station auxiliary transformer or the separate 13.8

to 6.9 kv transformers are each capable of supplying the minimum safeguards loads and therefore provide separate sources of power immediately available for operation of these loads (see the Technical Specifications). The presence of these independent supply sources and redundant equipment allows periodic testing of any one supply and associated equipment without falling below minimum requirements.

The minimum testing and surveillance performed on the emergency power system are detailed in the Technical Specifications.

Control Room (Criterion 19)

Criterion: A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 Rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

The plant is equipped with a Control Room which contains those controls and instrumentation necessary for operation of the reactor and turbine generator under normal and accident conditions, including Loss-of-Coolant Accidents.

The principal criteria of control station design and layout were that all controls, instrumentation displays and alarms required for the safe operation and shutdown of the plant are readily available to the operators in the Control Room.

Alarms and annunciators in the Control Room provide the operators with warning of abnormal plant conditions which might lead to damage of components, fuel or other unsafe conditions.

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.
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 and indicated or recorded from the Control Room, access to which is supervised. The quantity and type of process instrumentation provided ensures safe and orderly operation of all systems and processes over the full operating range of the plant.

Consideration was given to the fact that certain systems normally require more attention from the operator. The control system therefore is centrally located on a three section board.

Sufficient shielding, distance, and containment integrity is maintained to assure that control room personnel shall not be subjected to radiation doses, under postulated accident conditions, during occupancy of, ingress to and egress from the Control Room which, in the aggregate, would exceed the limits in 10 CFR 100.

All doors into the Control Room lead to enclosed areas like the Turbine Building and Control Building stairwell, and not to the outside. The Control Room was constructed of concrete with all openings around cables sealed airtight with a fireproof compound; hence the infiltration and exfiltration through walls, floor and ceiling is negligible. The fresh air intake for the control room air conditioning system is located in the east wall of the control building below the electrical tunnel between elevations 30'-0" and 18'-0". This is sheltered by an enclosure formed by the electrical tunnel floor above and the concrete walls on the south and east sides.

Since the Control Building is physically separate from the Containment and Primary Auxiliary Building, there are no sources of contaminated leakage which could result in airborne concentrations in the Control Building, Turbine Building or at the Control Room fresh air intake in excess of those associated with containment and auxiliary building releases. Also, should it become necessary, the Control Room was provided with self-contained breathing apparatus.

All fresh air entering the Control Room passes through a filter system consisting of a roughing filter, a one inch charcoal filter and a HEPA filter.

The charcoal filters were replaced on June 16, 1980 with a more efficient type of charcoal filter.
The installed Nuclear Grade Activated Charcoal is tested in accordance with ASTM D3803-1989, per the IP3 response to Generic Letter 99-02.

Separate chlorine, anhydrous ammonia, and carbon dioxide diffusion type probes will detect the presence of these gases in the control room outside the air intake duct. Upon a toxic gas detection alarm, the operator will be able to place the control room air conditioning system in the 100% recirculation mode to stop the intake of outside air. An additional toxic gas detection system indicates the oxygen, chlorine, and anhydrous ammonia levels in the control room atmosphere.

As a further measure to assure safety, provisions were made so that plant operators can shutdown and maintain the plant in a safe condition by means of controls located outside the Control Room. During such a period of control room inaccessibility the reactor will be tripped and the plant maintained in a hot shutdown condition. If the period extends for a long time, the Reactor Coolant System can be borated to maintain shutdown as xenon decays.

Local controls are located so that the stations to be manned and the times when attention is needed are within the capability of the plant operating staff. The plant intercom system will provide communication among the personnel so that the operation can be coordinated.

If the Control Room should have to be evacuated suddenly and no control room operator action is possible, the reactor can be tripped by either of the following actions:

1) Open rod control breakers in the Cable Spreading Room of the Control Building.
2) Actuate the manual turbine trip at the control station in the Turbine Building.

Following evacuation of the Control Room, the following systems and equipment are provided to maintain the plant in a safe shutdown condition from outside the Control Room:

a) Residual heat removal
b) Reactivity control, i.e., boron injection to compensate for fission product decay
c) Pressurizer pressure and level control
d) Electrical system as required to supply the above systems
e) Other equipment, as described below:
    - Level indication for each steam generator
    - Pressure indication for each steam generator
    - Level indication for the pressurizer
    - Pressure indication for the pressurizer
    - Auxiliary Feedwater Pump controls
    - Charging Pump controls
    - Boric Acid Transfer Pump controls
    - Service Water Pump controls
    - Containment Air Recirculation Fan controls
    - Control Room Air Handling Unit controls
    - Control Room Air Inlet Damper controls
    - Main Feedwater controls
    - Turbine Driven Auxiliary Feed Pump speed control
    - Auxiliary feedwater controls
    - Atmospheric dump controls
    - Pressurizer Heater backup controls
    - All other valves required for hot stand-by

Local STOP/START Motor Controls with a selector switch are provided at each of the motors for the above mentioned equipment. Placing the local selector switch in the local operating position gives an annunciation alarm in the Control Room and turns out the motor control position lights on the control room panel.

Following a normal plant shutdown an automatic steam dump control system bypasses steam to the condenser and maintains the reactor coolant temperature at its no load value. This implies the continued operation of the steam dump system, condensate circuit, condenser cooling water, feed pumps and steam generator instrumentation. Failure to maintain water supply to the steam generators would result in steam generator dry out after some 34 minutes and loss of the secondary system for decay heat removal. Redundancy and full protection, where necessary, was built into the system to ensure the continued operation of the steam generator units. If the automatic steam dump control system is not available, independently controlled relief valves on each steam generator units maintain the steam pressure. These relief valves are further backed up by ASME Code safety valves on each steam generator. Numerous calculations, verified by startup tests on the Connecticut-Yankee and San Onofre Power Plants have shown that with the steam generator safety valves operating alone the Reactor Coolant System maintains itself close to the nominal no load condition. The steam relief facility is adequately protected by redundancy and local protection. For decay heat removal it is only necessary to maintain control on one steam generator.

For the continued use of the steam generators for decay heat removal, it is necessary to provide a source of water, a means of delivering that water and, finally, instrumentation for pressure and level indication.

For plant shutdown the normal source of water supply is the auxiliary feed circuit; this implies satisfactory operation of the condenser, air ejector, condenser cooling circuit etc. In addition to the normal feed circuit the plant may fall back on:

1) The condensate storage tanks
2) The city water storage tank
3) The city water supply

Auxiliary feedwater may be supplied to the steam generators by the motor driven auxiliary feed pumps or by the steam driven auxiliary feed pump; these pumps and associated valves have local controls.

1.3.3 Protection and Reactivity Control Systems (Criteria 20 to 29)

Protection System Functions (Criterion 20)

Criterion: The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems to assure that specified acceptable Fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

The basic reactor tripping philosophy is to define a region of power and coolant temperature conditions allowed by the primary tripping functions, the overpower Δ T trip, the overtemperature Δ T trip and the nuclear overpower trip. The allowable operating region within these trip settings is provided to prevent any combination of power, temperatures and pressure which would result in DNB 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, low primary coolant flow trip, steam and feedwater flow mismatch trip, steam generator low-low water level trip, turbine trip, safety injection trip, source and intermediate range trips, and manual trip are provided to back up the primary tripping functions for specific accident conditions and mechanical failures.

A dropped rod signal blocks automatic rod withdrawal and also provides a turbine load cutback if above a given power level. The dropped rod is indicated from individual rod position indicators or by a rapid flux decrease on any of the power range nuclear channels.

Overpower Δ T, overtemperature Δ T, and T_{avg} deviation rod stops prevent abnormal power conditions which could result from excessive control rod withdrawal initiated by a malfunction of the Reactor Control System or by operator violation of administrative procedures.

Instrumentation and controls provided for the protective systems were designed to trip the reactor, when necessary, 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 Feature 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 were 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. The Engineered Safety Features Instrumentation System actuates (depending on the severitiy of the condition) the Safety Injection System, the Containment Isolation System, the Containment Air Recirculation System or 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 on Section 7.2.

The containment air recirculation coolers are normally in use during plant operation and would therefore not normally require an initiating signal. These units are, however, in the automatic sequence which actuates the Engineered Safety Features upon receiving the necessary actuating signals indicating an accident condition.

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 actuation of containment isolation is by coincident and redundant containment high pressure signals.

<u>Protection System Reliability and Testability (Criterion 21)</u>

Criterion: The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated.

> The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

The reactor uses the high speed version of 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 reactor internals, fuel assemblies, RCC assemblies and drive system components were 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. 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.

The Rod Cluster Control assembly guide system is locked together with pins throughout its length to ensure against misalignments which might impair control rod movement under normal operating conditions and credible accident conditions. An analogous system successfully underwent 4132 hours of testing in the Westinghouse Reactor Evaluation Center during which about 27,200 feet 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 primary reactor trip protection channels required during power operation were supplied with suffiicient redundancy to provide the capability for channel calibration and test at power.

Removal of one trip circuit is accomplished by placing that circuit in a tripped mode; i.e., a two-out-of-three circuit becomes a one-out-of-two circuit. Testing does not trip the system unless a trip condition exists in a concurrent channel.

Reliability and independence were 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 de-energize 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. (Section 7.2)

The Reactor Protection Systems were designed so that the most probable mode of failure (loss of voltage, relay 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 was carried throughout the system extending from the sensor to the relay actuating the protective function. The protective and control functions are fully isolated, 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 was used for pressurizer pressure and water level channels, steam generator water level, T_{avg} and Δ T channels, steam flow-feedwater flow and nuclear instrumentation channels.

The engineered safety features equipment is actuated by one or the other 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.

In the Reactor Protection System, two reactor trip breakers are provided to interrupt power to the full length i drive mechanisms. The breaker main contacts are connected in series (with power supply) so that opening either breaker interrupts power to all full length mechanisms, permitting them to fall by gravity into the core.

In summary, reactor protection was designed to meet all presently defined reactor protection criteria and is in accordance with the IEEE "Standard for Nuclear Plant Protection Systems 323".

The analog 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. The trip logic circuitry includes means to test each logic channel through the trip breakers. Thus, the operability of each trip channel can be determined conveniently and without ambiguity.

Testing of the diesel generator starting may be 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 volt bus for loading. Complete testing of the starting of diesel generators can be accomplished by tripping the associated 480 volt under- voltage relays or providing a coincident simulated safeguards signal. The ability of the units to start within the prescribed time and to carry load can be periodically checked. (The Electrical Systems are discussed in detail in Section 8.2.3)

<u>Protection System Independence (Criterion 22)</u>

Criterion: The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

The basic reactor tripping philosophy was to define a region of power and coolant temperature conditions allowed by the primary tripping functions, the overpower DT trip, the overtemperature DT trip and the nuclear over-power trip. The allowable operating region within these trip settings was provided to prevent any combination of power, temperatures and pressure which would result in DNB with all reactor coolant pumps in operation. Additional tripping functions such as high pressurizer pressure trip, low pressurizer pressure trip, high pressurizer water level trip, low reactor coolant 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, and manual trip are provided to back up the primary tripping functions for specific accident condition and mechanical failures. (Section 7.2)

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 were 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 protective function.

All primary reactor trip protection channels required during power operation are supplied with sufficient redundancy to provide the capability for channel calibration and test at power.

Removal of one trip circuit is accomplished by placing that circuit in a tripped mode; i.e., a two-out-of-three circuit becomes a one-out-of-two circuit. Testing does not trip the system unless a trip condition exists in a concurrent channel.

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 de-energize 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.

The Reactor Protection Systems were designed so that the most probable modes of failure (loss of voltage, relay 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 protection system trip, or violate reactor protection criteria.

The Engineered Safety Features Systems and the Reactor Protection System have a seismic classification of Class I and were designed in accordance with the Seismic Class I criteria set forth in Section 16.1.3.

In addition to the environmental design bases listed in IEEE-279 (1968), the Engineered Safety Features Systems and Reactor Protection System met the environmental requirements listed in Appendix 6F.

The components of the protection system were 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.

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 was used to the maximum practical extent to achieve separation of redundant transmitters. Separation of field wiring was achieved using separate wireways, cable trays, conduit runs and containment penetrations for each redundant channel. Redundant analog equipment was separated by locating redundant components in different protection racks. Each redundant channel is energized from a different AC instrument bus.

Each reactor trip circuit was designed so that trip occurs when the circuit is de-engerized; 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 de-energize when required is a mode of malfunction that affects only one channel. The trip signal furnished by the two remaining channels is unimpaired in this event.

Reactor trip is implemented by interrupting power to the magnetic latch mechanisms on all drives allowing the full length rod clusters to insert by gravity. The protection system is thus inherently safe in the event of a loss of power.

The engineered safety features actuation circuits were designed on the “energy to operate” principle unlike the reactor trip circuits which utilize the “de-engerize to operate” principle.

The physical arrangement of all elements associated with the protective system reduces the probability of a single physical event impairing the vital functions of the system.

System equipment was distributed between instrument cabinets so as to reduce the probability of damage to the total system by some single event.

Wiring between vital elements of the system outside of equipment housing was routed and protected so as to maintain the true redundancy of the systems with respect to physical hazards.

The same channel isolation and separation criteria as described for the reactor protection circuits were applied to the engineered safety features actuation circuits.

<u>Protection System Failure Modes (Criterion 23)</u>

Criterion: The protection system shall be designed to fall into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, pain, and radiation) are experienced.

Each reactor trip circuit was designed so that trip occurs when the circuit is de-energized; 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 de-energize when required is a mode of malfunction that affects only one channel. The trip signal furnished by the two remaining channels is unimpaired in this event.

Reactor trip is implemented by interrupting power to the magnetic latch mechanisms on all drives allowing the full length rod clusters to insert by gravity. The protection system is thus inherently safe in the event of a loss of power.

The engineered safety features actuation circuits were designed on the “energize to operate” principle unlike the reactor trip circuits which utilize the “de-engergize to operate” principle.

The steam line isolation signal on high-high containment pressure, which uses the same circuitry as the containment spray actuation signal, was also designed on the “engergize to operate” principle. The three high-high containment pressure instrument channels are powered from three separate independent sources (one channel from Instrument Bus No. 31 powered from Battery No. 31, the second channel from Instrument Bus No. 33 powered from Battery No. 33, and the third channel from Instrument Bus No. 34 powered from Battery No. 34). This assures operation of these containment pressure instruments in the event of a power failure to one of the instrument channels.

Automatic starting of the emergency diesel generators is initiated by under-voltage on any of the 480 volt buses or by the safety injection signal. Engine cranking is accomplished by a stored energy system supplied solely for the associated diesel generator. The undervoltage relay scheme was designed so that loss of 480 volt power does not prevent the relay scheme from functioning properly. (Section 7.2)

<u>Separation of Protection and Control Systems (Criterion 24)</u>

Criterion: The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

Channel independence is maintained throughout the protection system extending from the sensor to the relay actuating the protection function. The protection and control functions are fully isolated, 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 all reactor trip channels.

The design basis for the control and protection system permits the use of a detector for both protection and control functions. Where this is done, all equipment common to both the protection and control circuits are classified as part of the protection system. Isolation amplifiers prevent a control system failure from affecting the protection system. In addition, where failure of a protection system component can cause a process excursion which requires protective action, the protection system can withstand another, independent failure without loss of function. Generally, this is accomplished with two-out-of-four trip logic. Also, wherever practical, provisions are included in the protection system to prevent a plant outage because of a single failure of a sensor.

In summary, reactor protection was designed to meet all presently defined reactor protection criteria and is in accordance with the IEEE “Standard for Nuclear Plant Protection Systems 279” (Section 7.2).

Protection System Requirements for Reactivity Control Malfunctions (Criterion 25)

Criterion: The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

Reactor shutdown with rods is completely independent of the normal control functions since the trip breakers completely interrupt the power to the full length rod mechanisms regardless of existing control signals. Effects of continuous withdrawal of a rod control assembly and of deboration are described in Sections 7.3.1 and 7.3.2, and in Chapters 9 and 14.

The control rod drive mechanisms are wired into preselected banks, and these bank configurations are not altered during core life. The assemblies are therefore physically prevented from being withdrawn in other than their respective banks. Power supplied to the rod banks is controlled such that no more than two banks can be withdrawn at any time. The control 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 analysis assuming the simultaneous withdrawal of the combination of the two banks of the maximum combined worth at maximum speed.
Should a continuous control rod assembly withdrawal be initiated from a subcritical condition, the transient will be terminated by the following automatic safety features:

1) Power range flux level trip (low setting)
2) Power range control rod stop
3) Power range flux level trip (high setting)

Termination of the startup incident by the above protection channels prevents core damage. In addition, the reactor trip from high reactor pressure serves as a backup to terminate the incident before an overpressure condition could occur. (Section 14.1.1)

Taking into account the conservative assumptions under which the incident has been analyzed, it is concluded that in the unlikely event of a control rod assembly withdrawal incident, the core and Reactor Coolant System are not adversely affected, since the thermal power reached is only 84 percent of the nominal value and the core water temperature reached is 556.6 F for the nominal conditions.

This combination of thermal power and core water temperature results in a DNBR well above the applicable limit (per table 14.1-0). The peak average clad temperature of 590^{o}F is less than the nominal full power of 647.5^{o}F and thus there is no clad damage.

The automatic features of the Reactor Protection System which prevent core damage in a control rod assembly withdrawal incident at power include the following:

1) Nuclear power range instrumentation actuates a reactor trip if two out of four channels exceed an overpower setpoint

2) Reactor trip is actuated if any two out of four Δ T channels exceed an overtemperature Δ T setpoint

3) Reactor trip is actuated if any two out of four Δ T channels exceed an overpower Δ T setpoint

4) A high reactor coolant system pressure reactor trip

5) A high pressurizer water level reactor trip

6) In addition to the above listed reactor trips, there are the following control rod assembly withdrawal blocks:

   a) High nuclear flux (one out of four)
   b) Overpower Δ T (one out of four)
   c) Overtemperature Δ T (one out of four).

In the unlikely event of a control rod assembly withdrawal incident during power operation, the core and Reactor Coolant System are not adversely affected since the minimum value of DNBR reached is in excess of the applicable limit for all control rod assembly reactivity rates. Protection is provided by the high nuclear flux and the overtemperature Δ T trips. Section 14.1.2 describes the effectiveness of these protection channels.

The opening of the primary water makeup control valve provides the only supply of makeup water to the Reactor Coolant System which can dilute the reactor coolant. Inadvertent dilution can be readily terminated by closing this valve.

Information on the status of reactor coolant makeup is continuously available to the operator. Lights are provided on the control board to indicate the operating condition of pumps in the Chemical and Volume Control System. Alarms are actuated to warn the operator if boric acid or makeup water flow rates deviate from preset values as a result of system malfunction.

To cover all phases of unit operation, boron dilution during refueling, startup, and power operation were considered in the analysis presented in Section 14.1.5.

Because of the procedures involved in the dilution process, an erroneous dilution is considered incredible. Nevertheless, if an unintentional dilution of boron in the reactor coolant does occur, numerous alarms and indications are available to alert the operator to the condition. The maximum reactivity addition due to the dilution is slow enough to allow the operator to determine the cause of the addition and take corrective action before excessive shutdown margin is lost.

Reactivity Control System Redundancy and Capability (Criterion 26)

Criterion: Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

One of the two reactivity control systems employs rod cluster control assemblies to regulate the position of Ag-In-Cd neutron absorbers within the reactor core. The other reactivity control system employs the Chemical and Volume Control System to regulate the concentration of boron (neutron absorber) in the Reactor Coolant System.

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 change 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;

4) reactivity changes associated with instantaneous load changes.

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 specified to provide shutdown with sufficient margin with the most reactive rod stuck out of the core.

Control rod reactivity requirements at the beginning and end of life are recalculated for each reload.

Excess Reactivity Insertion Upon Reactor Trip

The control requirements are based on providing a 1.3% shutdown margin at hot, zero power conditions with the highest worth rod stuck in its fully withdrawn position and to prevent return to criticality following a credible steam-line break.

Calculated Rod Worths

The complement of 53 full length control rods arranged in the pattern shown in Figure 3.2-1 meets the shutdown requirements. Table 3.2-3 lists the calculated worths of this rod configuration for beginning of life, first cycle and end of life, first cycle.

In order to be sure of maintaining a conservative margin between calculated and required rod worths, an additional amount has been added to account for uncertainties in the control rod worth calculations. The calculated reactivity worths listed were decreased in the design by 10 percent to account for any errors or uncertainties in the calculation. This worth was 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 operating reactors shows the calculation to be well within the allowed uncertainty of 10 percent. The second reactivity control system is capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded.

Chemical Shim Control

Control to render the reactor subcritical at temperatures below the operating range is provided by a chemical neutron absorber (boron). The boron concentration during refueling was established as shown in Table 3.2-1, line 30. This concentration, together with the control rods, provides approximately 5 percent shutdown margin for these operations. For cold shutdown, at the beginning of core life, a concentration is sufficient for 1.3 percent shutdown with all but the highest worth rod inserted. The boron concentration during operation is well within solubility limits at ambient temperature. A boron concentration, sufficient to prevent dilution during refueling, is maintained in the spent fuel pit since the pit is directly connected with the refueling canal. This prevents dilution in the reactor cavity when the cavity is connected to the spent fuel pit.

The initial full power boron concentration with and without equilibrium xenon and samarium is calculated for each reload.

This initial boron concentration is that which permits the withdrawal of the control banks to their operational limits. The xenon-free hot, zero power shutdown with all but the highest worth rod inserted, can be maintained with a cycle specific boron concentration. This concentration is less than the full power operating value with equilibrium xenon.

Combined Reactivity Control System Capability (Criterion 27)

Criterion: The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

The reactor core, together with the Reactor Control and Protection Systems, was designed so that the minimum DNBR is greater than the applicable limit (Table 14.1-0) and there is no fuel melting during normal operation, including anticipated transients.

Sufficient shutdown capability is also provided to maintain the core subcritical 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 addition via the emergency core cooling system.

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 (Section 3.1).

Normal reactivity shutdown capability is provided within two seconds following a trip signal by control rod actuation. Boric acid injection is used to compensate for long term xenon decay transient and for plant cooldown. The shutdown capability prevents return to critical as a result of the cooldown associated with a primary/secondary 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 inject boric acid into the reactor coolant. Any charging pump and either boric acid transfer pump can be operated form diesel generator power on loss of station power. Using either one of the two boric acid transfer pumps, in conjunction with any one of the three charging pumps, the RCS can be borated to hot shutdown even with the control rods fully withdrawn. Additional boration would be used to compensate for xenon decay. At a minimum CVCS design boration rate of 132 ppm/hr, the boron concentration required for cold shutdown can be reached well before xenon decays below its pre-shutdown level. The RWST is a suitable backup source for emergency boration. When two charging pumps are used to transfer borated water from the RWST to the reactor coolant, the boron concentration required for cold shutdown can be reached before xenon decays below its full-power pre-shutdown level.

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.

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 intact (Section 6.2).

Reactivity Limits (Criterion 28)

Criterion: The reactivity control system shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

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 rods 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 coil actuation is sequenced to provide variable speed rod travel. The maximum reactivity insertion rate was analyzed in the detailed plant analysis assuming two of the highest worth groups to be accidentally withdrawn at maximum speed yielding reactivity insertion rates no greater than 75 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 80 steps per minute (~ 50 inches per minute).

The Reactor is capable of meeting the performance objectives throughout core life under both steady state and transient 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.

The control system and the operational procedures provide adequate control of the core reactivity and power distribution. The following control limits are met:

a) A minimum hot shutdown margin as shown in the Technical Specifications is available assuming a 10% uncertainty in the control rod calculation (Section 3.2)

b) This shutdown margin is maintained with the most reactive RCCA in the fully withdrawn position

c) The shutdown margin is maintained at ambient temperature by the use of soluble poison (Section 3.1)

The reactor coolant boundary is shown in Section 14.2.6 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.

Even on the most pessimistic basis, the rod ejection accident analysis indicated that the fuel and clad limits were not exceeded. It was concluded that there was no danger of sudden fuel dispersal into the coolant. The pressure surge was shown to be insufficient to exceed 3000 psia, and it was concluded that there was no danger of consequential damage to the primary circuit. The amount of fission products released as a result of clad rupture during DNB is considerably less than in the case of the double ended main coolant pipe break (the design basis accident), and therefore within the guidelines of 10 CFR 100.

Protection Against Anticipated Operational Occurrence (Criterion 29)

Criterion: The protection and reactivity control systems shall be designed to assume an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

The Reactor uses the high speed version of the Westinghouse magnetic-type control rod drive mechanisms. Upon a loss of power to the coils, the rod cluster control assemblies with full length absorber rods are released and fall by gravity into the core.

The reactor internals, fuel assemblies, RCC assemblies and drive system components were 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 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.

The Rod Cluster Control (RCC) assembly guide system is locked together with pins throughout its length to ensure against misalignments which might impair control rod movement under normal operating conditions and credible accident conditions. An analogous system successfully underwent 4132 hours of testing in the Westinghouse Reactor Evaluation Center during which about 27,200 feet of step-driven travel and 1461 trips were accomplished with test misalignment in excess of the maximum possible misalignment that may be experienced when installed in the plant.

The Reactor Protection System was designed so that the most probable modes of failure (loss of voltage, relay 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 a 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 protection system 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 are fully isolated, control being derived form 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 Δ T channels, steam flow, feedwater flow and nuclear instrumentation channels.

In the Reactor Protection System, two reactor trip breakers are provided to interrupt power to the full length rod drive mechanisms. The breaker 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.

The components of the protection system were 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.

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 was used to the maximum practical extent to achieve separation of redundant transmitters. Separation of field wiring was 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 redundant channel is energized from a different AC instrument bus (Section 7.2).

Each reactor trip circuit was designed so that trip occurs when the circuit is de-energized; therefore, loss of channel power causes the system to go into 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 de-energize when required is a mode of malfunction that affects only one channel. The trip signal furnished by the two remaining channels is unimpaired in this event.

Reactor trip is implemented by interrupting power to the magnetic latch mechanism on all drives allowing the full length rod clusters to insert by gravity. The protection system is thus inherently safe in the event of a loss of power.

The engineered safety features actuation circuits are designed on the “energize to operate” principle unlike the reactor trip circuits which utilize the “de-energize to operate” principle.

A loss of power in the Reactor Protective System causes the affected channel to trip. All bistables operate in a normally energized state and go to a de-energized state to initiate action. Loss of power thus automatically forces the bistables into the tripped state.

Availability of power to the engineered safety features instrumentation is continuously indicated. The loss of instrument power to the sensors, instruments, logic or actuating devices in the engineered safety features instrumentation, starts the engineered safety features equipment associated with affected channels, except for containment spray which requires instrument power for actuation. Steam line isolation on high-high containment pressure, which utilizes the same actuation circuitry as the containment spray actuation, also required power to actuate. The three high-high containment pressure instrument channels are powered from three separate, independent sources to assure operation in the event of a power failure to one of the instrument channels.

The Engineering 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 were 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 protective function. 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 Containment Spray System.

The passive accumulators of the Safety Injection System do not require signal or power sources to perform their function.

The core protective systems, in conjunction with inherent plant characteristics, were designed to prevent anticipated abnormal conditions from fuel damage exceeding limits established in Chapter 3, or Reactor Coolant System damage exceeding effects established in Chapter 4. In addition, the systems were designed to ensure that limits for energy release to the Containment and for radiation exposure (as in 10 CFR 100) are not exceeded.

Thus, due to the high quality of the components, system separation and redundancy, along with the inherent characteristics of the design, these systems have a very high probability of accomplishing their safety function in the event of anticipated operational occurrences.

1.3.4 Fluid Systems (Criterion 30 to 46)

Quality of Reactor Coolant Pressure Boundary (Criterion 30)

Criterion: Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

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 and 4.5. Particular emphasis was placed on the assurance of quality of the reactor vessel to obtain material whose properties are uniformly within tolerances appropriate to the application of design methods of the code.

The Reactor Coolant System in conjunction with its control and protective provisions was 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.

Fabrication of the components which constitute the pressure retaining boundary of the Reactor Coolant System was 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 material 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 lifetimes. 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.

Isolatable 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. (Section 4.1)

All pressure containing components of the Reactor Coolant System were designed, fabricated, inspected and tested in conformance with applicable codes (Section 4.1).

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 NRC has concluded that the current IP3 leakage detection system capability is adequate to continue to support the technical bases cited in the NRC's March 10, 1986, SE approving Leak Before Break (LBB) for the IP3 Primary Coolant Loop piping. This position was further clarified in the IP3 Supplement to Safety Evaluation re: Leakage Detection Systems (TAC No. MB3328, dated 01/30/02).

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. This equipment provides 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 was the detection of deviations from normal containment environmental conditions including air particulate activity, radiogas activity, humidity, condensate runoff. In addition, assuming no operator action, the liquid inventory in the process systems and containment sump can be used for gross indication of leakage. However, sensitivity of the processing systems and containment sump system can be improved with incorporation of a CR alarm (VC Sump Pump running) or operator actions to increase monitoring of the processing system (i.e. sump flow monitor once every 4 hours). (Section 4.1)

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) The containment sump pump system with incorporation of a CR alarm (VC Sump Pump running) or operator actions to increase monitoring of the processing system (i.e. sump flow monitor once every 4 hours), provides the capability of detecting a 1 gpm leak within four hours.

2) Two radiation sensitive instruments provide the capability for detection of leakage from the Reactor Coolant System. The containment air particulate monitor is quite sensitive to low leak rates. The containment radiogas monitor is much less sensitive but can be used as a backup to the air particulate monitor.

3) A third instrument used in leak detection is the humidity detector. This provides a means of measuring overall leakage from all water and steam systems within the containment but furnishes a less sensitive measure.

4) A leakage detection system is included which determines leakage losses from all water and steam systems within the containment including that from the Reactor Coolant System. This system collects and measures moisture condensed from the containment atmosphere by the cooling coils of the main recirculation units. It relies on the principle that all leakages up to sizes permissible with continued plant operation will be evaporated into the containment atmosphere. This system provides a dependable and accurate means of measuring integrated total leakage, including

leaks from the cooling coils themselves which are part of the containment boundary. This method provides a backup to the radiation monitoring methods.

5) An increase in the amount of coolant makeup water which is required to maintain normal level in the pressurizer; an increase in containment sump level; or the coolant inventory balance are alternate means of detecting leakage.

Leakage detection methods are described in detail and evaluated in Chapter 6.

Fracture Prevention of Reactor Coolant Pressure Boundary (Criterion 31)

Criterion: The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

The Reactor Coolant System, in conjunction with its control and protective provisions, was designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions, while maintaining stresses within applicable code stress limits.

Fabrication of the components which constitute the pressure retaining boundary of the Reactor Coolant System was 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 to insure conservative design. Details are given in Chapter 4.

The material 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 automatically initiated to maintain the required cooling capability and to limit system conditions so that continued safe operation is assured.

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.

Isolatable 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.

The Reactor Coolant Pressure Boundary was designed to reduce the probability of a rapidly propagating type failure to an acceptable level.

Assurance of adequate fracture toughness in the reactor vessel material is provided by compliance insofar as possible with the requirements for fracture toughness testing included in the Summer 1972 Addenda to Section III of the ASME Boiler and Pressure Vessel Code. In cases where it was not possible to perform all tests in accordance with these requirements, conservative estimates of material fracture toughness have been made using available information.

Assurance that the fracture toughness properties remain adequate throughout the service life of the plant is provided by a radiation surveillance program.

Deterministic analyses based on advanced fracture mechanics techniques demonstrated that detection of small cracks in the reactor coolant system piping would occur before the cracks could grow to critical or unstable sizes which could lead to large break areas, such as the double-ended guillotine break or its equivalent. These analyses demonstrated that the probability of breaks occurring in the reactor coolant system main loop piping is sufficiently low that these breaks need not be considered as a design basis for requiring installation of pipe whip restraints or jet impingement shields.

Safe operating heatup and cooldown limits are established according to Section III, ASME Boiler and Pressure Vessel Code, Appendix G 2000, Protection Against Nonductile Failure, issued in the Summer 1972 Addenda.

Changes in fracture toughness of the core region plates, weldments, and associated weld heat affected zone due to radiation damage are monitored by a surveillance program based on ASTM E-185, Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels. The evaluation of the radiation damage in this surveillance program is based on pre-irradiation testing by Charpy V-notch, tensile, and wedge opening loading specimens carried out during the lifetime of the reactor vessel. Specimens are irradiated in capsules located near the core mid-height and removed from the vessel at specified intervals. Further details are given in Chapter 4. All pressure containing components of the Reactor Coolant System were designed, fabricated, inspected and tested in conformance with the applicable codes. Further details are given in Section 4.1.7.

All components in the Reactor Coolant System were 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 operation. The number of thermal and loading cycles used for design purposes and the bases thereof are given in Table 4.1-8.

To provide the necessary high degree of integrity for the equipment in the Reactor Coolant System, the transient conditions selected for equipment fatigue evaluation were based on a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from normal operation, normal and abnormal load transients.

The criteria adopted for allowable stresses and stress intensities in vessels and piping subjected to normal loads plus seismic loads are defined in Appendix 4B. These criteria assure the integrity of the Reactor Coolant System under seismic loading. (Section 4.1)

Each of the materials used in the Reactor Coolant System was selected for the expected environment and service conditions. The major component materials are listed in Table 4.2-1.

All Reactor Coolant System materials which are exposed to the coolant are corrosion resistant. They consist of stainless steels and Inconel, and they were chosen for specific purposes at various locations within the system for their superior compatibility with reactor coolant.

The chemical composition of the reactor coolant is maintained within the specification given in Table 4.2-2. Reactor coolant chemistry is further discussed in Section 4.2.8.

The reactor vessel is the only component of the Reactor Coolant System exposed to a significant level of neutron irradiation and it is therefore the only component subject to material radiation damage effects.

The NDTT shift of the vessel material and welds, due to radiation damage effects is monitored by a radiation damage surveillance program which conforms with ASTM E-185 standards.

Reactor vessel design was 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 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. (Section 4.1.4)
The heatup and cooldown curves for the plant are based on the actual measured fracture toughness properties of the vessel materials. Maximum allowable pressures as a function of the rate of temperature change and allowable pressures as a function of the rate of temperature change, and the actual temperature, are established according to the methods given in Appendix G, Protection Against Non-Ductile Failure, published in the 1972 Summer Addenda of Section III the ASME Pressure Vessel and Boiler Code. The original RCS heatup and cooldown curves for up to 9.26 effective full power years (EFPYs) of reactor operation were developed from the analysis of capsule T. Subsequent analyses of capsules Y and Z did not require that changes be made to these curves until a new methodology to predict the effect of neutron radiation on reactor vessel materials was presented by Generic Letter 88-11. Consequently, new heatup and cooldown curves for 9 EFPYs based on the

analysis of capsule Z in accordance with the methodology of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." With increasing years of service, these curves were revised for 11, 13.3 and 16.2 EFPYs. These curves are given in the IP3 Technical Specifications. (Further details are given in Chapter 4.)

Inspection of Reactor Coolant Pressure Boundary (Criterion 32)

Criterion: Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leak-tight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Post-operational examinations as set forth in ASME Section XI are performed to the fullest extent practical at the required intervals.

The structural integrity of the Reactor Coolant System is maintained at the level required by the original acceptance standards throughout the life of the plant. Any evidence, as a result of the inspections required by 10 CFR 50.55a(g), that potential defect implications have initiated or enlarged shall be investigated, including evaluation of comparable areas of Reactor Coolant System.

Nondestructive test methods, personnel, equipment and records conform to the requirements of ASME B&PV Code, Section XI.

The use of conventional nondestructive, direct visual and remote visual test techniques can be applied to the inspection of all primary loop components except for the reactor vessel. The reactor vessel presents special problems because of the radiation levels and remote underwater accessibility to this component. Because of these limitations on access to the reactor vessel, several steps were incorporated into the design and manufacturing procedures in preparation for nondestructive test techniques. These are:

1) Shop ultrasonic examinations were performed on all thermally clad surfaces to an acceptance and repair standard to assure an adequate cladding bond to allow later ultrasonic testing of the base metal. Size of cladding bonding defect allowed is ¼" x ¾".

2) The design of the reactor vessel shell in the core area is a clean, uncluttered cylindrical surface to permit future positioning of test equipment without obstruction.

3) Selected areas of the reactor vessel were ultrasonic tested and mapped during the manufacturing stage to facilitate the in-service inspection program by establishing baselines for later testing.

   The areas selected for ultrasonic testing mapping are:
   a) Vessel flange radius, including the vessel flange to upper shell weld
   b) Middle shell course
   c) Lower shell course above the radial core supports
   d) Exterior surface of the closure head from the flange knuckle to the cooling shroud

e) Nozzle to upper shell weld
f) Middle shell to lower shell weld
g) Upper shell to middle shell weld

The preoperational ultrasonic testing of these areas was performed after hydrostatic testing of the reactor vessel. A qualified inspector employed by an insurance company authorized to write boiler and pressure vessel insurance certified all examinations.

Access to areas to be inspected is limited by the design of the plant, having been designed before the In-Service Inspection Code was developed. However, wherever possible, modifications have been made to components and insulation to facilitate the inspection program and allow easy access for examinations.

The data and results of the preoperational examinations serve as base line data for the in-service inspection program to determine if new indications have appeared or old indications have enlarged. (See Section 4.5.4)

Reactor Coolant Makeup (Criterion 33)

Criterion: A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

Makeup to the Reactor Coolant System is provided by the Chemical and Volume Control System.

Makeup for normal plant leakage is regulated by the reactor makeup control which is set by the operator to blend water from the Primary Water Storage Tank with concentrated boric acid to match the reactor coolant boron concentration. One primary makeup water pump and one boric acid transfer pump are normally aligned for operation on demand from the reactor makeup control system.

The “automatic makeup” mode of operation of the reactor makeup control provides boric acid solution preset to match the boron concentration in the Reactor Coolant System. The automatic makeup compensates for minor leakage of reactor coolant without causing significant changes in the coolant boron concentration. (Section 9.2)

Rupture of small cross section will cause expulsion of the coolant at a rate which can be accommodated by the charging pumps which would maintain an operational water level in the pressurizer permitting the operator to execute an orderly shutdown. The coolant which would be released to the Containment contains the fission products existing in it. (Section 14.3)

Should a break occur which is larger than the capacity of the charging pumps depressurization of the Reactor Coolant System causes fluid to flow to the Reactor Coolant System from the pressurizer resulting in a pressure and level decrease in the pressurizer. Reactor trip occurs when the pressurizer low pressure trip or overtemperature Δ T setpoint is reached. The Safety Injection System is actuated when the appropriate pressurizer low pressure setpoint is reached. Reactor Trip and Safety Injection System actuation are also initiated by a high containment pressure signal. The consequences of the accident are limited in two ways:

1) Reactor trip and borated water injection supplement void formation in causing rapid reduction of nuclear power to the residual power level corresponding first to delayed fission and ultimately to fission product decay.

2) Injection of borated water ensures sufficient flooding of the core to prevent excessive clad temperature.

A high degree of functional reliability was assured in the Chemical and Volume Control System by providing standby components where performance is vital to safety and by assuring fail-safe response to the most probable mode of failure. Special provisions include duplicate heat tracing with alarm protection of lines, valves, and components normally containing concentrated boric acid.

The system has three high pressure charging pumps, each capable of supplying the normal reactor coolant pump seal and makeup flow.

The electrical equipment of the Chemical and Volume Control System is arranged so that multiple items receive their power from various 480 volt buses. Each of the three charging pumps are powered from separate 480 volt buses. The two boric acid transfer pumps are also powered from separate 480 volt buses. One charging pump and one boric acid transfer pump are capable of meeting cold shutdown requirements shortly after full-power operation. In cases of loss of AC power, a charging pump and a boric acid transfer pump can be placed on the emergency diesels if necessary. (Section 9.2)

Three charging pumps inject coolant into the Reactor Coolant System. The pumps are the variable speed positive displacement type, and all parts in contact with the reactor coolant were fabricated of austenitic stainless steel or other material of adequate corrosion resistance. These pumps have mechanical packing followed by a leakoff to collect reactor coolant before it can leak to the outside atmosphere. Pump leakage is piped to the drain header for disposal. The pump design prevents lubricating oil from contaminating the charging flow, and the integral discharge valves act as check valves.

Each pump was designed to provide the normal charging flow and the reactor coolant pump seal water supply during normal seal leakage. Each pump was designed to provide rated flow against a pressure equal to the sum of Reactor Coolant System normal maximum pressure (existing when the pressurizer power operated relief valve is operating) and the piping, valve and equipment pressure losses at the design charging flows.

At least two separate and independent flow paths are available for reactor coolant boration; i.e., either the charging line or the reactor coolant line. The malfunction or failure of one component will not result in the inability to borate the Reactor Coolant System. An alternate supply path is always available for emergency boration of the reactor coolant. As backup to the boration system, the operator can align the Refueling Water Storage Tank outlet to the suction of the charging pumps.

On loss of seal injection water to the reactor coolant pump seals, seal water flow may be re-established by manually starting a standby charging pump. Even if the seal water injection flow is not re-established, the plant can be operated indefinitely since the thermal barrier cooler has sufficient capacity to cool the reactor coolant flow which would pass through the thermal barrier cooler and seal leakoff from the pump volute.

Boration during normal operation to compensate for power changes will be indicated to the operator from two sources: (a) the control rod movement and (b) the flow indicators in the boric acid transfer pump discharge line. When the emergency boration path is used, two indications are available to the operator. The charging line flow indicator will indicate boric acid flow since the charging pump suction is aligned to the boric acid transfer pump suction for this mode of operation. The change in boric acid tank level is another indication of boric acid injection.

<u>Residual Heat Removal (Criterion 34)</u>

Criterion: A system to remove residual heat shall be provided. The safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capability shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available), the system safety function can be accomplished, assuming a single failure.

Systems have been provided to remove residual heat from the reactor core and to cool down the reactor as follows:

During the first phase of cooldown, the reactor coolant system temperature is reduced by transferring decay heat and sensible heat to the Steam and Power Conversion System. From hot standby conditions, residual heat removal requirements are normally satisfied by steam bypass to the condensers (see Section 10.2.1). Core decay heat can be continuously dissipated via the steam bypass to the condenser as feedwater in the steam generator is converted to steam by heat absorption. Normally, the capability to feed the steam generator is provided by operation of the turbine cycle feedwater system. In the unlikely event of complete loss of electrical power to the station, decay heat removal would be assured by the availability of the safety grade Auxiliary Feedwater System and steam discharge to the atmosphere via the main steam safety valves and atmospheric relief valves. The steam-driven auxiliary feedwater pump can supply sufficient feedwater for removal of decay heat from the core and reduce the reactor coolant system temperature.

The above systems/equipment are used for residual heat removal until the reactor coolant system temperature and pressure are reduced to between 250 ^{0} and 350 ^{0} F and 400 to 425 psig. At this point, the residual heat removal loop of the Safety Injection System is placed into service to remove the decay heat and sensible heat from the Reactor Coolant System. Normally, two pumps and two heat exchangers are used for residual heat removal by taking suction from the Reactor Coolant System and discharging through the heat exchangers back into the Reactor Coolant System. All active loop components which are relied upon to perform their function are redundant. Only one pump and one heat exchanger are needed to accomplish the decay heat removal and reduce the reactor coolant system temperature, but this will occur at a reduced rate. Cooling water for the residual heat removal heat exchangers is supplied by the Component Cooling Water System. The Component Cooling Water System (CCWS), described in Section 9.3, is a closed system which serves as an intermediate system between the reactor coolant and the Service Water System. This double barrier arrangement reduces the probability of leakage of radioactive reactor coolant to the Service Water System. Radiation monitors are provided in the component cooling water system to detect reactor coolant leakage. Active loop components which are relied upon to perform the cooling function are redundant. Three CCWS pumps exist and two are normally initialized to remove heat during plant shutdown. Two CCWS heat exchangers are normally utilized to remove the residual heat during plant shutdown. If one of the pumps or one of the heat exchangers is not operable, safe shutdown is not affected. However, the cooldown time is extended. Components of the CCWS required for residual heat removal can be supplied by the onsite emergency power system.

Cooling water for the CCWS is provided by the Service Water System described in Section 9.6.1.

Sufficient redundancy of active and passive components is provided to ensure that cooling is maintained for vital heat loads for short and long periods in accordance with the single failure criteria. The system consists of two independent supply headers with each header being supplied by three pumps. Either of the two headers can be used to supply cooling for the essential loads. Either one of the two sets of three service water pumps can be powered by the emergency diesels. (See Section 9.6.1)

Emergency Core Cooling (Criteria 35)

Criterion: A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

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 system assures that the core will remain intact and in place with its essential heat transfer geometry preserved following a rupture in the Reactor Coolant System. It also assures that the extent of metal-water reaction is limited such that the amount of hydrogen generated from this source, in combination with that from other sources, is tolerable in the Containment. This capability is provided during the simultaneous occurrence of a Design Basis Earthquake. 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 primary function of the Emergency Core Cooling System for the ruptures described above, is to remove the stored and fission product decay heat from the core such that fuel damage, to the extent that would impair effective cooling of the core, is prevented. This implies that the core remain intact and in place, with its essential heat transfer geometry preserved. To assure effective cooling of the core, limits on peak clad temperature and local metal-water reaction will not be exceeded. It has been demonstrated in the Westinghouse Rod Burst Program that for conditions within the area of safe operation, fuel rod integrity is maintained.

To limit the production of hydrogen in the Containment, the overall metal water reaction is limited to 1%.

In evaluating the ECCS performance, consideration is given to core geometry distortion caused by swelling or fuel rod bursting.

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 intact.

Redundancy and segregation of instrumentation and components was incorporated to assure 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 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 of a loss of coolant, the system is tolerant of a loss of any part of the flow path since back up alternative flow path capability is provided.

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.

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 where the core has been uncovered for postulated large area ruptures. The result of this performance is to sufficiently limit any increase in clad temperature below a value where emergency core cooling objectives are met. (Section 6.2.1)

With minimum onsite emergency power available (two-of-three diesel generators) the emergency core cooling equipment available is two out of three safety injection pumps, one out of two residual heat pumps, and three out of four accumulators for a cold leg break and four accumulators for a hot leg break. With these systems, the calculated maximum fuel cladding temperature is limited to a temperature less than that which meets the emergency core cooling design objectives for all break sizes up to and including the double-ended severance of the reactor coolant pipe. (Section 14.3.1)

For large area ruptures analyzed, the clad temperatures are turned around by the accumulator injection. The active pumping components serve only to complete the refill started by the accumulators. Either two safety injection pumps or one residual heat removal pump provides sufficient addition of water to continue the reduction of clad temperature initially caused by the accumulator injection. (Section 6.2)

<u>Inspection of Emergency Core Cooling System (Criterion 36)</u>

Criterion: The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

Design provisions were made to the extent practicable 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 non-destructive test inspection where such techniques are desirable and appropriate. (Section 6.2)

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.

<u>Testing of Emergency Core Cooling System (Criterion 37)</u>

Criterion: The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practicable, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

The design of the Safety Injection System provides for capability to initially test, to the extent practicable, the full operational sequence up to the design conditions to demonstrate the state of readiness and capability of the system. (Section 6.2)

The design provides for periodic testing of active components 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 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 and can be tested periodically. All remote operated valves are exercised and actuation circuits are tested during routine plant maintenance.

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.

In addition, to the extent practicable, the critical parts of the reactor vessel internals, pipes, valves and pumps are inspected visually or by boroscopic examination for erosion, corrosion, and vibration wear evidence, and by non-destructive test inspection where such techniques are desirable and appropriate.

Initial functional tests of the core cooling portion of the Safety Injection System were conducted before initial plant startup. These tests were performed following the flushing and hydrostatic testing of the system and with the Reactor Coolant System cold. The Safety Injection System valving was set initially to simulate the system alignment during plant power operation.

These functional tests were divided into two parts:

a) Demonstrate the proper function of instrumentation and actuation circuits, confirm valve operating times, confirm pump motor starting times, and demonstrate the proper automatic sequencing of load addition to the emergency diesels.

The tests were repeated for the various modes of operation needed to demonstrate performance at partial effectiveness, i.e., 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 tests were performed without delivery of water to the Reactor Coolant System, but included starting of all pumping equipment involved in each test.

b) Demonstrate the proper delivery rates of injection water to the Reactor Coolant System.

To initiate the first part of 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 the low water level and low-pressure signals from the pressurizer instrumentation. Simultaneously, the breakers supplying outside power to the 480 volt 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 started automatically following the prescribed diesel loading sequence. The valves operated automatically to align the flow path for injection into the Reactor Coolant System.

The second portion of the test was initiated by manually starting individual pumps on mini-flow and manually opening the appropriate isolation valves to deliver water to the Reactor Coolant System. Data was taken to verify proper pump performance and flow delivery rates.

Pre-operational 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. NPSH was established at design flow 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 hydrostatically tested at 150 percent of design pressure.

The service water and component cooling water pumps were thoroughly tested prior to initial operation.

Periodic testing of the Safety Injection System components and all necessary support systems at power is a portion of Entergy's test program. The safety injection and residual heat removal pumps are to be tested in accordance with the Indian Point 3 Inservice Testing Program, to check the operation of the starting circuits, verify the pumps are in satisfactory running order, and verification is made that required discharge head is attained. No inflow to the Reactor Coolant System will occur whenever the reactor coolant pressure is above 1500 psi. 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 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.

The operation of the remote stop valves in the accumulator tank discharge line may be tested by opening the remote test valves just downstream of the stop valve. Flow through the test line can be observed on instruments and the opening and closing of the discharge line stop valves can be sensed on this instrumentation. Test circuits are provided to periodically examine the leakage back through the check valves and to ascertain that these valves seat whenever the reactor system pressure is raised.

This test is routinely performed when the reactor is being returned to power after an outage and the reactor pressure is raised above the accumulator pressure. If leakage through a check valve should become excessive, the isolation valve would be closed. (The safety injection actuation signal will cause this valve to open should it be in the closed position at the time of a Loss-of-Coolant Accident.) The performance of the check valves has been carefully studied and it is highly unlikely that the accumulator lines would have to be closed because of leakage.

The recirculation pumps are normally in a dry sump. Minimum flow testing of these pumps is performed during refueling operation by filling the recirculation sump and opening the miniflow valve on the discharge of the pump and directing the flow back to the sump. Those service water and component cooling pumps which are not running during normal operation may be tested by alternating them with the operating pumps.

The content of the accumulators and the Refueling Water Storage Tank are sampled periodically to assure that the required boron concentration is present.

System testing is conducted during plant shutdown to demonstrate proper automatic operation of the Safety Injection System. An actual or simulated safety injection test signal is applied to initiate automatic action and verification is made that the components receive the safety injection signal in the proper sequence. The Safety Injection and Residual Heat Removal pumps are blocked from starting. Isolation valves in the injection lines are blocked closed so that flow is not introduced into the Reactor Coolant System. The system test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry. The test is considered satisfactory if control board indication and visual observations indicate all components have operated and sequenced properly. A complete system test cannot be performed when the reactor is operating because a safety injection signal would cause a reactor trip. The method of assuring complete operability of the Safety Injection System is to combine the system test performed during plant shutdown with more frequent component tests, which can be performed during reactor operation.

The accumulators and the safety injection piping up to the final isolation valve are maintained sufficiently filled of borated water while the plant is in operation to ensure the system remains operable and performs properly. The accumulators and the high head injection lines are refilled with borated water as required by using the safety injection pumps to recirculate refueling water through the injection lines. A small test line is provided for this purpose in each injection header.

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 eight-switch sequence for recirculation operation is tested following the above injection phase test to demonstrate proper sequencing of valves and pumps. The recirculation pumps are blocked from start 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 utilized during external recirculation (safety injection and residual heat removal pumps) in turn at near shutoff head conditions and checking the discharge and recirculation test lines. The suction lines are tested by running by 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 packing, pump seals, leak-off connections or other potential points of leakage are 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 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.

<u>Containment Heat Removal (Criterion 38)</u>

Criterion: A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptable low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

The Reactor Containment heat removal function is accomplished by two independent, full capacity systems.

The Containment Spray System, as described in Section 6.3, sprays cool water into the containment atmosphere, when appropriate, in the event of a Loss-of-Coolant Accident and thereby ensures that the containment pressure does not exceed the design value.

The Containment Air Recirculation Cooling and Filtration System, described in Section 6.4, was designed to recirculate and cool the containment atmosphere in the event of a Loss-of-Coolant Accident to limit containment pressure within the design value.

In the event of a Design Basis Accident, any one of the following combinations will provide sufficient cooling to reduce containment pressure at rate consistent with limiting offsite doses to acceptable values: (1) five fan cooler units; (2) two containment spray pumps; or (3) three fan cooler units and one spray pump. For design basis accidents in which failure of any single diesel generator is assumed, the resulting equipment configuration is adequate to perform containment cooling and iodine removal requirements. (See FSAR Section 14.3.6 for details.) Capability of the containment heat removal systems to reduce the containment pressure and temperature levels is discussed in Section 14.3.6. The containment cooling system capability assumed in the analysis was one of two available containment spray subsystems and three of five available containment fan cooler units. This is the minimum equipment available considering the single failure criteria in the emergency power system, the spray system and the fan cooler system.

<u>Inspection of Containment Heat Removal System (Criterion 39)</u>

Criterion: The containment heat removal system shall be designed to permit appropriate periodic inspection of important components such as torus, sumps, spray nozzles and piping to assure the integrity and capability of the system.

Where practicable, all active components and passive components of the Containment Spray System are inspected periodically to determine 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 are inspected for leaks. Design provisions for inspection of the safety injection components which also function as part of the Containment Spray System, are described in Section 6.2.5.

Provisions were made to the extent practical to facilitate access for periodic visual inspection of all important components of the Containment Air Recirculation Cooling and Filtration System.

The Containment Air Recirculation Cooling and Filtration System was designed to the extent practicable so that the components can be tested periodically, and after any component maintenance, for operability and functional performance.

The air recirculation and cooling units, and the service water pumps, which supply the cooling units, are in operation on an essentially continuous schedule during plant operation, and no additional periodic tests are required.

Means are provided to initially test to the extent practical the full operational sequences of the air recirculation system including transfer to alternate power sources.

Testing of Containment Heat Removal System (Criterion 40)

Criterion: The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure:

1) The structural and leaktight integrity of its components.

2) The operability and performance of the active components of the system.

3) The operability of the system as a whole, and, under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

The Containment Spray System active components are tested periodically and also after maintenance is performed. Permanent test lines for the containment spray loops are provided so that system components may be tested. Air test lines for checking that the spray nozzles are unobstructed are provided. Capability was provided to test initially, to the extent practicable, the operational startup sequence of the Containment Spray System including the transfer to alternate power sources. Periodic testing is conducted to verify proper sequencing of valves and pumps on initiation of the containment spray signal and to demonstrate the proper operation of all remotely operated valves and the spray pumps. (Section 6.3)

Surveillance testing of the Containment Spray System is described in the Technical Specifications. System tests are performed at least once per 24 months. A test is considered satisfactory if visual observations indicate all components have operated satisfactorily. The pressure containing components are inspected for leaks from pump seals, valve packing, flanged joints and safety valves. During operational testing of containment spray pumps, the portions of the system subjected to pump pressure are inspected for leaks.

The Containment Air Recirculation Cooling and Filtration System was designed to the extent practicable that components can be tested periodically for both operability and functional performance, and after any component maintenance. The air recirculation and cooling units and the service water pumps, which supply the cooling units, are in operation on an essentially continuous basis during plant operation and, therefore, no additional periodic tests are necessary. Means were provided to test initially, to the extent practicable, the full operational sequence of the system, including transfer to alternative power sources.

The containment air cooling units are supplied with cooling water from the Service Water System, as described in Section 9.6.1. One of the three nuclear service water pumps is required for operation during normal containment cooling. For emergency operation, two independent, full flow isolation valves open automatically in the event of an engineered safeguards actuation signal.

Containment Atmosphere Cleanup (Criterion 41)

Criterion: Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

The Containment Air Recirculation Cooling and Filtration System described in Section 6.4, was designed to remove fission products from the containment atmosphere should they be released in the event of an accident. The filtration capacity of the system is sufficient to reduce the concentration of fission products in the containment atmosphere following a loss of reactor coolant to levels ensuring that the 2 hour and 30 day radiation (TEDE) doses will be limited to within the guidelines of 10 CFR 100. Details of the site boundary dose calculation are given in Section 14.3.5. The Containment Spray System also serves as a removal mechanism for fission products postulated to be dispersed in the containment atmosphere. Suitable redundancy has been provided for these systems. In the event of a Design Basis Accident, three charcoal filters and their associated recirculation fans (five recirculation fans and filters have been provided), along with one of two containment spray pumps, will reduce air-borne organic and molecular iodine activities sufficiently to limit offsite doses to acceptable values. These constitute the minimum safeguards for iodine removal and are capable of being operated on emergency power with one diesel generator inoperable. A single failure analysis has been made on all active components of the Containment Air Recirculation and Filtration System to show that the failure of any single active component will not prevent fulfilling the design function. The analysis is summarized in Table 6.4.1. A similar analysis for the Containment Spray System is summarized in Table 6.3-4.

Two full rated recombination systems, as described in Section 6.8, are provided to control the hydrogen evolved in the containment following a Loss-of-Coolant Accident. Either system is capable of preventing the hydrogen concentration from exceeding 2% by volume within the Containment (Section 14.3.7). Power supplies for the recombiners are separate so that loss of one power supply will not affect the remaining system. Sufficient system redundancy and independence are provided such that no single active or passive component failure can negate the minimum requirements of operation.

Inspection of Containment Atmosphere Cleanup Systems (Criterion 42)

Criterion: The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

Access is available for visual inspection of the Containment Air Recirculation Cooling and Filtration System components including fans, cooling coils, dampers, filter units and ductwork. Provisions were made for ready removal of the filters for inspections and testing. (Section 6.4)

Design provisions were made to the extent practicable to facilitate access for periodic visual inspections of all important components of the Containment Spray System.

Periodic operating checks of the Hydrogen Recombiner System are described in the FSAR. The portion of the system located outside the Containment should be readily accessible for inspection at any time, while the Containment portion would be accessible for inspection during reactor shutdown.

Testing of Containment Atmosphere Cleanup Systems (Criterion 43)

Criterion: The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

The Containment Air Recirculation Cooling and Filtration system was designed to the extent practicable so that components can be tested periodically and after maintenance for operability and functional performance. The air recirculation and cooling units are in operation on an essentially continuous schedule during plant operation and no additional periodic tests are required. (Section 6.4)

The closed dampers, which allow for charcoal filter bypass during normal operation, can be periodically tested in a non-operating unit by activating controls and verifying deflection by instruments in the Control Room. Representative sample elements in each of the activated carbon filter plenums are removed periodically during shutdowns and tested to verify their continued efficiency. After reinstallation, the filter assemblies are tested in place in accordance with the Technical Specifications to determine integrity of the flow path.

Means were provided to test initially under conditions as close to design as was practicable the full operational sequence that would bring the system into action, including transfer to the emergency diesel generator power source. Surveillance testing of the containment air filtration system is covered in the Technical Specifications.

One of the design bases for the Hydrogen Recombination System was that the system shall be testable during normal operating conditions of the plant. Operating checks of the system are performed periodically, as discussed in the FSAR.

Testing of the Containment Spray System is addressed in the discussion for Criterion 40.

<u>Cooling Water (Criterion 44)</u>

Criterion: A system to transfer heat from structures, systems, and components important to safety to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

A closed-loop Component Cooling Water System and a once-through Service Water System are provided to transfer heat loads from structures, systems and components important to safety to an ultimate heat sink. The component cooling system transfers heat loads to the Service Water System via component cooling heat exchangers. The Service Water System takes water from the Hudson River and supplies cooling water for the Component Cooling Water

System and other various heat loads from structures, systems and components important to safety. The Hudson River is used as the ultimate heat sink.

Six identical service water pumps and three backup service water pumps are provided to supply cooling water to the Service Water System. Sufficient redundancy of active and passive components is provided to ensure that cooling is maintained to vital loads for short and long periods in accordance with the single failure criteria. Two independent service water headers are provided with each being supplied by three pumps. Either of the two supply headers can be used to supply the essential loads, with the other feeding the non-essential loads. Leak detection in the Service Water System for both inside and outside of the Containment can be accomplished by the methods described in Section 6.7.1.

The only major components in the Service Water System located inside the Containment are the fan cooling units. Each unit is provided with valves so that it may be isolated for leak testing. Should there be a failure in the piping or valves at the header supplying water to the fan cooling units, one of the two series valves in the center of the header can be closed and service will continue on the side of the header opposite the failure. Likewise, operation of at least one component cooling heat exchanger (located outside the containment) is assured despite the postulated failure of any single active or passive component in the system from the service water pumps to the heat exchangers themselves. Refer to Section 9.6.1 for details.

A single failure analysis of nuclear service water pumps and discharge line isolation valve is shown on Table 6.4-1.

The Component Cooling Water System was designed to remove residual and sensible heat from the Reactor Coolant System during plant shutdown, cool the letdown flow during power operation, and to provide cooling to dissipate waste heat from various primary plant components. Active loop components in the system are redundant. The Component Cooling Water System is provided with two main headers. Isolation valves are furnished to allow each loop to be isolated and operated as an independent component cooling loop. The loop design provides for detection of radioactivity entering the loop from a reactor coolant source and also provide for isolation. (Section 9.3)

Water leakage from the component cooling loop can be detected by a falling level in the component cooling surge tanks. The component which is leaking can be located by sequential isolation or inspection of equipment in the loop. A failure analysis of pumps, heat exchangers and valves is presented in Table 9.3-5.

Other means of leak detection in the Component Cooling Water System are covered in Section 6.7.1.2.

Three components cooling pumps and two heat exchangers are provided for the component cooling system. During normal full power operation, two pumps and one heat exchanger will accommodate the heat removal loads. The third pump provides 50% backup and the second heat exchanger provides 100% backup during normal operation. All three pumps and two heat exchangers are to be utilized to remove the residual and sensible heat during normal plant shutdown.

The cooling water system pumps can be operated on onsite electric power or offsite power to accomplish the system safety function. In addition, the onsite sources of emergency power from three diesel generator sets are provided to the safety related pumps required to perform the system safety function upon loss of the normal power supply. (Section 8.2)

<u>Inspection of Cooling Water System (Criterion 45)</u>

Criterion: The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

The cooling water system operates when the reactor is in operation, and is continuously monitored for satisfactory performance which indicates the integrity and capability of the system.

Indian Point 3 has a program for periodic inspection of pipe welds and components for leakage. The program has been reviewed (April 1-3, 1980) and audited by NRC and subsequently revised to incorporate NRC's recommendations and comments.

Testing of Cooling Water System (Criterion 46)

Criterion: The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

The active components of the Auxiliary Coolant System are in either continuous or intermittent use during normal plant operation. Their satisfactory functional performance is thus assured. (Section 9.3)

The residual heat removal loop is used to remove residual and sensible heat from reactor coolant during normal shutdown. Provisions for periodic hydrostatic testing to applicable code test pressure were included in the design.

A comprehensive program of plant testing has been formulated and implemented 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, 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.

The initial tests of individual components and the integrated tests of the system as a whole complement each other to assure performance of the system as designed and to ensure proper operation of the actuation circuitry. (Section 6.1.1)

The external recirculation flow paths are hydrotested during periodic retests at the operating pressure. This is accomplished by running each pump which could be utilized during external recirculation (safety injection and residual heat removal pumps) in turn at near shutoff head conditions 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 packing, pump seals, leakoff connections or other potential points of leakage are 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 put through an operating cycle before a final leakage examination is made.

During the above test, all system joints, valve packing, pump seals, leakoff connections or other potential points of leakage are 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 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 feature components. The recirculation line to the residual heat removal pump is capable of being hydrotested during plant shutdown and it is also tested at the time of periodic retests of the Containment. (Section 6.7.2.5)

1.3.5 Reactor Containment (Criteria 50 to 57)

Containment Design Basis (Criterion 50)

Criterion: The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources that have not been included in the determination of the peak conditions, such as energy in steam generators and, as required by 50.44, energy from metal, water and other chemical reactions that may result from degradation, but not total failure, of emergency core cooling functioning; (2) the limited experience and experimental data available for defining accident phenomena and containment responses; and (3) the conservatism of the calculational model and input parameters.

The containment structure and all penetrations were designed to withstand, within design limits, the combined loadings of the design basis accident and design seismic conditions.

All piping systems which penetrate the vapor barrier are anchored at the liner. The penetrations for the main steam, feedwater, blowdown and sample lines were designed so that the penetration is stronger than the piping system and that the vapor barrier is not breeched due to a hypothesized pipe rupture combined, for the case of the steam line, with the coincident internal pressure. The pipe capacity in flexure was assumed to be limited to the plastic moment capacity based upon the ultimate strength of the pipe material. All lines with the exception of small bore line, 2” and smaller connected to the primary coolant system that penetrate the vapor barrier are also anchored in the secondary shield walls (i.e., walls surrounding the steam generators and reactor coolant pumps) and are provided with at least one valve between the anchor and the Reactor Coolant System. These anchors were 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 basis seismic conditions.

Appendix 4B includes 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 (as required by 10 CFR 50.44), structural heat sinks and the operation of the engineered safeguards, the latter utilizing only the emergency electric power supply. (Section 5.1)

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

a) In calculating the containment pressure, rupture sizes up to and including a double-ended severance of reactor coolant pipe were considered. In considering post-accident pressure effects, various malfunctions of the emergency systems were evaluated. Contingent mechanical or electrical failures were assumed to disable the most limiting of the diesel generators, thus disabling one or 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.

b) 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 below:

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 onsite diesel generators (the equipment configuration varies with the presumed failed generator regardless of which generator fails, safety requirements are met), 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 selection and use of containment materials comply with the applicable codes and standards tabulated in Section 5.1.1.

The concrete containment is not susceptible to a low temperature brittle fracture.

Systems relied upon to operate under post-accident 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 were designed for the maximum potential earthquake ground motion at the site combined with the simultaneous loads of the design basis accident as follows:

1) The liner was designed to ensure that no average strains greater than the strain at the guaranteed yield point occur at the factored loads

2) The mild steel reinforcement was generally 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 pressure retaining components of the Containment subject to deterioration or corrosion in service are provided with appropriate protective means or devices (e.g., protective coatings).

The containment structure was designed to safely withstand several conditions of loading and their credible combinations. The limiting extreme conditions are:

a) Occurrence of gross failure of the Reactor Coolant System which creates a high pressure and temperature condition within the Containment.

b) Coincident failure of the Reactor Coolant System with an earthquake or wind.

The design pressure and temperature of the Containment is equal to 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 severance of a reactor coolant pipe. Energy contribution from the steam system was 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 were designed to withstand the blowdown forces associated with the sudden severance of the reactor coolant piping so that the coincidental rupture of the secondary side steam system is not considered credible.

The design pressure and temperature on the containment structure are those created by the hypothetical Loss-of-Coolant Accident. The Reactor Coolant System contains approximately 512,000 lbs of coolant at a weight average enthalpy of 595 Btu/lb for a total energy of 304,000,000 Btu. In a hypothetical accident, this water is released through a double-ended break in the largest reactor coolant pipe, causing a rapid pressure rise in the Containment. The reactor coolant pipe used in the accident is the 29-in. ID section, because rupture of the 31-in ID section requires that the blowdown go through both the 29-in and the 27.5-in ID pipes and would, therefore, result in a less severe transient.

Additional energy release was considered from the following sources:

a) Stored heat in the reactor core
b) Stored heat in the reactor vessel piping and other reactor coolant system components
c) Residual heat production
d) Limited metal-water reaction energy and resulting hydrogen-oxygen reaction energy

The following loadings were considered in the design of the Containment in addition to the pressure and temperature conditions described above:

a) Structure dead load
b) Live loads
c) Equipment loads

d) Internal test pressure
e) Earthquake
f) Wind (Tornado)

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 containment mat.

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. The concrete walls and floor are lined with 0.25-inch thick stainless steel plate. The linings provide a leakproof membrane that is resistant to abrasion and damage during fuel handling operations. (Appendix 5A)

Fracture Prevention of Containment Pressure Boundary (Criterion 51)

Criterion: The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing and postulated accident conditions (1) its ferritic materials behave in a non-brittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual steady-state, and transient stresses, and (3) size of flaws.

The selection and use of containment materials comply with the applicable codes and standards tabulated in Section 5.1.1.

The concrete containment is not susceptible to a low temperature brittle fracture.

Acceptance criteria for containment pressure boundary material used considered examination for inclusion content and consideration of the effect of inherent flaws on mechanical properties.

The containment liner is enclosed with 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^{O} F. This includes both hot operating and cold shutdown conditions. The minimum service metal temperature of the containment liner is well above the NDT temperature +30^{O} F for the liner material. The equipment hatch, penetration sleeves and personnel lock meet the Charpy V-notch impact valves for a minimum of 15 ft-lbs at -50 ^{O} F.

The pressure retaining components of the containment structure were 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 was designed to ensure that no average strains greater than the strains at the guaranteed yield point occur at the factored loads

2) The mild steel reinforcement was generally 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 pressure retaining components of the Containment subject to deterioration or corrosion in service are provided with appropriate protective means or devices (e.g., protective coatings).

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 minimum tensile strength of 60,000 psi with an elongation of 22 percent in an 8-inch gauge length at failure.

The liner is 0.25-inch thick at the bottom, 0.5-inch thick in the first three courses and 0.375-inch thick for the remaining portion of cylindrical walls except that it is 0.75-inch thick at penetrations and 0.5-inch thick in the dome. The liner material has been tested to assure an NDT temperature more than 30F 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 the liner anchors was made prior to pouring concrete.

The structural design of the containment structure was based upon limiting load factors which are used as the ratio by which loads will be multiplied for design purposes to assure that the loading deformation behavior of the structure is one of elastic, tolerable strain behavior. The load factor approach was used in this design as a means of making a rational evaluation of the isolated factors which must be considered in assuring an adequate safety margin for the structure. This approach permitted 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, this approach placed 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 presented in Appendix 5A.

The load factors utilized were based upon the load factor concept employed in Part IV-B, “Structural Analysis and Proportioning of Members Strength Design” of ACI 318-63. Because of the refinement of the analysis and the restrictions on construction procedures, the load factors in the design primarily provide for a safety margin on the load assumptions. Specific combined load equations used in design are presented in Appendix 5A.

All structural components were designed to have a capacity required by the most severe loading combination.

The design included consideration of both primary and secondary stresses. The load capacity in structural members was based on the ultimate strength values presented in Part IV-B of ACI-318 as reduced by the capacity reduction factor “ϕ” 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 undercapacity.

For the liner steel the factor "ϕ" is 0.95 for tension. For compression and shear, the primary membrane liner stress was maintained below 0.95 yield and elastic stability has been assured as a function of liner anchorage requirements.

The liner was designed to assure that no strains greater than the strain at the guaranteed yield point will occur at the factored loads except in regions of local stress concentrations or stresses due to secondary load effects in which case the liner strain is limited to 0.5 percent. Sufficient anchorage is provided to assure elastic stability of the liner. The basic design concept utilizing stud anchorage of the liner plate to the concrete structure assures stud failure due to shear, tension or bending stress without the stud connection causing failure or tear of the liner plate. The studs in the 0.5-inch plate were installed on a 24" horizontal and 28" vertical grid and, in the 0.375 inch plate, on a 24" horizontal and 14" vertical grid. The design considered the possibility of daily stress reversals due to ambient temperature changes for the life of the plant and the fatigue limit of the studs exceeds the design requirements. (Section 5.1.2)

Service temperatures during operation, maintenance and testing are less severe than those accompanying the containment design basis conditions and therefore will not induce brittle fracture of the containment liner.

Capability for Containment Leakage Rate Testing (Criterion 52)

Criterion: The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

After completion of the containment structure and installation of all penetrations and weld channels, integrated leakage rate tests were performed prior to initial plant operations at a reduced test pressure and at the calculated peak accident pressure to establish the respective measured leakage rates and to verify that the leakage rate at the peak accident conditions was no greater than 0.075 percent by weight per day of the containment steam-air atmosphere at the calculated peak accident conditions. Leak rate testing of the containment is performed in accordance with Technical Specification 5.5.15. The leakage rate program is in accordance with the guidance contained in Regulatory Guide 1.163, except as noted in the Technical Specifications.

The peak accident pressure integrated leakage rate test is conducted at periodic intervals during the life of the plant, and also as appropriate in the event of major maintenance or major plant modifications. A leak rate test at the peak accident pressure using the same test 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. (Section 5.1.7)

Penetrations were 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 utilizes a supply of clean, dry compressed air, which places the penetrations under an internal pressure above the peak calculated accident pressure.

A permanently piped monitoring system continuously measures 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.

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.

<u>Provisions for Containment Testing and Inspection (Criterion 53)</u>

Criterion: The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

Penetrations were 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 utilizes a supply of clean, dry compressed air which 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 (Section 5.1.7).

The containment is completely closed whenever the primary system temperature is above 200 F, except as required for brief periods necessary to relieve the Containment to keep the pressure below a reasonable level (1- 2 psig) or to purge containment in preparation for containment entry.

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 for special maintenance on periodic inspections. Access at power is normally restricted to the areas external to the reactor equipment compartment primarily for inspection and maintenance of the air recirculation equipment, incore instrumentation chamber drives, and instrumentation calibration.

The primary reactor shield was designed so that access to the primary equipment is limited by the activity of the primary system equipment and not the reactor. (Section 5.1.4)

All penetrations, the Personnel Air Lock and the Equipment Hatch were designed with double seals, which are normally pressurized at a minimum pressure greater than the calculated peak accident pressure. Individual testing at 115% 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 up to the penetration pressurization system. The space can be pressurized to above accident 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 locating leaks with a Freon sniffer. Most liner welds are also continuously pressurized during power operation at a minimum pressure greater than the calculated peak accident pressure. Liner welds that are not pressurized during power operation are those welds associated with disconnected sections of the Weld Channel Pressurization System. The integrity of the welds associated with any disconnected portions of the Weld Channel Pressurization System is verified by integrated leak rate testing.

A 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 leak existed.

After it was assured that there were no defects remaining from construction, a sensitive leak rate test was conducted. The sensitive leak rate test included only the volume of the weld channels and double penetrations. This test was more sensitive than the integrated leakage rate test, as the instrumentation used permits a direct measurement of leakage from the pressurized zones. The sensitive leak rate test was conducted with the penetrations and weld channels at a pressure greater than the calculated peak accident pressure 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% of the containment free volume per day.

The double penetrations and most weld seam channels installed on the inside of the liner in the Containment are continuously pressurized to provide a continuous, sensitive and accurate means of monitoring their status with respect to leakage.

Periodic peak pressure containment integrated leakage rate tests are performed in accordance with the Technical Specifications. Peak pressure tests are conducted as appropriate in the event of major maintenance plant modifications.

The Containment Penetration and Weld Channel Pressurization System components located outside the Containment can be visually inspected at any time. Components inside the Containment are 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. (Section 6.6.5)

No special testing of system operation or components is necessary to assure that the penetrations and liner weld channels are pressurized above containment calculated peak accident response pressure since the system is in continuous operation during all reactor operations.

Should one zone indicate a leak during operation, the specific penetration or weld channel containing the leak is 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 be detected before leakage limits are exceeded. Remedial action is then taken before the limit is reached.

To provide for testing the larger penetrations, branch pressurizing lines are installed from one of the zones to:

a) The double-gasketed space on each hatch of the Personnel Air Lock
b) The double-gasketed space at the Equipment Hatch Flange
c) The pressurized zones in the spent fuel transfer tube
d) The spaces between the two butterfly valves in the purge supply and exhaust ducts
e) The two spaces between the three butterfly valves in the containment pressure relief line
f) The spaces between double containment isolation valves in the steam jet air ejector return line to containment and in the containment radiation monitor inlet outlet lines

The makeup air flow to the penetrations and liner weld joint channels during normal operation is only an indication of the potential leakage from the containment. It does indicate the leakage from the pressurization system, and degree of accuracy is increased when correlated with the results of the full scale containment leak rate tests. The criteria for selection of operating limits for air consumption of the pressurization system are based upon the integrated containment leak rate test acceptance criterion 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 preoperational integrated containment leak rate tests. Unexplained increases from the consumption rate require routine investigation and location of the point of leakage.

2) The upper limit for long-term air consumption for the pressurization system is 0.2% of the containment volume per day (sum of four headers) 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 air receivers (4) and standby nitrogen cylinders (3) are charged. This is consistent with maintenance of a 24 hour supply.

A variable area flow sensing device is located in each of the headers supplying makeup air to the four pressurization zones. Signal output from each of the four flow sensors is applied to an integrating recorder located in the control room. High flow alarms are available to alert the operator in the control room. The sensitivity of the flowmeters is well within the maximum leakage of the pressurization system.

<u>Piping Systems Penetrating Containment (Criterion 54)</u>

Criterion: Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems.

Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

Piping systems which penetrate the Reactor Containment have isolation capabilities commensurate with the systems' design in accomplishing safety related functions or providing isolation of the containment from the outside environment under postulated accident conditions.

Piping penetrating the Containment was designed for pressures at least equal to the containment design pressure and, as a minimum, those portions of piping systems which are essential to the isolation function are capable of withstanding the maximum potential seismic loads.

For those systems which require isolation, redundant barriers are utilized to ensure that the failure of one valve to close will not prevent isolation of the penetration. The containment isolation provisions are discussed in detail in Section 5.2.

Isolation valves which are located in lines connecting to the Reactor Coolant System or which could be exposed to the containment atmosphere under postulated accident conditions are sealed by an Isolation Valve Seal Water System which injects water or gas at a pressure slightly higher than the containment design pressure between the isolation barriers. Containment penetrations and welds are sealed by the Containment Penetration and Weld Channel Pressurization System. In addition to providing seals on penetration isolation barriers, these systems may be utilized for leakage detection. The design and operation of the seal system are described in Sections 6.5 and 6.6. Additional leakage detection provisions are discussed in Section 6.7.

Leakage rate testing of the containment is performed in accordance with Technical Specification 5.5.15. The leakage rate program is in accordance with the guidance contained in Regulatory Guide 1.163, except as noted in the Technical Specifications.

Provisions for leakage testing are discussed in Section 5.1.7 and 5.2.

Reactor Coolant Pressure Boundary Penetrating Containment (Criterion 55)

Criterion: Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

3) One locked closed isolation valve inside and one automatic isolation valve outside of containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for in-service inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

Lines which penetrate the Containment and connect to the Reactor Coolant Pressure Boundary were designed with isolation capabilities sufficient to preclude the release of significant amounts of radioactivity. The containment isolation provisions of each line are consistent with the function of the line during normal operation and safety function of the line during and after the Design Basis Accident.

Effluent lines, except the residual heat removal cooldown line, which connect to the Reactor-Coolant Pressure Boundary and which are normally or intermittently open during reactor operation, but which are not required for safe shutdown, are provided with at least two automatic isolation or two closed manual isolation valves in series located outside of containment. Automatic seal water injection is provided for these valves, as described in Sections 5.2.2. and 6.5.2.

A single, normally closed, manually operated, double disc gate valve is provided on the residual heat removal cooldown line outside of containment for isolation purposes. Because of pressure considerations described in Section 5.2.2, the valve can be sealed between the discs by nitrogen from the Isolation Valve Seal Water System. The seal is manually initiated, as required.

Effluent and influent lines connected to the Reactor Coolant Pressure Boundary which are part of systems with essential safety functions and, therefore, must function during or after the Design Basis Accident are provided with containment isolation capabilities commensurate with the required function of the system which the line serves.

Lines in the above category, which are connected to closed systems outside of containment, utilize a single remote manually motor operated double disc gate valve outside of containment as the first isolation barrier. The closed system functions as the second barrier. Manually initiated sealing systems are available to pressurize between the discs of the gate valve after use of the penetration is terminated post accident.

Lines associated with the Residual Heat Removal System which do not terminate in a closed system were designed with isolation provisions which reflect the required function of the line. (Section 5.2)

The residual heat removal return line isolation provisions consist of a check valve inside containment and outside of the missile barrier as the inboard isolation barrier and a normally open, remote manual motor operated double disc gate valve located outside of containment as the outboard isolation barrier. A manually initiated nitrogen seal is available to the outboard valve.

Two normally closed isolation gate valves in series are provided outside of containment for the recirculation pump discharge sample line. The lines between the isolation valve may be sealed by manually initiating a nitrogen seal from the Isolation Valve Seal Water System, as described in Section 5.2.2.

Since the minimum flow line for the residual heat removal pumps must be open upon pump start, two normally open motor operated valves are provided for containment isolation. During the accident, system pressure is adequate to preclude leakage from the Containment. If these valves are shut post-accident, an additional leakage barrier is provided by manually injecting nitrogen between the globe valves.

Motor operated isolation valves were provided for the changing water line with a manual type bypass isolation valve. As shown in Plant Drawing 9321-F-27363 [Formerly Figure 5.2-3], each of the three containment isolation valves are located outside of containment. The inboard and outboard isolation valves in the charging header are normally open to avoid compromising the system function. The control valve bypass is not normally on line and is, therefore, normally shut. If it is necessary to isolate this line post-accident, seal water will be manually initiated to pressurize the line volume between the inboard and outboard isolation valves. Additional protection against reactor coolant leakage from the charging line is provided by a closed system outside of containment, the Chemical and Volume Control System.

The safety injection header is valved in accordance with safeguards requirements. The header bypassing the Boron Injection Tank is provided with either a normally open motor operated double disc gate valve or two normally open motor operated gate valves arranged in series. The header including the Boron Injection Tank has single, normally open, motor operated double disc gate isolation valves. After termination of safety injection, these valves can be manually or remote manually, as applicable, shut, and a nitrogen seal between the discs of each valve can be manually initiated.

Branch lines from the safety injection header which could communicate with the Reactor Coolant Pressure Boundary and which are not required for the safety function are provided with two normally shut isolation valves in series with automatic seal water injection between the valves. Each of the isolation valves is located outside of primary containment, as shown in Plant Drawings 9321-F-27353 and -27503 [Figure 5.2-7].

Each containment penetration is designed as an extension of containment and is seismic Class I and missile protected from the penetration or, if located inside containment, from the inboard isolation barrier to and including the outboard barrier. If automatic or remotely operable isolation valves are provided, the valve was designed to fail in the most safe position as delineated by Table 5.2-3.

<u>Primary Containment Isolation (Criterion 56)</u>

Criterion: Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provision for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Except for the lines or class of lines addressed in subsequent paragraphs, lines penetrating containment, which are not connected to the Reactor Coolant Pressure Boundary and which are not missile protected or which can otherwise communicate with the containment atmosphere following an accident, are provided with a minimum of two isolation valves in series located outside of containment and are provided with automatic seal water injection. Lines which are normally open during reactor operation are provided with automatic isolation valves. Lines which are closed during reactor operation are provided with locked closed or administratively controlled closed manual isolation valves. Isolation provisions for these lines are described in detail in Section 5.2.2., Table 5.2.3 and Plant Drawings 9321-F-27473, -27203, -27353, -27453, -27503, -27513 Sh. 1, -27363, -27193, -27233, -27473, -20253, -27263, -70453, -20173, -20193, -27293 Sh. 1 & 2, -27223, -20353, -40223, -26533, -20363, -26533, and -27243 [Formerly Figures 5.2-1 through 5.2-28], as applicable.

Due to operations and safeguard considerations, there are several lines penetrating containment which, although equivalent in safety, differ from those described above.

The reactor coolant pump seal water supply lines have normally open motor operated isolation valves in series, located outside of containment. As these lines terminate in the Chemical and Volume Control System, a closed system outside primary containment provides a normal and post accident leakage barrier. Post-accident, when coolant pumps are no longer running, these valves are shut and seal water can be initiated to pressurize the line volume between isolation valves and stem packings on the valves closest to containment above post-accident containment pressure. This arrangement is shown in Plant Drawing 9321-F-27363 [Formerly Figure 5.2-4].

The fuel transfer tube penetration inside the Containment was designed to present a missile protected and pressurized double barrier between the containment atmosphere and the atmosphere outside the containment . A positive pressure is maintained between the double gaskets to complete the double barrier between the containment atmosphere and the inside of the fuel transfer tube.

Containment spray headers, because of safeguard considerations, are each provided with a locked open manual double disc gate valve as the inboard isolation valve and a check valve as an outboard isolation valve. The test lines connect between the header isolation and are, therefore, provided with locked closed manual isolation valves. Each isolation valve is located outside of containment, as shown in Plant Drawing 9321-F-27353, and -27503 [Formerly Figure 5.2-6]. After termination of containment spray or in the event of a leak, the inboard isolation valves may be shut and seal water manually initiated between the discs.

The nitrogen supply lines to the Reactor Coolant Drain Tank has a similar arrangement for containment isolation, except that the inboard isolation valve is an air diaphragm valve and no seal injection is available. (Plant Drawings 9321-F-27193 and -27473 [Formerly Figure 5.2-9])

Isolation provisions for the reactor coolant pump cooling water lines, shown in Plant Drawings 9321-F-27203 and -27513 [Formerly Figure 5.2-10] include two motor operated gate valves in series outside of containment which automatically close on a Phase B containment isolation signal. Prior to the valve closure, leakage control is provided by the Component Cooling Water Loop, which is a closed system inside containment on the basis of leak before break criteria. Provisions have been made for manually initiating seal water injection into the line volume between the isolation valves, after valve closure.

Isolation provisions for the recirculation pump discharge sample line include two normally closed gate valves in series located outside of containment. Since leakage through the inboard valve would be at a higher pressure than that of the available seal water, a seal between the valves is provided by manually initiating nitrogen injection form the isolation valve seal water system.

Plant Drawing 9321-F-27503 [Formerly Figure 5.2-21] shows the arrangement of the containment pressure instrumentation lines. These instruments are essential of safe shutdown and the isolation provisions, therefore, consist of a single, locked open, manual globe valve installed in each sensing line.

Post-accident containment sampling lines for the hydrogen monitoring system utilize normally closed, fail closed solenoid operated valves for containment isolation. Each sample supply line and return line has an installed valve serving as an inboard isolation barrier. The outboard isolation barrier is provided by a single valve installed in the supply header and a return valve in the return header.

(Plant Drawing 9321-F-26533 [Formerly Figure 5.2-22]). Since these lines are specifically designed for use in a post-accident condition, all isolation valves are located outside of containment and no sealing connection is required. However, as an additional measure of isolation capability, the space between the inboard and the outboard isolation valves in both supply and return headers for each two containment hydrogen monitors can be pressurized from the weld channel system by two parallel normally open, fail shut solenoid operated valves for each line. During testing of system operation the weld channel valves are manually isolated. An automatic Containment Phase A Isolation signal is provided but is not required since the system is essential for post-accident operation. Capability to close these CIVs is assured during a single failure of the Containment Phase A Isolation signal by using the control room switches.

Lines associated with post-accident containment atmosphere control, except the venting supply line, are provided with automatic outboard isolation valves and manual, normally closed, inboard isolation valves. The inboard isolation valve for venting supply line is a check valve. No seal injection is required and all valves are located of outside containment since the system must operate post accident.

Containment leakage test penetrations are used only when the reactor is shut down. Therefore, no valving is required. The penetrations were provided with flanges or a cap on the inside of containment and the outside of containment to meet dual barrier criteria. Upon a containment isolation signal, each of the subject lines is automatically pressurized with air at a pressure greater than the containment pressure.

The equipment access closure is a bolted, gasket closure, which is sealed during reactor operation. The Personnel Air Locks consist of two doors in series with mechanical interlocks to assure 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 Penetration and Weld Channel Pressurization System. (Section 6.6)

Each containment penetration was designed as an extension of containment and is seismic Class I and missile protected from the penetration or, if located inside containment, from the inboard isolation barrier to and including the outboard barrier. If automatic or remotely operable isolation valves are provided, the valve is designed to fail in the most safe position, as delineated by Table 5.2-1.

Closed System Isolation Valves (Criterion 57)

Criterion: Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

Closed systems within containment not connected to the Reactor Coolant Pressure Boundary include the Steam Generator, Auxiliary Coolant System, Component Cooling Water, Service Water to the Containment Fan Coolers and Motor Coolers, and the Weld Channel Pressurization Air Supply. Each closed system within the Containment is missile protected, as described in Section 5.2.2, and is seismic Class I at least up to and including the isolation valves.

Main steam lines from the steam generators and feedwater lines to the steam generators were designed in accordance with the desirability to maintain the function of the steam generators to transfer heat from the Reactor Coolant System during postulated transients and accidents. Therefore, these lines and the penetrations are designed to permit flow under conditions other than a steam line break. A discussion of the steam and feedwater systems is given in Section 10.2. The associated containment penetrations are illustrated in Plant Drawings 9321-F-20173, -20193, -27293, and -27293 [Formerly Figure 5.2-15] and described by Table 5.2-3.

Component cooling water lines to the residual heat exchangers are required to function post LOCA. Isolation of the inlet lines is accomplished by the installation of single check valve in each inlet line, located just outside of containment. A motor operated gate valve is located just outside of containment in each residual heat exchanger component cooling water return line. During normal operation, the return isolation valves remain closed. Upon the occurrence of a LOCA, the motor operated valves are automatically opened by a safety injector signal. In the unlikely event of a leak, the isolation valves may be remote manually closed to isolate the affected residual heat exchanger. The penetration arrangement for the component cooling water lines to the residual heat exchangers is shown in Plant Drawings 9321-F-27203, and -27513 [Formerly Figure 5.2-11].

Closed systems within the Containment which are required to function during normal operation and during and after the LOCA are provided with a single, normally open, manual isolation valve located outside of containment on each supply and return line. The systems utilizing this arrangement include:

1) Component Cooling Water for Recirculation Pumps (Plant Drawing 9321-F-27203, and -27513 [Formerly Figure 5.2-12])

2) Ventilation Cooling Water for the Containment Fan Coolers and Motor Coolers (Plant Drawing 9321-F-27223 [Formerly Figure 5.2-16])

3) Weld Channel Pressurization Air Supply (Plant Drawings 9321-F-20353, and -27263 [Formerly Figure 5.2-17])

Additional protection is afforded for penetrations associated with Component Cooling Water, ventilation cooling water and Weld Channel Pressurization Air Supply Systems because these systems operate at pressures higher than the peak containment pressure and have makeup sources available. The ventilation cooling water and Weld Channel Pressurization Air Supply Systems are also Closed systems outside of containment. Additional information for isolation provisions of closed systems is within Section 5.2.

The OEH provides two operating conditions during outage:
1. Standby condition which allows the passage of equipment and personnel through the opening.
2. Closed condition capable of retaining air pressure within Containment Building.

The following design loads are used for the design of the OEH

DEAD LOAD:	Weight of the OEH (approximate = 14,000 lbs.)
Design Temperature:	240^o F (inside temperature)
Design Pressure:	-3 psig (outside pressure) to +6 psig (inside pressure)
Normal Temperature Range:	+120^{o}F / -10^{o}F
Seismic Design Loads:	(OBE) and (DBE) in accordance with FSAR Section 16 and IP3 Response Spectra
Maximum Integrated Dose:	$1x10^6$ RAD

1.3.6 Fuel and Radioactivity Control (Criteria 60 to 64)

Control of Releases of Radioactive Materials to the Environment (Criterion 60)

Criterion: The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

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 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 radioactive waste, 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 releases in excess of the limits of 10 CFR 20.

The bulk of the radioactive liquids discharged from the Reactor Coolant System are processed and retained inside the plant by the Chemical and Volume Control System recycle train. This minimizes liquid input to the Waste Disposal System which processes relatively small quantities of generally low-activity level wastes. The processed water from the waste disposal system, from which most of the radioactive material has been removed, is discharged through a monitored line into the circulating water discharge.

Liquid wastes are generated primarily by plant maintenance and service operations.

To maintain the concentration of tritium in the reactor coolant at a level which precludes hazard to personnel during access to the containment, liquid effluent held in the CVCS holdup tanks processed and transferred to the monitor tanks for sampling. Subsequent discharge from the monitor tanks is dependent on the results of sampling analysis. The effluent may be sent back to the holdup tanks for reprocessing, it may be pumped to the primary water storage tank or discharged to the environment with condenser circulating water when within allowable activity concentration as discussed in Chapter 11. Discharged from the sample tanks to the environment is continuously monitored by the Waste Disposal System liquid effluent monitor.

Tritium concentration for effluent discharge is periodically determined to establish the quantity of tritium released in the environment through the waste disposal discharge line to the condenser circulating water. If the total amount of tritium released to the reactor coolant over a year were to be released to the Circulating Water System, the annual average concentration of tritium released to the environment would be between 10^{-2} and 10^{-4} of 10 CFR 20 limits.

Analysis for tritium concentration prior to each batch release is not necessary providing that the criterion of Note 5 of 10 CFR 20, Appendix B is met.

During normal operation, waste radioactive 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 the release of radioactive effluents to the atmosphere is controlled within the limits set in the Technical Specifications.

Gaseous wastes consist primarily of hydrogen stripped from coolant discharged to the CVCS holdup tanks during boron dilution, nitrogen and hydrogen gases purged from the CVCS volume control when degasing the reactor coolant, and nitrogen from the closed gas blanketing system. The gas decay tank capacity will permit 45 days decay of waste gas before discharge.

In the event of a pipe or tank rupture, the maximum anticipated quantity of waste gas that could be released from any one tank in the system, is less than 2.0 x 104 curies of equivalent Xe-133, which would result in a dose of less than 0.5 rem beyond the site exclusion boundary.

Spent resins from demineralizers, filter cartridges and other contaminated solid wastes are packaged and stored onsite until shipment offsite for disposal. Suitable containers are used to package these solids at the highest practical concentrations to minimize the number of containers shipped for burial.

Solid wastes consist of waste liquid concentrates, spent resins and miscellaneous materials such as paper and glassware.

All solid waste is placed in suitable containers and stored onsite until shipment offsite is made for disposal. The Interim Radwaste Storage Facility (IRSF) may be utilized for temporary onsite storage of solidified radioactive wastes. (See Section 11.1)

Hydrological and meteorological conditions indigenous to the Indian Point site were considered in the design of the Waste Disposal System. The hydrology meteorology of the site are described in Sections 2.5 and 2.6, respectively.

<u>Fuel Storage and Handling and Radioactivity Control (Criterion 61)</u>

Criterion: The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety; (2) with suitable shielding for radiation protection; (3) with appropriate containment, confinement, and filtering systems; (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal; and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

All fuel and waste storage facilities are contained and their equipment was designed so that accidental releases of radioactivity directly to the atmosphere are monitored and do not exceed the guidelines of 10 CFR 100.

The reactor cavity, refueling canal and spent fuel storage pit are reinforced concrete structures with a seam-welded stainless steel plate liner. These structures were designed to withstand the anticipated earthquake loadings as seismic Class I structures so that the liner prevents leakage even in the event the reinforced concrete develops cracks. (Section 9.5.1)

The refueling water provides a reliable and adequate cooling medium for spent fuel transfer and heat removal from the spent fuel pit is provided by an auxiliary cooling system. Natural radiation and convection is adequate for cooling the waste holdup tanks.

The structural steel and metal siding building surrounding the spent fuel pit is seismic Class III, as is the Fuel Storage Building Crane.

The Auxiliary Coolant System consists of three loops as shown in Plant Drawings 9321-F27203 and -27513 [Formerly Figures 9.3-1, 9.3-2A and 9.3-2B]: the component loop, the residual heat removal loop, and the spent fuel pit cooling loop.

The spent fuel pit cooling loop was designed to remove the heat generated by stored spent fuel elements from the spent fuel pit. Alternate cooling capability can be made available under anticipated malfunctions or failures (expected fault conditions).

The active components of the Auxiliary Coolant System are in either continuous or intermittent use during normal plant operation so that no additional periodic tests are required. Periodic visual inspections and preventive maintenance are conducted following normal industry practice. (Section 9.3.5)

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, less than 2.0 mr/hr, for periodic occupancy of the area by operating personnel. Pit water level is indicated, and water removed from the pit must be pumped out since there are no gravity drains. Shielding is provided for waste handling and storage facilities to permit operation within requirements of 10 CFR 20.

Gamma radiation is continuously monitored in the auxiliary building. A high level signal is alarmed locally and is annunciated in the Control Room.

Auxiliary shielding for the Waste Disposal System and its storage components was designed to limit the dose rate to levels not exceeding 0.75 mr/hr in normally occupied areas, to levels not exceeding 2.0 mr/hr in intermittently occupied areas and to levels not exceeding 15 mr/hr in limited occupancy areas. (Section 11.2)

A controlled leakage building designed for a negative pressure of 0.50 inches of water minimum, permanently encloses the fuel pool. The design features of the fuel handling building that provide this leaktightness include the following items:

1) Special sealing features at joints that include:
   a) Sealing off all edges and ends of the walls with a combination of caulking and relatively soft neoprene strip,
   b) Installation of necessary additional closure flashings at the extremities and at openings,
   c) Supplying additional caulking in vertical and horizontal joints of liner panels,
   d) Furnishing liner panels in sufficient thickness to seat well on girt spacings and resist flexing in addition to withstanding the normal design loads, and
   e) Providing additional fastening for liner panels.

2) Personnel and rolling steel truck doors with inflatable air seals. These seals are inflated upon a high radiation alarm from R-5, although R-5 operability does not require this function.

3) Motor operated dampers designed to fail closed are installed on the discharge side of the two supply fans.

Prior to handling operations when irradiated fuel is within the fuel handling building, tests were performed to verify the building leak tightness. Fuel handling operations in the Fuel Storage Building are detailed in Section 3.7.13 of Technical Specifications and Section 9.5 of the FSAR.

Carbon filters, together with suitable face dampers and manual isolation devices are part of the normal Fuel Storage Building Ventilation System and operate as follows:

1) The carbon filters and motor operated dampers are located in the fan plenum downsteam from the roughing and HEPA filters.

2) The fuel storage building supply air fans are provided with motor operated dampers located on the discharge side of the fan. These dampers are interlocked with their respective fan motors and arranged to close when fan motor stops and open during fan motor operation.

3) Manual isolation devices will be installed during all fuel handling operations and leak tested to ensure that all of the air from the fuel storage building is discharged through the roughing HEPA and charcoal filters.

4) A radiation indicator located in the spent fuel pit area automatically initiates the emergency mode of operation by:
   a) Stopping fuel storage building supply fans, thereby closing their respective dampers.
   b) Opening carbon filter face dampers.

5) The exhaust system has a capacity of approximately 20,000 cfm which maintains a negative pressure in the fuel storage building.

The probability of inadvertently draining the water from the cooling loop of the spent fuel pit is exceedingly low. The only means is through actions such as opening a valve on the cooling line and leaving it open when pump is operating. In the unlikely event of the cooling loop of the spent fuel pit being drained, the spent fuel storage pit itself cannot be drained and no spent fuel is uncovered since the spent fuel pit cooling connections enter near the top of the pit. With no heat removal, the time for the spent fuel pit water to rise from 120 ^{0} F to 180 ^{0} F with 76 fuel assemblies stored in the pit is approximately 13.8 hours. The temperature and level indicators in the spent fuel pit would warn the operator of the loss of cooling. This slow heatup rate of the spent fuel pit would allow sufficient time to take any necessary action to provide adequate cooling while the cooling capability of the spent fuel pit cooling loop is being restored.

Assuming that the reactor has recently been refueled and 76 assemblies are stored in the pool, and 76 assemblies were placed in the spent fuel pit, a fission product decay period of approximately 47.5 days would be required after the spent fuel was placed in the pit before the natural heat loss from the pit would be equivalent to the decay heat.

The most serious failure of this loop is complete loss of water in the storage pool. To protect against this possibility, the spent fuel storage pool cooling connections enter near the water level so that the pool cannot be either gravity-drained or inadvertently drained. For this same reason, care is also exercised in the design and installation of the fuel transfer tube. The water in the spent fuel pit below the cooling loop connections could be removed by using a portable pump.

Loss of water in the spent fuel pit and the resultant uncovering of the spent fuel by way of drains, and permanently connected systems cannot take place for the following reasons:

a) The suction of the Spent Fuel Pit Pump is taken from a point approximately six (6) feet below the top of the pool wall; therefore, this pump cannot be used to uncover fuel, even accidentally

b) The Spent Fuel Pump discharge pipe terminates in the pool at elevation 74' – 4 ¾". This elevation is approximately five (5) feet above the top of the spent fuel assemblies; therefore, this pipe could not accidentally become a syphon to uncover the fuel.

c) The skimmer pump takes suction from, and discharges to the surface of the pool; therefore, it could not accidentally or otherwise uncover the spent fuel

d) There are no drains on the bottom or side walls of the spent fuel; draining has to be done deliberately by a temporary pump

e) The spent fuel pit cooling loop was designed to seismic Class II, the cleanup equipment and skimmer loops were designed to seismic Class III criteria; however, their failure could not result in the uncovering of the spent fuel, as explained above

The primary source of makeup water to the spent fuel pit is the Primary Makeup Water Storage Tank, which is a seismic Class I component. The pumps and most of the piping associated with this system are also seismic Class I. The makeup water loop to the Spent Fuel Pit is seismic Class II, as is the spent fuel pit cooling loop. The cleanup equipment and skimmer loops are seismic Class III. Redundant sources of makeup water to the spent fuel pit are from the refueling water storage tank and the city water supply. In addition, there are provisions for the connection of a temporary cooling system. See Section 9.5 for further discussion of possible loss of water from the spent fuel pit and makeup capabilities.

In addition, a second Spent Fuel Pool Cooling System pump was installed identical to and in parallel with the original pump to provide installed standby pumping capability to the Spent Fuel Pool Cooling System.

The reliability of pumping capability is further enhanced by powering the two pumps from different electrical power buses. Associated piping and valves were designed and analyzed to requirements consistent with the existing system. The modified portions of the system were as a minimum, equal to the standards of the original system and in most cases represents an upgrading of design, material, fabrication, testing and/or quality assurance.

Prevention of Criticality in Fuel Storage and Handling (Criterion 62)

Criterion: Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

The Indian Point 3 spent fuel storage pool is equipped with seismic Class I maximum density spent fuel storage racks for an expanded total storage capacity of 1345 fuel assemblies. A stainless steel liner insures against a loss of water. Detailed instructions, licensing bases including Technical Specification requirements and the design of the fuel handling equipment incorporating interlocks and safety features, provide assurance that no incident could occur during refueling, fuel handling, or storage operations that could result in a hazard to the public health and safety. (Section 9.5)

Borated water is used to fill the spent fuel storage pit to a concentration matching that of the reactor cavity and refueling canal during refueling operations. A shutdown margin of 5% Dk/k is maintained, in the cold condition, with all rods inserted. Periodic checks of refueling water boron concentration and the residual heat removal pump operation insure the proper shutdown margin. Direct communications between the control room operator and the manipulator operator allows immediate notification of any impending unsafe condition detected during fuel movement.

The racks are arranged and categorized in two regions based on fuel assembly enrichment and burn-up. All storage cells are bounded on four sides by boral poison sheets, except on the periphery of the pool rack array. The rack are designed to assure that a keff of less than or equal to 0.95 is maintained provided that Technical Specifications dictating placement of fuel in the spent fuel pit are followed.

The storage rack design is a free-standing welded honeycomb array of stainless steel boxes which has no grid frame structure. The racks are supported and leveled on four screw pedestals which bear directly on the pool floor. The racks are free to move horizontally, and strong hydrodynamic coupling between racks causes the racks to move together without rack-to-rack impact. The free-standing design allows any single or combination of racks to withstand a design basis seismic event without toppling or causing damage to fuel assemblies inserted within them.

The core subcritical neutron flux is continuously monitored by two source range neutron monitors, each with continuous visual indication in the Control Room and with one audible indication in the Containment whenever core geometry is being changed.

Fuel Handling System cranes are dead-load tested before fuel movement begins. The test load assumed by the hoists or cranes must be equal to or greater than the maximum load assumed during the refueling operation. Additionally, a thorough visual inspection is made following the dead-load test and prior to fuel handling. A test of interlocks is also performed each refueling, prior to movement of core components. As excess weight interlock is provided to prevent movement of more than one fuel assembly at a time.

Monitoring Fuel and Waste Storage (Criterion 63)

Criterion: Appropriate systems shall be provided in the fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in the loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

Monitoring and alarm instrumentation are provided for fuel and waste storage and handling areas to detect inadequate cooling and to detect excessive radiation levels.

Radiation monitors are provided to maintain surveillance over the waste release operation. The permanent record of activity releases is provided by radiochemical analysis of known quantities of waste.

There is a controlled ventilation system for fuel storage and waste treatment areas of the auxiliary building which discharges to the atmosphere via the plant vent. Radiation monitors are in continuous service in these areas to actuate a high-activity alarm on the control board annunciator.

Auxiliary shielding for the Waste Disposal System and its storage components was designed to limit the dose rate to levels not exceeding 0.75 mr/hr in normally occupied areas, to levels not exceeding 2.00 mr/hr in intermittently occupied areas and to levels not exceeding 15 mr/hr in limited occupancy areas.

The fuel handling mechanisms were designed so that it is unlikely that an accidental release of radioactivity can take place. These components are also contained within the fuel storage building which further reduces the chance of a “leak” and assists in maintaining the guidelines set up by 10 CFR 100. Furthermore, gamma radiation levels in the Fuel Storage Building itself are continuously monitored by a local (R-5) Area Radiation Monitor. The monitoring serves to warn the operator of impending high radiation levels for such cases as low water level, contaminated water or improper handling of irradiated equipment or fuel elements. If the set point is reached, it is alarmed locally and in the Control Room.

Whenever the ventilation system is required to be in operation, the bypass damper around the charcoal filter must be manually closed. On a high radiation alarm, the following actions automatically take place:

1) Building ventilation supply fans are secured,
2) Dampers at ventilation supply fan close,
3) If open, rolling door closes,
4) Inflatable seals on main doors and truck doors are actuated (R-5 operability does not require this function, however), and
5) Exhaust fans continue to run.

Under these conditions, the maximum calculated in-leakage to the building (as a result of non-air tight construction) would be 20,000 cfm with a one-half inch of water negative pressure inside the building. Thus, there will be zero air leakage from the building proper, and the entire exhaust from the building will pass through roughing HEPA and charcoal filters before passing into the plant vent.

A spent fuel pit cooling loop which is a part of the Auxiliary Coolant System is provided to remove from the spent fuel pit the heat generated by the stored fuel elements. Both the water level and temperature are continuously monitored. High and low levels in the pit (6" above or below the 93' –8" normal) are alarmed in the Control Room, as is high temperature of the water in the pit (135 ^{0} F).

Two monitor tanks are provided to collect liquid wastes processed by the liquid waste disposal system that are suitable for direct release to the river. When a monitor tank is full, it will be isolated and the second tank will be placed in service. The isolated tank is then recirculated and a sample is taken.

The sample taken will be analyzed for gross activity. If the water is considered unsuitable for discharge, it will be returned to the waste holdup tank for reprocessing.

A direct measurement of the activity can be made by means of the radiation detector R-18 located in the monitor pumps' discharge line so that the liquid wastes can be monitored during both recirculation and discharge. The activity level will be indicated on the Waste Disposal Panel and in the Control Room (on the Radiation Monitoring System Cabinet). If the activity exceeds the high alarm setpoint, an alarm will be annunciated at the waste disposal panel, "WDS Liquid Monitor Hi Radiation." In addition, if the activity reaches the alarm point, the control valve RCV-018 in the waste release line will be tripped shut via an electrical interlock. This radiation detector thus provides a backup to the sample analysis in preventing the accidental release of high activity liquid.

If sample analysis indicates the fluid is suitable for discharge to the river, the allowable release rate will be determined and a radioactive waste release permit will be filled out for the particular tank to be dumped. RCV-018 will be opened from the waste disposal panel and locked closed valve No. 1785 will be opened. The pump discharge valve will be throttled to maintain the flow rate, as indicated on the waste disposal panel. The released liquid waste enters the discharge canal via the service water return line from the component cooling water heat exchangers.

The activity of the fluid being released is continuously monitored by R-18 so that in the event that high activity water inadvertently entered the fluid being released, it would be detected and RCV-018 would be tripped closed. This valve cannot be reopened until the high radiation condition is corrected and the alarm is reset from the Control Room.

Four 525 cubic foot large gas decay tanks and six 40 cubic foot small gas decay tanks are provided to hold radioactive waste gases for decay.

The large gas decay tanks are sampled periodically by a gas analyzer. Only the tank in the process of being filled will be sampled; the other tanks will be bypassed. (Operation of the large gas decay tank sample discharge valves, PCV-1036B to 1039B, are controlled by the gas analyzer during this sampling process.) A radiation monitor, R-20, will indicate its reading on the Waste Disposal Panel and will alarm when gaseous activity in the tank being filled reaches the high alarm setpoint (variable setpoint). This alarm is annunciated on the waste disposal panel as “Gas Activity Monitor Hi-Activity” and also in the Control Room. The alarm is provided so the operator can stop the filling operation before the curie limit on the tank is reached. A maximum of 50,000 curies of equivalent Xenon-133 is allowed in any one tank so that the site boundary limits of 10CFR20 will not be exceeded if the tank fails.

Monitoring Radioactivity Releases (Criterion 64)

Criterion: Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of LOCA accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

The containment atmosphere, the plant vent, the containment fan coolers service water discharge, the Waste Disposal System gas and liquid effluent, the condenser air ejectors, the component cooling loop liquid, the component cooling water heat exchanger service water discharge, the discharges from the condensate polisher Low Total Dissolved Solids (LTDS) and High Total Dissolved Solids (HTDS) Waste Collection Tanks and the steam generator blowdown are monitored for radioactivity released during normal operations, from anticipated transients, and from accident conditions. (Section 11.2)

All gaseous effluent from possible sources of accident releases of radioactivity external to the reactor containment (e.g., the spent fuel pit and waste handling equipment) will be exhausted from the plant vent. All accidental spills of liquids are maintained within the auxiliary building and collected in a drain tank. Any waste disposal system liquid effluent discharged to the condenser circulating water canal is monitored by R-18. For the case of leakage from the reactor containment under accident conditions, the plant area radiation monitoring system, supplemented by portable survey equipment, is provided for monitoring of accident releases.

All fuel and waste storage facilities are contained and equipment designed so that accident releases of radioactivity to the atmosphere are monitored. (Sections 9.5 and 11.2)

Instruments for monitoring radioactivity releases are located at selected points in and around the plant to detect, indicate and record the radiation levels. The system consists of the channels detailed below.

The Containment Air Particulate Monitor (R-11) measures air particulate beta radioactivity in the containment, and ensures that the release rate through the containment vent during purging is maintained below specified limits.

High radiation level for the channel initiates closure of the containment purge supply and exhaust duct valves and pressure relief line valves.

A Containment Radioactive Gas Monitor (R-12) is provided to measure gaseous beta radioactivity in the containment to ensure that the radioactivity release rate during purging is maintained below specified limits. High gas radiation level initiates closure of the containment purge supply and exhaust duct valves and pressure line relief valves.

The Plant Vent Gas Monitor (R14) detects radiation passing through the plant vent to the atmosphere.

Remote indication and annunciation are also provided on the Waste Disposal System control board. On high radiation level alarm the gas release valve in the Waste Disposal System is automatically closed, thus assuring that gaseous releases from the Waste Disposal System are within the specified limits.

The condenser Air Ejector Gas Monitor (R-15) uses a gamma sensitive sodium iodide (NaI) crystal scintillator/photomultiplier to monitor the discharge from the air ejector exhaust header of the condensers for gaseous radiation which is indicative of a primary to secondary system leak. The normal gas discharge is routed to the turbine roof vent. On high radiation level alarm, the condenser exhaust gases are diverted to the containment through a blower.

Waste Disposal System Liquid Effluent Monitor (R-18) continuously monitors all Waste Disposal System liquid releases from the plant. A scintillation counter and sample chamber assembly monitor these effluent discharges. Automatic valve closure action is initiated by this monitor to prevent further release after a high radiation level is indicated and alarmed. Remote indication and annunciation are also provided on the Waste Disposal System control board.

The measurement ranges of these monitors are given in Section 11.2.

1.4 COMPARISON OF DESIGN PARAMETER

The design parameters listed in Table 1.4-1 represent those design parameters for Indian Point 2 and Indian Point 3 in effect at the time of the original license application by Consolidated Edison. These have been retained for historical reference; future comparisons or revisions will not be made.

1.4.1 Thermal and Hydraulic Design Parameters

Most of the significant changes which were made in the thermal and hydraulic parameters between Indian Point 2 and 3 resulted from the increase in power level in Indian Point 3.

The Indian Point 3 power level was about 10% higher (3025 MWt vs 2758 MWt) than for Indian Point 2 (Line 1)* and this resulted in similar increases in total heat output (Line 2), average heat flux (Line 25) and average thermal output (Line 27).

The fuel and burnable poison arrangement was optimized in Indian Point 3 to obtain the hot channel factors (Line 7 and 8). These hot channel factors resulted in maximum heat flux (Line 26) and maximum thermal output (Line 28) values associated with the higher heat flux and thermal output average values as pointed out above.

The use of helium pressurized fuel rods with larger pellet clad gaps (Line 41) resulted in maximum fuel center line (rod) temperature which was about the same as for Indian Point 2 (Lines 30 and 31).

The coolant temperature parameters listed in Line No. 15-21 were slightly higher than for Indian Point 2 due to increased power level.

The rest of the parameters listed in this section of Table 1.4-1 were very similar to those listed for Indian Point 2; the differences are not significant.

1.4.2 Core Mechanical Design Parameters

Indian Point 3 reflected a more advanced core design than Indian Point 2. Westinghouse experience in the design and operation of PWR's, together with continuing development programs, resulted in the modified fuel assembly design employed in Indian Point 3. Similar fuel assembly designs have been used in H. B. Robinson Unit No. 2, Turkey Point No. 3 and Surry Unit No. 1. This design is characterized by a reduced number of grids per assembly (Line 38), Zr-4 guide tubes (Indian Point 2 had stainless guide tubes, at the time of this comparison), helium pressurized rods and fuel pellet diameters

* Lines referenced in this section are for Table 1.4-1

(Lines 41 and 46) which vary with the power output and lifetime of the particular core region.

The combination and inter-relationship of higher power, use of zircaloy instead of stainless guide tubes (higher reactivity), and a desired negative moderator coeffiencient at power required the use of more burnable poison rods in Indian Point 3 than Indian Point 2 (Line 53).

Other core mechanical design parameters were as in Indian Point 2.

1.4.3 <u>Final Nuclear Design Data</u>

The fuel and poison arrangement coupled with a higher desired power output and improved core design yielded control characteristics for Indian Point 3 somewhat different from Indian Point 2. The effective multiplication parameters (Lines 72 - 74) were slightly higher than for Indian Point 2; this is due to the interplay between higher reactivity (higher enrichment and improved fuel design) and the increased number of burnable poison rods for the first core (Line 53).

Improved core design also permitted improved expected performance as the fuel discharge burnups (Lines 66 and 67) show. Westinghouse acquired substantial experience on fuel design; this experience was incorporated into the Westinghouse design techniques and predicted core performance. Experience on Westinghouse cores has shown calculated parameters to be very close to measured parameters demonstrating the effectiveness of calculational techniques in core designs.

Fuel and control rod characteristics and improved (flatter) power shape have made Indian Point 3 RCC assemblies to be worth slightly more than those in Indian Point 2 (Line 75); this, again, due to an improved core design.

A flatter power shape was achieved by a combination of altered fuel enrichment (Line 68), and optimization of the fuel and poison rod arrangements.

The rest of the nuclear design parameters were very similar to those for Indian Point 2 and the slight differences can be explained by the differences in the core and improved calculational techniques based upon extensive experience gained by Westinghouse.

1.4.4 Reactor Coolant System

The design capacity and head of the reactor coolant pump (Lines 121 and 122) and tube side design flow rate in the steam generators (Line 108) were changed to reflect the increase in core resistance (due to a more advanced core design) and a modified pump impeller design (more efficient). No other significant changes have occurred in the Reactor Coolant or Steam Systems or their components.

TABLE 1.4-1

COMPARISON OF DESIGN PARAMETERS [1]

REFERENCE	INDIAN POINT 3	INDIAN POINT 2	LINE NO.
THERMAL AND HYDRAULIC DESIGN PARAMETERS			
Total heat output, MWt	3025	2758	1
Total heat output, Btu/hr	10,324 x 10^6	9413 x 10^6	2
Heat generated in fuel, %	97.4	97.4	3
Maximum thermal overpower, with design hot channel factors, %	12	12	4
System pressure, nominal, psia	2250	2250	5
System pressure, minimum steady state, psia	2220	2220	6
Hot Channel Factors			
Heat Flux, F_q	2.32	2.24	7
Enthalpy rise, $F_{\Delta H}$	1.55	1.55	8
DNB ratio at nominal conditions (Min)	1.89 (L-grid)	2.16 (W-3)	9
Minimum DNBR for design transients	1.30 (L-grid)	1.30 (W-3)	10
Coolant Flow			
Total flow rate, lb/hr	136.3 x 10^6	136.1 x 10^6	11
Effective flow rate for heat transfer through core, lb/hr	130.1 x 10^6	130.0 x 10^6	12
Average velocity along fuel rods, ft/sec	15.6	15.5	13
Average mass velocity through core, lb/hr-ft^2	2.54 x 10^6	2.54 x 10^6	14
Coolant Temperature, F			
Nominal inlet	542.6	543	15
Maximum inlet due to instrumentation error and deadband	546.6	547.0	16

These represent those parameters for Indian Point 3 and Indian Point 2 in effect at the time of the original license application and are retained for historical reference only.

Table 1.4-1
(Cont.)

COMPARISON OF DESIGN PARAMETERS [1]

REFERENCE	INDIAN POINT 3	INDIAN POINT 2	LINE NO.
Average rise in vessel	57.8	53.1	17
Average rise in core	62.4	55.4	18
Average in core	574.0	571.7	19
Average in vessel	571.5	569.5	20
Nominal outlet of hot channel	632.0	628.9	21
Average film coefficient, Btu/hr-ft^2 – F	5990	5970	22
Average film temperature drop, F	32.2	29.9	23
Heat Transfer at 100% Power			
Active heat transfer surface area, ft^2	52,100	51,400	24
Average heat flux, Btu/hr-ft^2	193,000	178,400	25
Maximum heat flux, Btu/hr-ft^2	447,500	399,700	26
Average thermal output, kW/ft	6.24	5.78	27
Maximum thermal output, kW/ft	14.5	12.94	28
Maximum clad surface temperature at nominal pressure, F	657	657	29
Fuel Central Temperature, F			
Maximum 100% power	3600	3300	30
Maximum at overpower	4500	4500	31
Thermal output, kW/ft at maximum overpower	21.1	21.1	32
CORE MECHANICAL DESIGN PARAMETERS			
Fuel Assembly			
Design	RCC Canless 15 x 15	RCC Canless 15 x 15	33
Rod Pitch, inches	0.563	0.563	34
Overall dimensions, inches	8.426 x 8.426	8.426 x 8.426	35
Fuel Weight (as UO_2), pounds	221,600	216,600	36
Total weight, pounds	284,000	276,000	37
Number of grids per assembly	7	9	38

TABLE 1.4-1
(Cont.)

COMPARISON OF DESIGN PARAMETERS [1]

REFERENCE	INDIAN POINT 3	INDIAN POINT 2	LINE NO.
Fuel Rods			
Number	39,372	39,372	39
Outside diameter, inches	0.422	0.422	40
Diametral gap, inches	0.0075	0.0065	41
Clad thickness, inches	0.0243	0.0243	42
Clad material	Zircaloy-4	Zircaloy-4	43
Fuel Pellets			
Material	UO_2 Sintered	UO_2 Sintered	44
Density (% of theoretical)	95	94-95-95	45
Diameter, inches	0.3659	0.366	46
Length, inches	0.600	0.600	47
Rod Cluster Control Assemblies			
Neutron absorber	5% Cd; 15% In; 80% Ag	5% Cd; 15% In; 80% Ag	48
Cladding material	Type 304 SS-Cold Worked	Type 304 SS-Cold Worked	49
Clad Thickness, inches	0.019	0.019	50
Number of clusters	53	53	51
Number of control rods per cluster	20	20	52
Number of burnable poison rods	1434 (1st cycle)	1412	53
Core Structure			
Core barrel ID/OD, inches	148.38/152.95	148.0/152.5	54
Thermal shield ID/OD, inches	158.5/164.1	158.5/164.0	55
FINAL NUCLEAR DESIGN DATA			
Structure Characteristics			
Clad Weight, pounds	42,915	44,600	56
Core diameter, inches	133.7	132.75	57
Core height, inches	144	144	58
Reflector Thickness and Composition			
Top – water plus steel, inches	10	10	59
Bottom – water plus steel, inches	10	10	60
Side – water plus steel, inches	20	15	61

TABLE 1.4-1
(Cont.)

COMPARISON OF DESIGN PARAMETERS [1]

REFERENCE	INDIAN POINT 3	INDIAN POINT 2	LINE NO.
H_2O/U unit cell (core volume ratio)	4.03	4.01	62
Number of fuel assemblies	193	193	63
UO_2 rods per assembly	204	204	64
Performance Characteristics			
Loading technique	3 region, checker board	3 region, checker board	65
Fuel Discharge Burnup, MWD/MTU			
Average first cycle	17,000	16,100	66
First core average	28,000	24,700	67
Feed Enrichments, w/o			
Region 1	2.28	2.21	68
Region 2 (first core with burnable poison)	2.80	2.80	69
Region 3	3.30	3.30	70
Equilibrium	3.2	3.2	71
Control Characteristics (Beginning of life)			
Effective Multiplication (with Burnable Poison)			
Cold, no power, clean	1.17	1.113	72
Hot, no power, clean	1.14	1.057	73
Hot, full power, Xe and Sm equilibrium	1.09	1.001	74
Rod Cluster Control Assemblies			
Total rod worth BOL (calculated) Hot, no power, clean, ∂ k	9.45	8.46	75
Boron Concentrations (First Cycle with Burnable Poison)			
To shut reactor down with no rods inserted, clean (k_{eff}=0.99) cold/hot, ppm	1500/1476	1370/1405	76
To control at power with no rods inserted clean/equilibrium xenon and samarium, ppm	1228/933	1186/890	77

TABLE 1.4-1
(Cont.)

COMPARISON OF DESIGN PARAMETERS(1)

REFERENCE	INDIAN POINT 3	INDIAN POINT 2	LINE NO.
Boron worth, hot (at power), ∂ k/k-ppm	1% / 104	1% / 89	78
Boron worth, cold (zero power), ∂ k/k-ppm	1% / 83	1% / 72	79
Kinetic Characteristics			
Moderator temperature coefficient, ∂ k/k/F	-0.0×10^{-4} to -3.5×10^{-4}	-2.5×10^{-4} to -3.00×10^{-4}	80
Moderator pressure coefficient, ∂ k/k/psi	0.3×10^{-6} to $+4.0 \times 10^{-6}$	$+0.2 \times 10^{-6}$ to 3.00×10^{-6}	81
Moderator void (density) coeffient, ∂ k / gm /cm^3	-0.0 to +0.43,	-0.1 to +0.3	82
Doppler coefficient, ∂ k/k/F	-1.0×10^{-5} to -2.0×10^{-5}	-1.1×10^{-5} to -1.8×10^{-5}	83
REACTOR COOLANT SYSTEM – CODE REQUIREMENTS			
Component			
Reactor vessel	ASME III Class A+	ASME III Class A+	84
Steam Generator			
Tube Side	ASME III Class A+	ASME III Class A+	85
Shell Side	ASME III Class A+	ASME III Class C+	86
Pressurizer	ASME III Class A+	ASME III Class A+	87
Pressurizer relief tank	ASME III Class C+	ASME III Class C++	88
Pressurizer safety valves	ASME III+	ASME III	89
Reactor coolant piping	ANSI B31.1-1955	ANSI B31.1-1955	90

\+ 1965 Code Edition with applicable Addenda
++ 1964 Code Edition with applicable Addenda

TABLE 1.4-1
(Cont.)

COMPARISON OF DESIGN PARAMETERS[1]

REFERENCE	INDIAN POINT 3	INDIAN POINT 2	LINE NO.
PRINICIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT SYSTEM			
Operating pressure, psig	2235	2235	91
Total reactor coolant system volume, cu. ft.	12,242	12,224	92
PRINICIPAL DESIGN PARAMETERS OF THE REACTOR VESSEL			
Material	SA 302 Grade B, low alloy steel, internally clad with type 304 austenitic stain-less steel	SA 302 Grade B, low alloy steel internally clad with type 304 austenitic stain-less steel	93
Design Pressure, psig	2485	2485	94
Design temperature, F	650	650	95
Operating pressure, psig	2235	2235	96
Inside diameter of shell, inches	173	173	97
Overall height of vessel and enclosure head, ft-inches	43 – 9 11/16	43 – 9 11/16	98
Minimum clad thickness, inches	5/32	5/32	99
Outlet nozzle radius, inches	122 13/16	123 1/16	100
Inlet nozzle radius, inches	130 31/32	131 7/32	101
PRINCIPAL DESIGN PARAMETERS OF THE STEAM GENERATORS			
Number of units	4	4	102
Type	Vertical, U-tube with integral-moisture separator	Vertical, U-tube with integral-moisture separator	103
Tube material	Inconel	Inconel	104
Shell material	SA 302 Grade B	SA 302 Grade B	105
Tube side design pressure, psig	2485	2485	106
Tube side design temperature, F	650	650	107
Tube side design flow, lb/hr	34.08×10^6	34.03×10^6	108
Shell side design pressure, psig	1085	1085	109
Shell side design pressure temperature, F	600	600	110

TABLE 1.4-1
(Cont.)

COMPARISON OF DESIGN PARAMETERS[1]

REFERENCE	INDIAN POINT 3	INDIAN POINT 2	LINE NO.
Operating pressure, tube side, nominal, psig	2235	2235	111
Operating pressure, shell side, maximum, psig	1005.3	1005.3	112
Maximum moisture at outlet at full load, %	¼	¼	113
Hydrostatic test pressure, tube side (code), psig	3110	3110	114
PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT PUMPS			
Number of units	4	4	115
Type	Vertical, single stage radial flow with bottom suction and horizontal discharge	Vertical, single stage radial flow with bottom suction and horizontal discharge	116
Design pressure, psig	2485	2485	117
Design temperature, F	650	650	118
Operating pressure, nominal, psig	2235	2235	119
Suction temperature, F	555	555	120
Design capacity, gpm	88,500	89,700	121
Design Head, ft	277	272	122
Hydrostatic test pressure (cold), psig	3110	3110	123
Motor type	AC induction single speed	AC induction single speed	124
	6000 hp	6000 hp	125
PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT PIPING			
Material	Austenitic SS	Austenitic SS	126
Hot leg ID, inches	29	29	127
Cold leg ID, inches	27-1/2	27-1/2	128
Between pump and steam generator ID, inches	31	31	129
Design pressure, psig	2485	2485	130

TABLE 1.4-1
(Cont.)

COMPARISON OF DESIGN PARAMETERS(1)

REFERENCE	INDIAN POINT 3	INDIAN POINT 2	LINE NO.
STEAM AND POWER CONVERSION DESIGN PARAMETERS			
Turbine-Generator Turbine type	Four-element, tandem-compound, six-flow exhaust	Four-element, tandem-compound, six-flow exhaust	131
Turbine capacity, kW			
Maximum guaranteed	1,021,793	1,021,793	132
Maximum calculated	1,068,701	1,068,701	133
Generator rating, kVa	1,125,600	1,125,600	134
Turbine speed, rpm	1800	1800	135
Condensers Type	Radial flow, single pass, divided water box	Radial flow, single pass. divided water box	136
Number	3	3	137
Condensing capacity, lb of steam/hr (each)	7,230,000	7,230,000	138
Condensate pumps Type	8 stage, vertical, pit type, centrifugal	8 stage, vertical, pit type, centrifugal	139
Number	3	3	140
Design capacity, (each – gpm)	7860	7860	141
Motor type	Vertical, induction	Vertical, induction	142
Motor rating, hp	3000	3000	143

TABLE 1.4-1
(Cont.)

COMPARISON OF DESIGN PARAMETERS[(1)]

REFERENCE	INDIAN POINT 3	INDIAN POINT 2	LINE NO.
Feedwater pumps			
Type	High speed, barrel casing, single stage centrifugal	High speed, barrel casing, single stage centrifugal	144
Number	2	2	145
Design capacity (each), gpm	15,300	15,300	146
Motor type	horizontal steam turbine	horizontal steam turbine	147
Motor rating, hp	8350	8350	148
Auxiliary feedwater source	360,000 gallons assured reserve in 600,000 gal condensate tank. Alternate supply from 1,500,000 gal city water tank	360,000 gallons assured reserve in 600,000 gal condensate tank. Alternate supply from 1,500,000 gal city water tank	149
Auxiliary feedwater pumps			
Number	3 (one steam-driven, two electric motor-driven)	3 (one steam-driven, two electric motor-driven)	150
Design capacity, gpm	800 (steam-driven)	800 (steam-driven)	151
	400 (each, motor-driven pump	400 (each, motor-driven pump)	

1.5 RESEARCH AND DEVELOPMENT REQUIREMENTS

The design of Indian Point 3 was based upon proven concepts which were developed and successfully applied to the design of numerous other pressurized water reactor systems.

The research and development programs discussed in this section were intended to confirm the engineering and design values normally used to complete equipment and system designs. These did not involve the creation of new concepts or ideas.

The technical information generated was used either to demonstrate the safety of the design and more sharply define margins of conservatism or to lead to design improvements.

Each research and development program is briefly summarized for identification. Detailed discussions of each program are available in more detail in the references given in this section.

1.5.1 Programs Required for Plant Operation

There were four programs identified in the initial license application as required for plant operation:

- Core Stability Evaluation
- Fuel Rod Burst Program
- Containment Spray Program
- Charcoal Filter Program.

1. Core Stability Evaluation

   The purpose of this program was to establish the means for detection and control of potential xenon oscillations and for shaping of the axial power distribution for improved core performance. This program has been completed. A xenon induced X-Y oscillation test was conducted at Indian Point 2. Therefore, no test was needed for Indian Point 3. Reference 1 provides a further discussion of the tests performed.

2. Fuel Rod Burst Program

   The original rod burst program, a study of the performance of zircaloy cladding under simulated Loss-of-Coolant Accident (LOCA) conditions, was completed. This program supplied empirical data from which the effect of geometry distortion on the ability of the Emergency Core Cooling System (ECCS) to meet the LOCA design criteria was determined using available analytical design techniques.

   The program included burst and quench tests on single rods and burst tests on rod bundles. As a result of single rod tests, specific design limits were established on peak clad temperature and allowable maximum metal water reaction to assure effective core cooling. The multirod burst tests demonstrated that even when rod to rod contact does occur after burst, the remaining flow area is always sufficient to ensure adequate core cooling.

Reference 2 (single rod) and Reference 3 (multirod) provide a further discussion of these tests.

On January 8, 1992, the Authority filed an application (supplemented by letter dated February 26, 1992) to address the use of ZIRLO™, as well as Zircaloy-4, fuel rod cladding. Amendment No. 117 to Facility Operating License DPR-64, and the associated safety evaluation (Reference 6), were issued by the NRC on May 15, 1992, and document NRC acceptance of the use of ZIRLO™ clad fuel.

3. Containment Spray Program

The purpose of this program was the development of technical information to substantiate the effectiveness of a chemically reactive spray for removal of fission product iodine from the containment atmosphere following a Loss-of-Coolant Accident.

The program has been completed. Results provided improved modeling capability and showed that the margins of safety inherent in the system design were not significantly diminished.

Reference 4 provides further discussion of these tests.

4. Charcoal Filter Program

An experimental program was conducted at the Oak Ridge National Laboratory (ORNL) at the request of Westinghouse to determine the efficiency of radioactive methyl iodide trapping from flowing steam-air by impregnated charcoal filters.

The program has been completed. The results show that for the conditions expected, the filter configuration used has an initial removal efficiency of at least 70% per pass for all post-accident containment atmosphere environmental conditions up to and including 100% relative humidity. Reference 5 provides further discussion of these tests.

References

1) Eggleston, F. T., “Safety Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries, Spring 1976, ” WCAP-8768, Westinghouse Electric Corp.

2) Roll, J. B., “Performance of Zircaloy Clad Fuel Rods During a Simulated Loss-of-Coolant Accident –Single Rod Tests,” Volume I, WCAP-7805, December 1971; Volume II, WCAP-7379, October 1971; Westinghouse Electric Corp.

3) Schreiber, R., et al, “Performance of Zircaloy Clad Fuel Rods During a Simulated Loss-of-Coolant Accident – Multi Rod Burst Tests,” Volume I, Test Set-up and Results, WCAP-7808 December 1971; Volume II (by C. L. Caso) Analysis of Results, WCAP-7808, December 1971; Westinghouse Electric Corp.

4) Pasedag, W. P., “Iodine Removal by Spray in the Zion Station Containment,” WCAP-7742, August 1971; Westinghouse Electric Corp.

5) Ackley, R. D., and R. E. Adams, “Trapping of Radioactive Methyl Iodine from Flowing Steam-Air: Westinghouse Test Series,” ORNL-TM-2728, October 1968.

6) Safety Evaluation by the Office of Nuclear Reactor Regulation, dated May 15, 1992, Related to Amendment No. 117 to Facility Operating License No. DPR-64.

1.6 IDENTIFICATION OF CONTRACTORS

1.6.1 Design and Construction [Historical]

Prior to December 31, 1975 Consolidated Edison was the sole owner of, and applicant for licenses for, Indian Point 3. As such, Consolidated Edison was solely responsible for the design and construction of the facility.

During the design and construction phase, Consolidated Edison engaged Westinghouse Electric Corporation to design and construct this unit. As prime contractor for Consolidated Edison, Westinghouse Electric Corporation undertook to provide a complete, safe and operable nuclear power plant which was ready for commercial service in 1976. The project was directed by Westinghouse through its wholly owned subsidiary, WEDCO, at the plant site, with project management and engineering liaison in Pittsburgh, Pennsylvania. Westinghouse engaged United Engineers & Constructors Inc. of Philadelphia, Pennsylvania to provide the engineering and architectural design of certain portions of the plant.

Plant construction was under the general direction of Westinghouse through its wholly owned subsidiary, WEDCO, which was responsible for the management of all site construction activities and performed itself, 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 Consolidated Edison personnel under the technical direction of WEDCO, assisted by Westinghouse Electric Corporation as required. The overall functional project organization as it existed during the construction phase is shown in [Historical] Figures 1.6-1 and 1.6-2.

The project management organization was established in the Fall of 1969, slightly modified in the Spring of 1970, and subsequently changed by internal rearrangement and personnel changes. The management approach which was maintained through the end of the construction period is described below.

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

Westinghouse-WEDCO-United Engineers 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:

a) Design and Engineering
b) Procurement
c) Construction Management and Construction
d) Quality Assurance (including Home Office Quality Control Engineering, Vendor Surveillance and On-Site Quality Control)

Westinghouse removed items (b) and (c) and the vendor surveillance and onsite quality control portions of item (d) 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 accomplishing the transition of work to Westinghouse and WEDCO. It should be noted that the UE&C Quality Assurance Program for the Indian Point plants was retained essentially unchanged except that the management and responsibility were made WEDCO functions.

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

In this final organizational structure, WEDCO exercised a high level quality and engineering reliability function. This function included the activities previously performed by the Westinghouse Corporation Nuclear Power Service Staff Resident Quality Assurance Engineer, and included the centralization and overall management for quality assurance activities previously performed by various organizations. This function was carried out by a Reliability Manager based at the site.

The WEDCO Reliability Manager was responsible for vendor surveillance for balance of plant components and for on-site quality control. These functions were previously performed by UE&C. UE&C still retained responsibility for the home office quality control engineering function without change. In addition, the Reliability Manager had reporting to him a systems Reliability Group which audited the home office Quality Assurance efforts of UE&C, and the WEDCO Vendor Surveillance and on-site Quality Control efforts.

Quality control functions previously performed at various Westinghouse organizational levels continued unchanged. At the Westinghouse headquarters level, the staff quality assurance audit team reviewed periodically the quality control program for Indian Point 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.

1.6.2 Operation

On December 24, 1975, the Nuclear Regulatory Commission (NRC) issued Amendment No. 1 to Facility Operating License (FOL) DPR-64 which authorized the Authority to purchase and acquire title to, but not operate Indian Point 3, under the terms of this authorization, Consolidated Edison retained responsibility for the operation of Indian Point 3 with the same operating license restrictions on subcritical operation.

On December 31, 1975, the Authority purchased and acquired title to Indian Point 3, including a portion of the Indian Point Site, buildings, facilities and equipment necessary to support operation of Indian Point 3. Certain systems and facilities, the use of which have been provided for by contract between the Authority and Consolidated Edison, are shared between Indian Point 3 and the other units at the site. With respect to facilities which were not acquired by the Authority, but which are shared by Indian Point 3, the Authority has acquired temporary easements.

Mutual use of the combine site as the restricted area and exclusion area for Indian Point 1, 2 and 3 has also been provided for by a contract which is presently in effect.

On April 6, 1976, NRC issued Amendment No. 2 to FOL DPR-64 permitting full term continuous power operation of Indian Point 3. This Amendment included a number of conditions in power level, environmental, geophysical and safety areas, and by reference, wholly incorporated the Technical Specifications (dated April 15, 1976).

Under the terms of its contract with the Authority, and upon completion of construction, pre-operational tests and initial startup, Consolidated Edison assumed and retained responsibility for plant operation. Services contracted for included operation, quality assurance, engineering, maintenance, training, health physics, water chemistry, environmental monitoring, plant and site security, testing modifications, repair and refueling.

On March 16, 1977, the Authority filed an application with NRC to amend FOL DPR-64 to permit the Authority to assume sole and full responsibility for plant operation and for all future design and construction activities at Indian Point 3. On March 8, 1978, NRC issued Amendment No. 12 to FOL DPR-64 permitting the Authority to assume sole responsibility for operation of Indian Point 3.

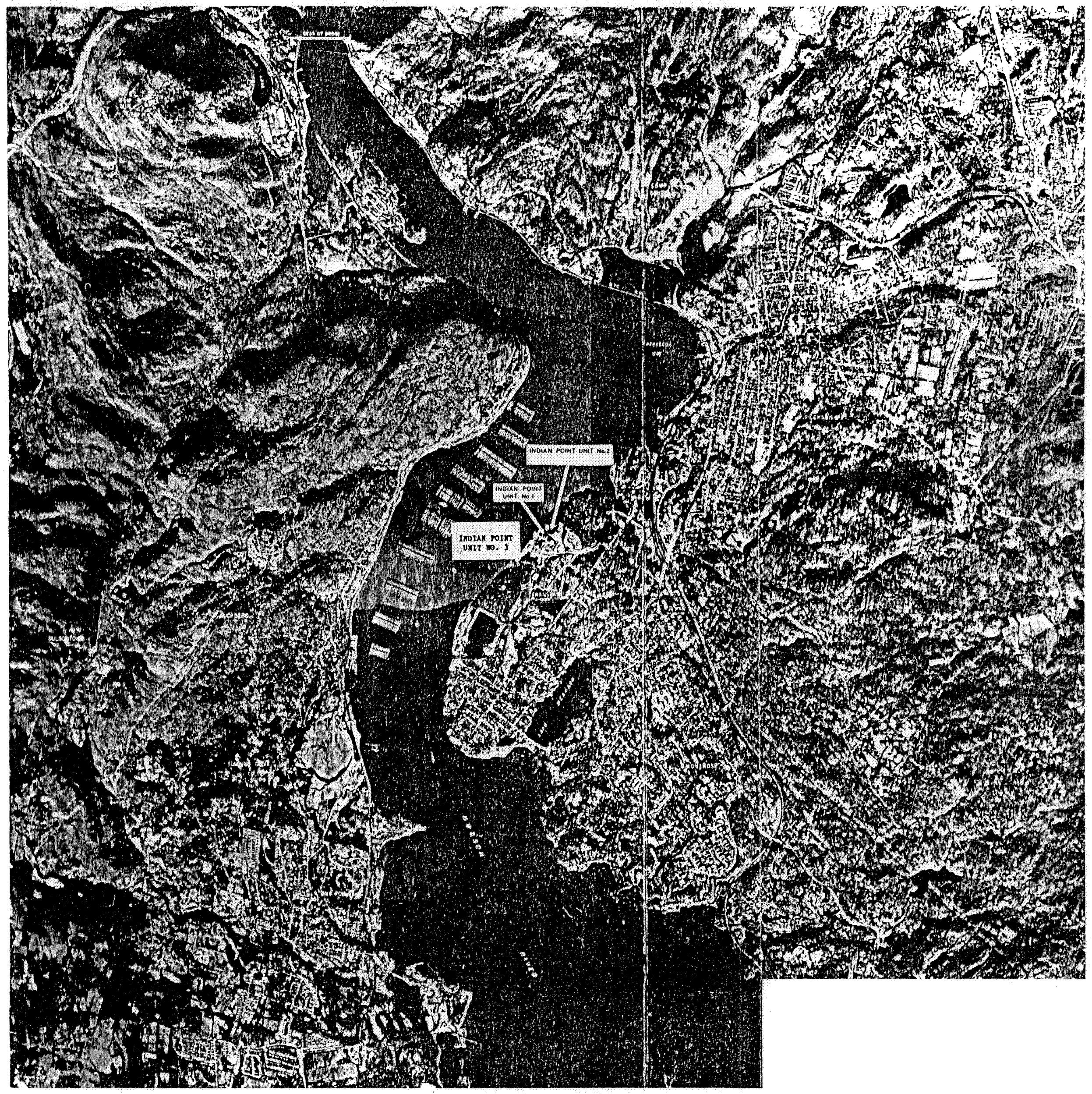
Since that time, the Authority contracts the support of the NSSS Vendor, the plant's original designer and other suitably qualified organizations as required to perform engineering, design, construction and testing services.

The NSSS Vendor and the plant's original designer, Westinghouse Electric Corporation and United Engineers and Constructors Inc., respectively, provide technical support services under the direction of the Authority. These services include: engineering and design services, consulting services, quality assurance and other specialized engineering services as specifically authorized by the Authority. Westinghouse also provides services associated with turbine maintenance and repairs, steam generator inspection, refueling and others as specifically authorized. In addition to Westinghouse and United Engineers, other Contractors are utilized to provide engineering and construction management services associated with major modifications and additional facilities to the plant. Crouse Nuclear Engineering services was retained to provide maintenance and installation services associated with plant improvements.

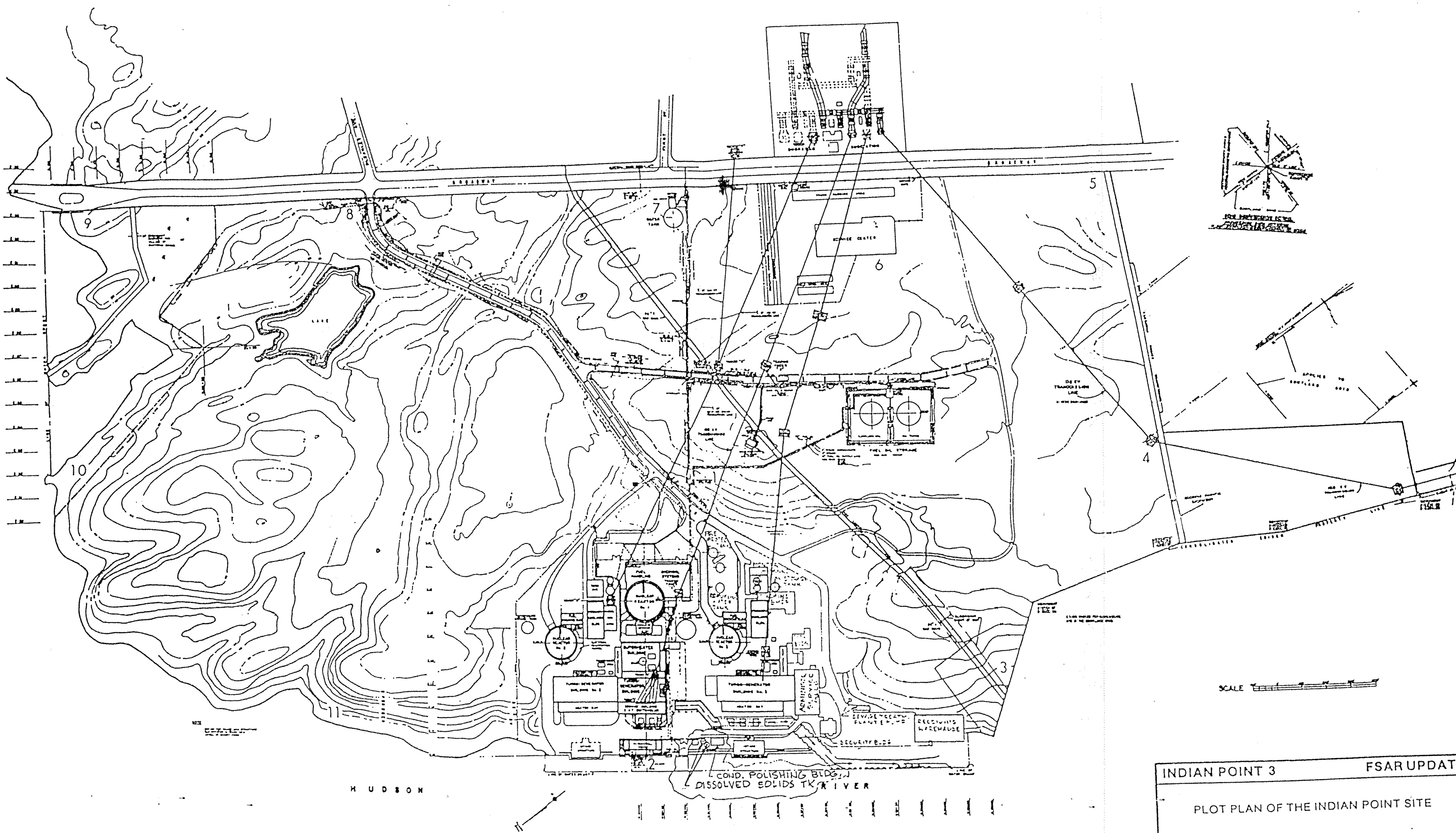
On a 3 - 4 year cycle, in accordance with Authority procedures, on-site maintenance and installation services associated with plant improvements are competitively bid and awarded to a qualified contractor. The overall functional organization chart for the operations phase is shown in [Historical] Figure 1.6-3.

Figure 1.2-3, drawing 9321-F-64513.

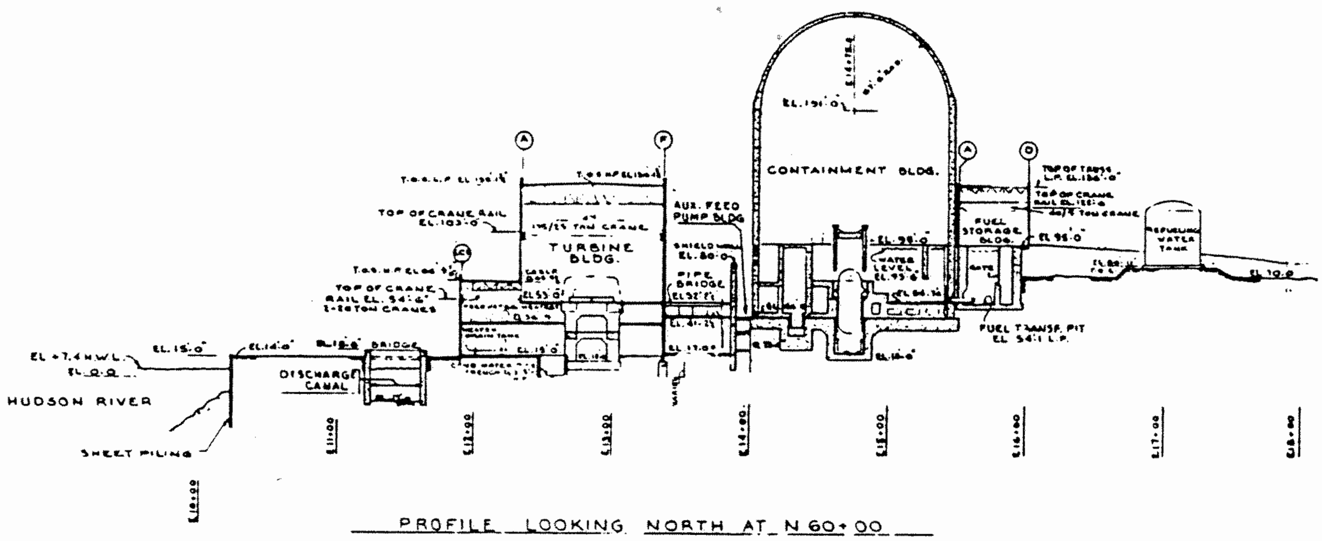
Redacted by NRC staff as sensitive information.



INDIAN POINT 3		FSAR UPDATE	
AERIAL PHOTOGRAPH AT INDIAN POINT SITE AND SURROUNDING AREA			
REV. 0	JULY, 1982	FIGURE NO.	1.2-1



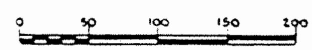
INDIAN POINT 3		FSAR UPDATE
PLOT PLAN OF THE INDIAN POINT SITE		
REV. 1	JULY, 1984	FIGURE NO. 1.2-2



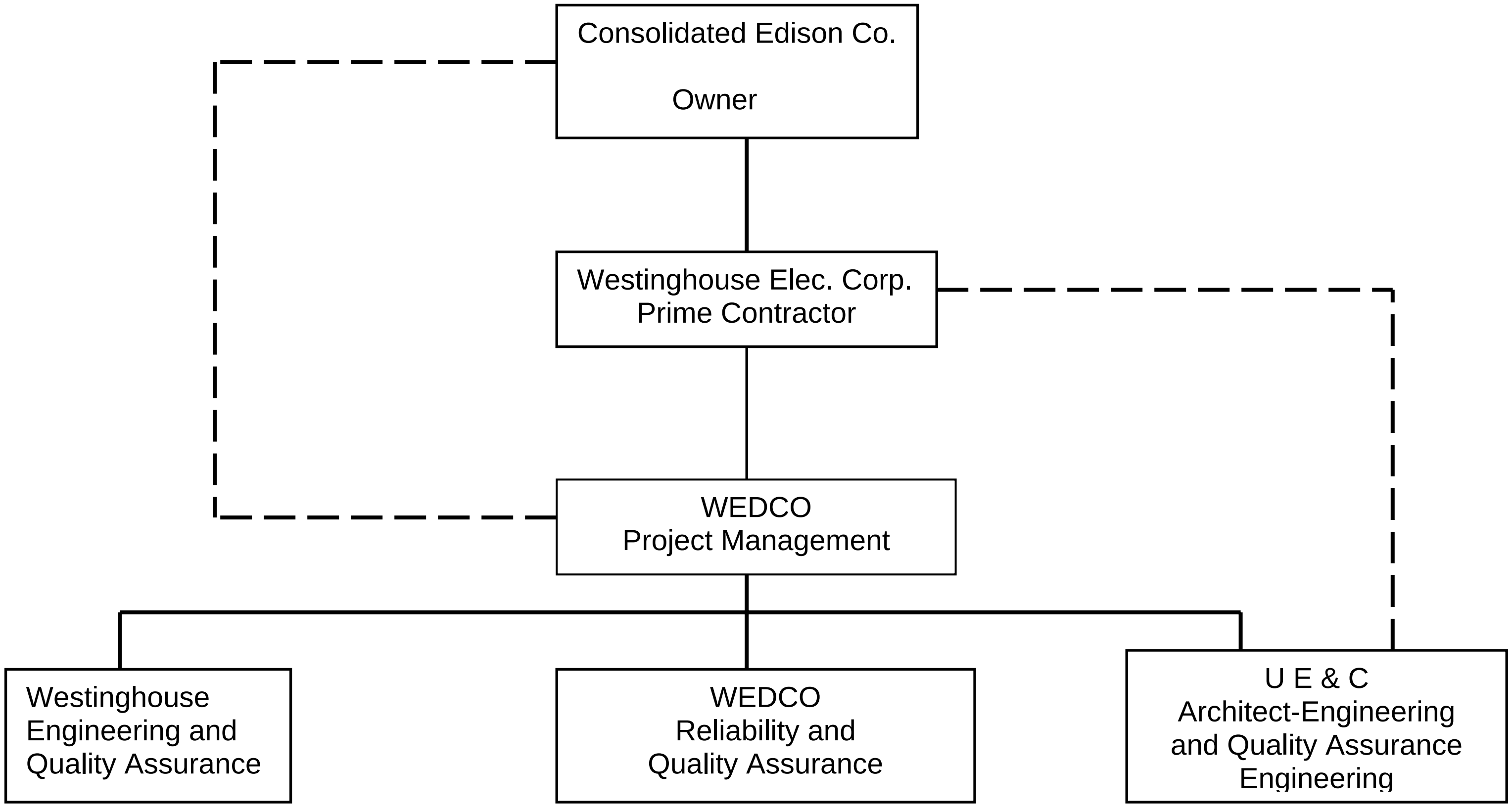
PROFILE LOOKING NORTH AT N 60+00

SCALE: 1" = 50'-0"

NOTE:
DWG. FOR SYSTEMS DESCRIPTION BOOKLET ONLY
NOT A CONSTRUCTION DWG.



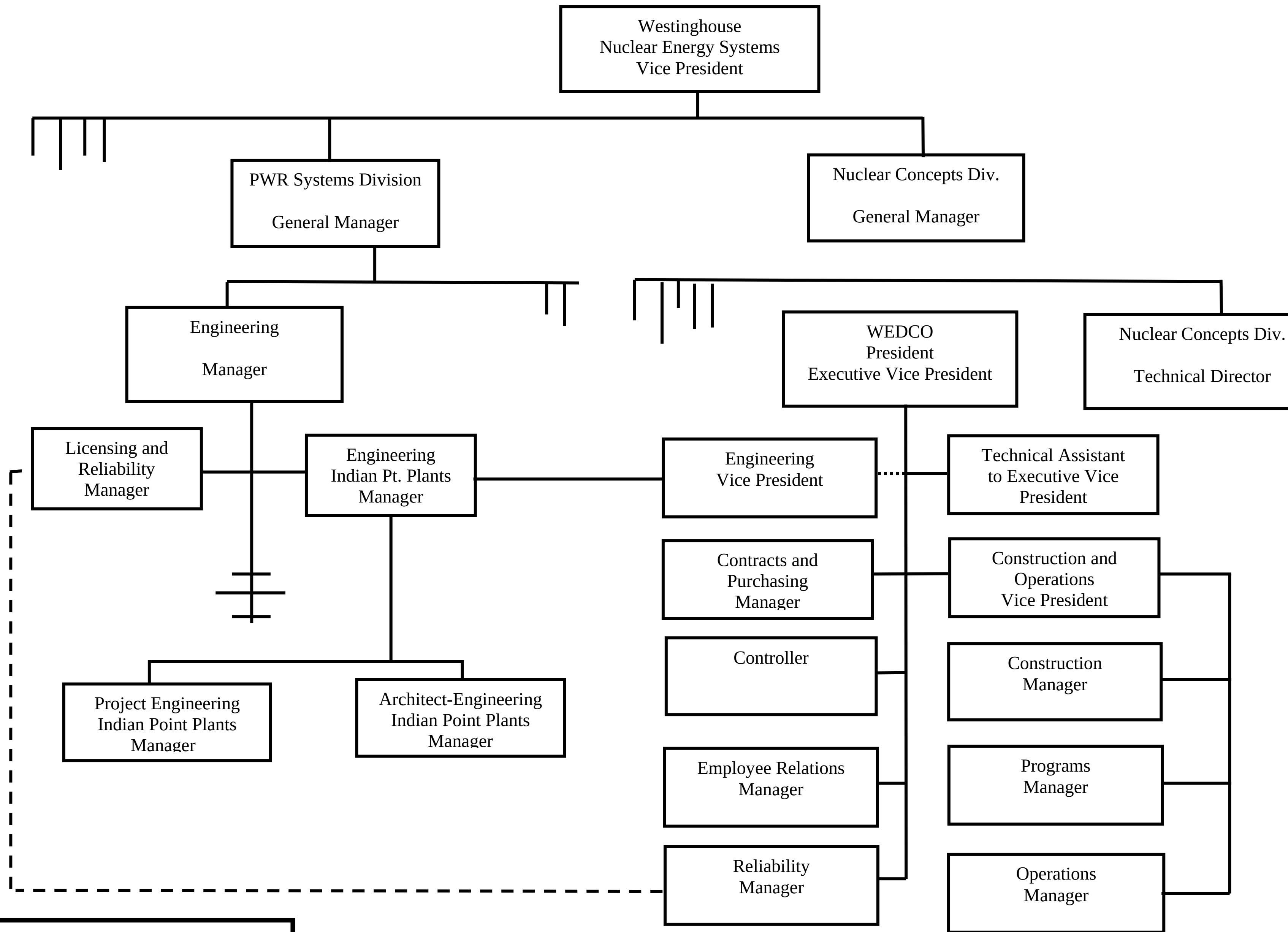
INDIAN POINT 3		FSAR UPDATE
PROFILE LOOKING NORTH THRU TURBINE, CONTAINMENT AND FUEL STORAGE BUILDINGS		
REV. 0	JULY, 1982	FIGURE NO. 1.2-4



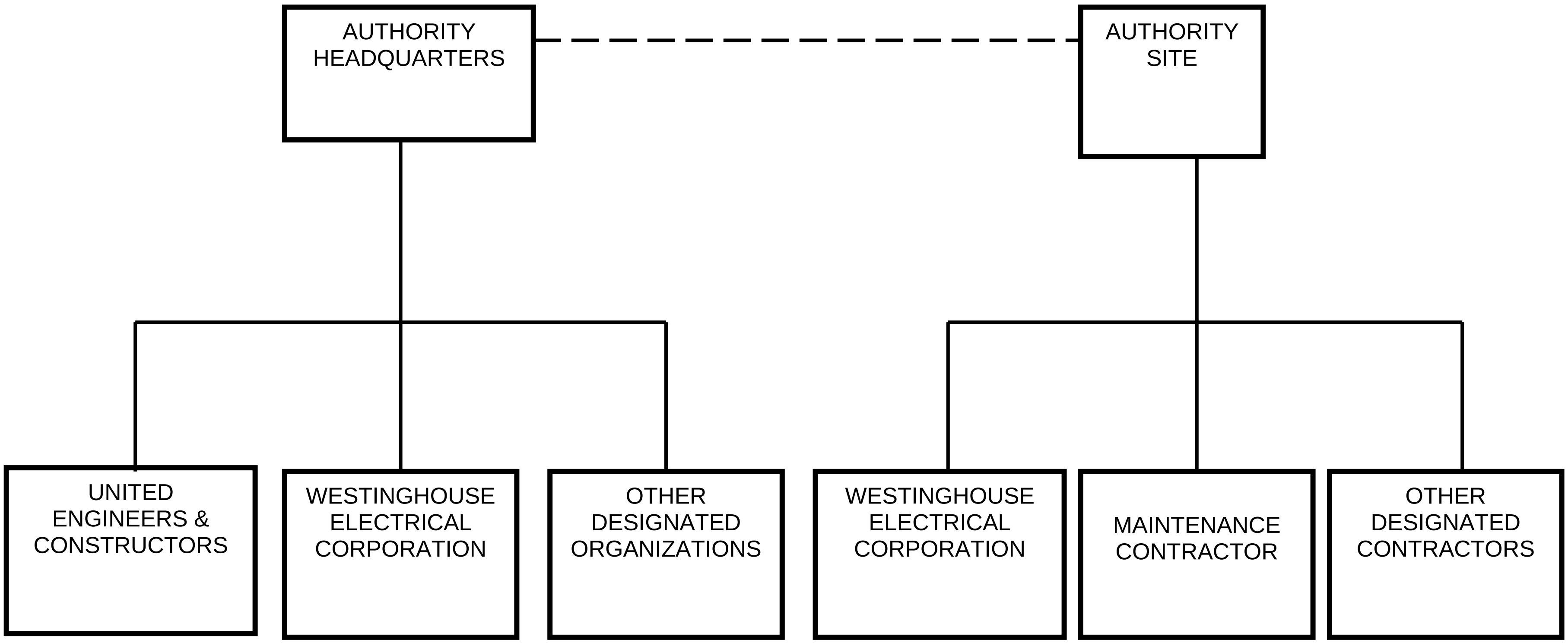
INDIAN POINT 3 FSAR UPDATE

INDIAN POINT 3
CONTRACTOR RELATIONSHIPS
(DESIGN AND CONSTRUCTION)
[Historical Information]

REV. 0 JULY, 1982 FIGURE NO. 1.6-1



INDIAN POINT 3 FSAR UPDATE
WESTINGHOUSE - WEDCO
INDIAN POINT PROJECT ORGANIZATION
(DURING DESIGN AND CONSTRUCTION)
[Historical Information]
REV. 0 JULY 1982 FIGURE NO. 1.6-2



INDIAN POINT 3 FSAR UPDATE

INDIAN POINT 3
CONTRACTOR FUNCTIONALRELATIONSHIPS -
(OPERATING PHASE)
[Historical Information]

REV. 0 JULY, 1982 FIGURE NO. 1.6-3

CHAPTER 2

SITE AND ENVIRONMENT

2.1 SUMMARY OF CONCLUSIONS

This Chapter sets forth the site and environmental data which together formed the basis for many of the criteria for designing the facility and for evaluating the routine and accidental releases 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 due to plant operation. The strength of this conclusion rests with these data and the determinations (also included in this report) of several independent consultants, each speaking within a particular area of expertise - health physics, demography, geology, seismology, hydrology or meteorology, as the case may be.

The task of evaluating the environmental characteristics of the area was facilitated by the fact that more than 12 years of studies and measurements of environmental characteristics were undertaken. For over twenty years, measurements have been made of the effects on the environment of releases from at least one operating nuclear power facility at the Indian Point Site.

Conservative projections have been made of the growth of population in the area and these projections have been taken into account in plant design and operation as to control the effects of accidents. Population estimates are presented in subchapter 2.4.

The census data for 1990 reveals that the population within a 10-mile radius of the site was approximately 238,043 whereas the 2000 estimated population is 564,200. The land is now zoned principally for residential and state park usage although there is some industrial activity and a little agricultural and grazing activity. The projections do not indicate that the land usage within this radius will shift appreciably during the period of plant operation.

Geologically, the site consists of a hard limestone formation in a jointed condition which provides a solid bed for the plant foundation. The bedrock is sufficiently sound to support any loads up to 50 tons per square foot, which is far in excess of any load 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 the 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 peak tidal flow. This flow provides additional mixing and dilution for liquid discharges from the facility. Plant design was based on the conservative assumption that the river water is used for drinking, thus radioactive discharges are reduced by dilution with ordinary plant effluent to concentrations that would be tolerable for drinking water. There is very little danger of flooding at the site.

Significant seismic activity in the Indian Point area is rare and no damage has resulted therefrom. As stated by the consultant on seismology, the site is “practically non-seismic” and is as safe as any area, at present known.” Notwithstanding such assurance, the plant is designed to withstand an earth- quake of the highest intensity ever recorded in this area.

Meteorological conditions in the area of the site were determined during a two-year test program (1955 to 1957). The validity of these conclusions was verified by a test program completed in October 1970. The meteorological analysis also includes data from periods of November 26, 1969 through October 1, 1970, and January 1, 1970 through December 31, 1971. These data were used in evaluating the effects of gaseous discharges from the plant during normal operations and during a 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, have been used to evaluate the rainout of fission gases into surface water reservoirs following a 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 for nearly twenty years in association with the operation of Indian Point 1, and the construction and operation of Indian Point 2 and 3. These measurements are being continued and reported as dictated by the Technical Specifications and ODCM. The radiation measurements of fallout, water samples, vegetation, marine life, etc. have shown no perceptible post-operative increase in radioactivity due to plant operations. 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 two-year post-operative study(1), found that environmental radioactivity in the vicinity of the site is no higher than anywhere else in the State of New York.

Consultants who participated in the preparation of the various reports, mea-surements and conclusions appearing in this Chapter included Dr. Merril Eisenbud, then 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 Halitski, then Senior Research Scientist, Department of Meteorology and Oceanography, New York University, College of Engineering; Dr. Edgar M. Hoover, then at the Regional Economic Development Institute, Inc.; Metcalf & Eddy Engineers, hydrology specialists; Rev. J. J. Lynch, S. J., then Director of the Seismic Observatory, Fordham University; Mr. Sidney Paige, then Consulting Geologist; Quirk, Lawler and Matusky Engineers, Environmental Science and Engineering Consultants; Mr. Karl R. Kennison, Consulting Civil and Hydraulic Engineer; Mr. Thomas W. Fluhr, P.E., Consulting Engineering Geologist; Reports by Captain Elliott B. Roberts, Chief of the Geophysics Division, U.S. Department of Commerce and by Mr. James Dorman, then at the Lamount Geologist Observatory, Columbia University. And also, Reports by Parsons, Brincherhoff, Quade and Douglas Inc., Engineers; Dames & Moore, Consultants; and Woodward-Clyde, Consultants.

References

1. Consolidated Edison Indian Point Reactor Post Operational Survey - August, 1965, Division of Environmental Health Services, New York State Department of Health, Hollis S. Ingraham, M.D., Commissioner.

2. Consolidated Edison Indian Point Reactor Environmental and Post Operation Survey – July, 1966, Division of Environmental Health Services, New York State Department of Health, Hollis S. Ingraham, M.D., Commissioner.

2.2 LOCATION

2.2.1 General

The Indian Point site comprises 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 3 is located adjacent to and south of Unit No. 1 with Indian Point 2 adjacent to and north of Unit No. 1, which has been retired. 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, Figure 2.2-1, shows the site and surrounding area.

The minimum distance from the Indian Point 3 reactor center line to the boundary of the site exclusion area and the outer boundary of the low population zone, as defined in 10 CFE 100.3 and 10 CFR 100.11, is 350 meters and 1100 meters, respectively.

2.2.2 Site Ownership and Control

Entergy is the sole owner of the Indian Point 3 Nuclear Power Plant. Figure 2.2-2 shows the land owned by Entergy at the Indian Point site. Plant Drawing 9321-F-64513 [Formerly Figure 2.2-3] shows a plot plan of Indian Point 3 and the boundary line and the Hudson River. Figure 2.2-4 shows the Indian Point Energy Center site ownership boundaries, the location of surrounding communities, and the Low Population Zone for Indian Point 3.

The Georgia – Pacific Corporation has an easement, (approximately 1610 feet long and 30 feet wide), along Entergy’s southerly property line. The Georgia – Pacific easement is used for overhead electrical power and telephone lines, and for 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.

As shown in Figure 2.2-2, the Algonquin Gas Transmission Company has a 24 inch gas mainline and a 30 inch loop line on a right-of-way (approximately 1350 feet long and 65 feet wide) running east to west through Entergy’s property. The threats posed by the rupture of these pipelines and the release of natural gas (essentially methane) from them were addressed in Item 7 of Supplement 1 to the original FSAR. The September 21, 1973 SER concluded the failure of these gas lines would not impair the safe operation of the plant.

A subsequent evaluation in 2008, (Reference 1), discussed the consequences of a pipeline rupture and the potential impact of that event on the sites Protected Area, Vital Areas, the Security Plan, safe shutdown, and other non-safety related structures, such as the waterfront warehouse. The hazards created by a breach and explosion of the pressurized above ground portions of the pipeline include:

a. potential missiles,
b. an over-pressurization event,
c. a vapor cloud or flash fire,
d. a hypothetical vapor cloud explosion, and
e. a jet fire.

A simultaneous rupture and ignition of both gas mains at the above ground locations inside the owner controlled area (OCA) is postulated to be the worst case scenario since this event will result in the most significant release of gas volume and have the potential to contribute to the largest potential fire. An attempt to uncover, breach and ignite a buried portion of the pipeline was not considered feasible. The report concluded that the event would not damage any safety related structure and there are no adverse effects on the gas pipeline event on vital areas, safe shutdown equipment, IPEC Security Plan, or essential personnel. Some damage to non-vital structures or non-essential personnel in the area of the pipeline may occur.

The Indian Point 3 protected area is enclosed by a chain link type security fence surmounted by three-strand barbed wire as indicated in Entergy's Security Plan. Appropriate control is maintained by Entergy at all access points into the Indian Point 3 security protected area. In addition, some areas within the protected area are designated as "vital areas" and access is controlled by a system of identification badges/card keys, locks and alarms.

Employees, who would need access to or through any portions of Entergy property, are required to adhere to the security provisions and check points operated and controlled by Entergy's security force. Details of Entergy's security program are given in the Security Plan for Indian Point Energy Center.

2.2.3 Access

The site is accessible by several roads in the Village of Buchanan. Two paved roads link the eastern boundary of the site to the exiting plant. The site is not served by rail. The Indian Point 3 protected area is bounded by chain link-type fencing, or the equivalent, and contains an interior roadway system, access to which is under the control of Entergy's Guard Force.

2.2.4 Control of Exclusion and Restricted Area

In the event of an emergency situation at any of the Indian Point Units, the person in charge at that Unit shall immediately inform the person in charge at the other units. Further action, evaluation or institution of the offsite emergency plan will then depend on the seriousness of the emergency situation. Further details are provided in the Indian Point Energy Center Emergency Plan.

Control of the Indian Point Site Restricted Area to exclude unauthorized personnel at all times is controlled by IPEC. IPEC has responsibility for maintaining direct and continuous control over the persons on its property.

2.2.5 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, utilization and production of special nuclear, source and by-product materials.

2.3 TOPOGRAPHY

The Indian Point Site is surrounded on almost all sides by high ground ranging from 600 to 1000 feet above sea level. The site is located on the east bank of the Hudson River, which runs northeast to southwest at this point but turns sharply northwest approximately two 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 fee and 1257 feet respectively) and Buckberg Mountain to the west-southwest (elevation 793 feet). These peaks extend to the west by other names and gradually rise to slightly higher peaks.

The general orientation of this mass of high ground is northeast to southwest. One mile northwest of the site, Dunderberg bulges to the east, and north of Dunderberg and the site, high ground reaching 800 feet forms the east bank of the Hudson as the 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 feet in height and there is a weak, poorly defined series of ridges which again seem to run in a north-northeast direction. The river south of the site makes another sharp bend to the southeast and the widens as it flows past Croton and Haverstraw.

An aerial photograph showing these topographic features of the site and surrounding area is shown in Figure 2.2-1.

2.4 POPULATION

2.4.1 General

An initial report was prepared by Environmental Analysis, Inc. in June 1972. The report, which is included herein (see pages 2.4.P-1 to 2.4.P-42), used the 1970 population census to update population estimates and population projections to the year 2010, in 10-year intervals, for an area within a sixty-mile radius of the Indian Point Nuclear Power Plant Site at Buchanan, New York.

The resident population distribution, based on a May 1970 report was presented graphically by indicating the population estimated for each of the individual area segments of the selected grid system. The population distribution for a 55 mile radius is presented in Figure 2.4-1. The grid was centered on the reactor facility and extended radially for a distance of 55 miles. This area was subdivided by concentric circles with radii of 15, 25, 35, 45, and 55 miles, and by equally spaced radial lines.

The grid system, superimposed on the geographical area surrounding the site, for a 60-mile radius, is shown in Figure 2.4-2. Each sector of 22.5^{o} was based on a line segment 11.25^{o} from north (resulting in two subdivisions of the north bisected 22.5o sector); the others were designed in a clockwise fashion. The number of persons residing in each segment, based on the 1972 report, is presented in Figures 2.4-3 and 2.4-4 for the 5-mile and 60-mile radii, respectively.

Table 2.4-1, based on the June 1972 report, is a summary of the cumulative ring population estimates for the years 1970 to 2010, in 10 year increments, for complete ring zones up to sixty miles from the site.

In 1992, the Authority submitted new demographic and population distribution data on the Nuclear Regulatory Commission as part of its License Extension Request. This data shows that the projections performed in 1972 by Environmental Analysts were quite realistic, although slightly conservative. The rate of growth within the 50 mile radius was slightly slower than projected. Projections through 2010 should, thus, be viewed as conservative.

In 2003, KLD Associates, Inc. updated population distribution data for the 50 mile radius surrounding Indian Point using 2000 Census data. Table 2.4-2 summarizes this data by Zone.

Tables 2.4-3 through 2.4-18 show population distribution by sector and zone. Table 2.4-19 shows population by segment. A comparison of the 1972 projections with 2000 Census data continues to show that projections through 2010 should be viewed as conservative.

A comparison of the 1972 projections found in Table 2.4-1 with 1990 and 2000 Census data is shown below:

	1990 Projection from 1972 Study	1990 Census	2000 Projection from 1972 Study	2000 Census
0-2 miles	15,673	16,774	20,698	12,442
0-5 miles	84,512	73,935	129,397	77,619
0-10 miles	408,198	237,338	564,220	257,475

Projections through 2010 should thus be viewed as conservative.

2.4.2 Population Centers

The closest population centers (defined in 10 CFR 100 as containing more than 25,000 residents) are Newburgh, N.Y. and White Plains, N.Y., both approximately 17 miles from the plant site. However, based on projected populations, the outer boundary of the more densely populated area of the City of Peekskill has been conservatively selected as the closest population center.

2.4.3 Low Population Zone

The Code of Federal Regulations, Title 10, Part 100 requires that a reactor be so situated that there is no population center, which is defined as a city of no less than 25,000 people, having its nearest boundary closer than 1-1/3 times the low population zone radius. Based on the 10 CFR 100 definition, and the outer boundary of the more densely populated area of the City of Peekskill as the population center, the low population zone (LPZ) for the plant is 1100 meters. (See Figure 2.2-4)

About 50 people reside within the low population zone, all of them to the east-southeast. This estimate of 50 people (a number which is expected to remain fairly static) is based on a survey of the area conducted by Consolidated Edison in September 1971.

2.4.4 Exclusion Area

The exclusion area for Indian Point 3 is shown in Figure 2.2-2. The minimum distance from the reactor containment to the boundary of the exclusion area is 350 meters. This exclusion area satisfied both 10 CFR 100.3 and 10 CFR 100.11.

2.4.5 Land Usage

Figures 2.4-6, 2.4-7, and 2.4-8 show, respectively, the land usage based upon official zoning maps, areas served by public utilities and areas served by sewage systems. The area surrounding the Indian Point Site is generally residential with some large parks and military reservations. The majority of the area to the east of the river within 15 miles of the site is zoned for residential usage as shown on the map in Figure 2.4-6. West of the river, within a fifteen-mile radius, the Palisades Interstate Park and residential areas are the dominant land usage. The only agricultural areas within fifteen miles are south and northwest of the plant, on the west side of the river.

References

1. IP-PRT-08-00032, “Consequences of Fire and Explosion Following the Release of Natural Gas from Pipelines Adjacent to Indian Point”, by David Allen, Risk Research Group, August 2008.

Table 2.4-1

SUMMARY OF CUMULATIVE RING POPULATION ESTIMATES
(JUNE 1972)

Radius of the Ring in Miles	Cumulative Ring Population Estimates				
	1970	1980	1990	2000	2010
Half	21	31	45	65	88
One	745	1,008	1,375	1,891	2,453
Two	9,255	11,981	15,673	20,698	26,016
Three	20,318	25,747	33,045	42,926	53,349
Four	34,553	44,338	57,544	75,482	94,451
Five	52,683	70,053	94,512	129,397	168,164
Ten	218,398	297,459	408,198	564,220	734,682
Fifteen	450,207	603,034	814,078	1,107,195	1,423,387
Twenty	888,163	1,179,611	1,577,851	2,125,429	2,711,048
Thirty	3,984,844	4,637,627	5,480,207	6,584,630	7,724,505
Forty	11,659,574	12,882,240	14,403,268	16,333,563	18,276,655
Fifty	17,471,479	18,991,980	20,923,966	23,400,331	25,899,727
Sixty	19,510,656	21,383,172	23,821,556	26,997,743	30,235,074

Table 2.4-2

2000 Population Estimates By 360 Degree Zone

ZONE	2000 Population	Accumulated Totals
0-1 mile	1,971	
1-2 miles	10,471	
2-3 miles	19,516	
3-4 miles	18,791	
4-5 miles	26,870	77,619 (within 5 miles)
5-6 miles	27,674	
6-7 miles	21,404	
7-8 miles	25,688	
8-9 miles	49,767	
9-10 miles	55,323	257,475 (within 10 miles)
10-15 miles	398,447	
15-20 miles	460,697	
20-25 miles	1,116,848	2,233,467 (within 25 miles)
25-30 miles	2,205,078	
30-35 miles	2,544,937	
35-40 miles	3,734,393	
40-45 miles	3,833,427	
45-50 miles	2,232,598	16,783,900 (within 50 miles)

Table 2.4-3

2000 Population Estimates
Population Distribution By Sector and Zone

SECTOR: 1 (North)

Zone	2000 Population
0-1 mile	0
1-2 miles	11
2-3 miles	109
3-4 miles	183
4-5 miles	343
5-6 miles	300
6-7 miles	1,322
7-8 miles	2,395
8-9 miles	7,460
9-10 miles	182
10-15 miles	1,366
15-20 miles	43,748
20-25 miles	32,751
25-30 miles	54,348
30-35 miles	48,971
35-40 miles	18,752
40-45 miles	20,142
45-50 miles	41,358

Table 2.4-4
2000 Population Estimates
Population Distribution By Sector and Zone

SECTOR: 2 (North-Northeast)

Zone	2000 Population
0-1 mile	0
1-2 miles	0
2-3 miles	314
3-4 miles	661
4-5 miles	1,716
5-6 miles	921
6-7 miles	1,044
7-8 miles	340
8-9 miles	838
9-10 miles	1,235
10-15 miles	2,453
15-20 miles	4,967
20-25 miles	23,111
25-30 miles	22,593
30-35 miles	8,417
35-40 miles	10,711
40-45 miles	8,153
45-50 miles	2,859

Table 2.4-5

2000 Population Estimates
Population Distribution By Sector and Zone

SECTOR: 3 (Northeast)

Zone	2000 Population
0-1 mile	17
1-2 miles	3,439
2-3 miles	9,774
3-4 miles	4,510
4-5 miles	2,973
5-6 miles	3,823
6-7 miles	3,356
7-8 miles	1,760
8-9 miles	1,196
9-10 miles	1,097
10-15 miles	14,549
15-20 miles	14,593
20-25 miles	14,498
25-30 miles	11,974
30-35 miles	18,004
35-40 miles	18,775
40-45 miles	6,511
45-50 miles	7,254

Table 2.4-6

2000 Population Estimates
Population Distribution By Sector and Zone

SECTOR: 4 (East-Northeast)

Zone	2000 Population
0-1 mile	7
1-2 miles	1,653
2-3 miles	2,841
3-4 miles	3,238
4-5 miles	2,178
5-6 miles	3,683
6-7 miles	2,473
7-8 miles	4,797
8-9 miles	6,936
9-10 miles	6,915
10-15 miles	24,469
15-20 miles	13,258
20-25 miles	20,375
25-30 miles	89,902
30-35 miles	35,294
35-40 miles	21,347
40-45 miles	22,265
45-50 miles	78,210

Table 2.4-7

2000 Population Estimates
Population Distribution By Sector and Zone

SECTOR: 5 (East)

Zone	2000 Population
0-1 mile	334
1-2 miles	315
2-3 miles	24
3-4 miles	950
4-5 miles	594
5-6 miles	620
6-7 miles	1,545
7-8 miles	1,355
8-9 miles	3,224
9-10 miles	3,426
10-15 miles	15,387
15-20 miles	8,093
20-25 miles	26,346
25-30 miles	22,542
30-35 miles	22,092
35-40 miles	172,384
40-45 miles	167,423
45-50 miles	88,353

Table 2.4-8

2000 Population Estimates
Population Distribution By Sector and Zone

SECTOR: 6 (East-Southeast)

Zone	2000 Population
0-1 mile	251
1-2 miles	192
2-3 miles	757
3-4 miles	656
4-5 miles	918
5-6 miles	304
6-7 miles	75
7-8 miles	319
8-9 miles	626
9-10 miles	2,113
10-15 miles	19,280
15-20 miles	11,355
20-25 miles	34,346
25-30 miles	141,922
30-35 miles	61,824
35-40 miles	18,609
40-45 miles	3,254
45-50 miles	29,437

Table 2.4-9

2000 Population Estimates
Population Distribution By Sector and Zone

SECTOR: 7 (Southeast)

Zone	2000 Population
0-1 mile	807
1-2 miles	922
2-3 miles	1,543
3-4 miles	2,490
4-5 miles	694
5-6 miles	4,590
6-7 miles	2,630
7-8 miles	3,004
8-9 miles	10,085
9-10 miles	6,001
10-15 miles	37,224
15-20 miles	18,930
20-25 miles	101,556
25-30 miles	37,702
30-35 miles	12,330
35-40 miles	80,302
40-45 miles	239,071
45-50 miles	309,332

Table 2.4-10

2000 Population Estimates
Population Distribution By Sector and Zone

SECTOR: 8 (South-Southeast)

Zone	2000 Population
0-1 mile	435
1-2 miles	2,042
2-3 miles	868
3-4 miles	136
4-5 miles	217
5-6 miles	0
6-7 miles	90
7-8 miles	0
8-9 miles	3,864
9-10 miles	9,817
10-15 miles	21,348
15-20 miles	116,963
20-25 miles	218,703
25-30 miles	295,031
30-35 miles	258,202
35-40 miles	603,637
40-45 miles	756,484
45-50 miles	464,715

Table 2.4-11

2000 Population Estimates
Population Distribution By Sector and Zone

SECTOR: 9 (South)

Zone	2000 Population
0-1 mile	68
1-2 miles	541
2-3 miles	0
3-4 miles	0
4-5 miles	1,229
5-6 miles	5,661
6-7 miles	942
7-8 miles	4,716
8-9 miles	7,829
9-10 miles	7,864
10-15 miles	45,274
15-20 miles	47,358
20-25 miles	354,441
25-30 miles	926,582
30-35 miles	1,620,749
35-40 miles	2,099,064
40-45 miles	1,934,401
45-50 miles	743,893

Table 2.4-12

2000 Population Estimates
Population Distribution By Sector and Zone

SECTOR: 10 (South-Southwest)

Zone	2000 Population
0-1 mile	52
1-2 miles	604
2-3 miles	420
3-4 miles	2,853
4-5 miles	10,900
5-6 miles	5,970
6-7 miles	3,378
7-8 miles	3,778
8-9 miles	6,101
9-10 miles	10,856
10-15 miles	106,089
15-20 miles	68,692
20-25 miles	160,698
25-30 miles	437,592
30-35 miles	325,993
35-40 miles	529,035
40-45 miles	452,790
45-50 miles	351,395

Table 2.4-13

2000 Population Estimates
Population Distribution By Sector and Zone

SECTOR: 11 (Southwest)

Zone	2000 Population
0-1 mile	0
1-2 miles	0
2-3 miles	2,115
3-4 miles	2,486
4-5 miles	3,853
5-6 miles	626
6-7 miles	4,496
7-8 miles	3,090
8-9 miles	1,388
9-10 miles	2,955
10-15 miles	17,690
15-20 miles	40,837
20-25 miles	48,470
25-30 miles	69,388
30-35 miles	49,562
35-40 miles	98,399
40-45 miles	139,699
45-50 miles	58,048

Table 2.4-14

2000 Population Estimates
Population Distribution By Sector and Zone

SECTOR: 12 (West-Southwest)

Zone	2000 Population
0-1 mile	0
1-2 miles	457
2-3 miles	518
3-4 miles	628
4-5 miles	221
5-6 miles	433
6-7 miles	48
7-8 miles	134
8-9 miles	0
9-10 miles	5
10-15 miles	5,052
15-20 miles	4,381
20-25 miles	17,860
25-30 miles	15,165
30-35 miles	23,446
35-40 miles	22,991
40-45 miles	52,496
45-50 miles	27,668

Table 2.4-15

2000 Population Estimates
Population Distribution By Sector and Zone

SECTOR: 13 (West)

Zone	2000 Population
0-1 mile	0
1-2 miles	295
2-3 miles	154
3-4 miles	0
4-5 miles	0
5-6 miles	0
6-7 miles	0
7-8 miles	0
8-9 miles	0
9-10 miles	20
10-15 miles	6,164
15-20 miles	11,313
20-25 miles	17,014
25-30 miles	9.836
30-35 miles	18,522
35-40 miles	15,831
40-45 miles	11,592
45-50 miles	5,663

Table 2.4-16

2000 Population Estimates
Population Distribution by Sector and Zone

SECTOR: 14 (West-Northwest)

Zone	2000 Population
0-1 mile	0
1-2 miles	0
2-3 miles	19
3-4 miles	0
4-5 miles	0
5-6 miles	0
6-7 miles	0
7-8 miles	0
8-9 miles	217
9-10 miles	1,633
10-15 miles	36,983
15-20 miles	10,970
20-25 miles	14,136
25-30 miles	47,249
30-35 miles	12,119
35-40 miles	6,654
40-45 miles	3,682
45-50 miles	6,307

Table 2.4-17

2000 Population Estimates
Population Distribution By Sector and Zone

SECTOR: 15 (Northwest)

Zone	2000 Population
0-1 mile	0
1-2 miles	0
2-3 miles	60
3-4 miles	0
4-5 miles	192
5-6 miles	0
6-7 miles	0
7-8 miles	0
8-9 miles	3
9-10 miles	1,204
10-15 miles	9,477
15-20 miles	10,441
20-25 miles	12,645
25-30 miles	10,588
30-35 miles	15,017
35-40 miles	9,956
40-45 miles	6,814
45-50 miles	14,385

Table 2.4-18

2000 Population Estimates
Population Distribution By Sector and Zone

SECTOR: 18 (North-Northwest)

Zone	2000 Population
0-1 mile	0
1-2 miles	0
2-3 miles	0
3-4 miles	0
4-5 miles	842
5-6 miles	743
6-7 miles	5
7-8 miles	0
8-9 miles	0
9-10 miles	0
10-15 miles	35,642
15-20 miles	34,798
20-25 miles	19,898
25-30 miles	12,664
30-35 miles	14,395
35-40 miles	7,946
40-45 miles	8,650
45-50 miles	3,721

TABLE 2.4-19

PERMANENT RESIDENT
POPULATION BY SEGMENT

Segment	Ring (Miles)	Residential Population
1-N	0-2	11
2-NNE	0-2	0
3-NE	0-2	3,456
4-ENE	0-2	1,660
5-E	0-2	649
6-ESE	0-2	443
7-SE	0-2	1,729
8-SSE	0-2	2,477
9-S	0-2	609
10-SSW	0-2	656
11-SW	0-2	0
12-WSW	0-2	457
13-W	0-2	295
14-WNW	0-2	0
15-NW	0-2	0
16-NNW	0-2	0
Ring 0-2 Miles		12,442
1-N	2-5	635
2-NNE	2-5	2,691
3-NE	2-5	17,257
4-ENE	2-5	8,257
5-E	2-5	1,568
6-ESE	2-5	2,331
7-SE	2-5	4,727
8-SSE	2-5	1,221
9-S	2-5	1,229
10-SSW	2-5	14,173
11-SW	2-5	8,454
12-WSW	2-5	1,367
13-W	2-5	154
14-WNW	2-5	19
15-NW	2-5	252
16-NNW	2-5	842
Ring 2-5 Miles		65,177
N	5-10	11,659
NNE	5-10	4,378
NE	5.10	11,232
ENE	5-10	24,804
E	5-10	10,170
ESE	5-10	3,437
SE	5-10	26,310
SSE	5-10	13,771

S	5-10	27,012
SSW	5-10	30,083
SW	5-10	12,555
WSE	5-10	620
W	5-10	20
WNW	5-10	1,850
NW	5-10	1,207
NNW	5-10	748
Ring 5-10 Miles		163,403

Cumulative Totals	
Total 0-2 Miles	12,442
Total 0-5 Miles	77,619
Total 0-10 Miles	257,475

[Historical Information]

POPULATION ESTIMATES FOR 1970
and
THE POPULATION PROJECTIONS TO 2010
for
SPECIFIED ZONES WITHIN A SIXTY-MILE RADIUS
of
INDIAN POINT NUCLEAR POWER PLANT SITE

FOR

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2.4.P-1

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INTRODUCTION

This report provides the 1970 population estimates and the population projections for the years 1980, 1990, 2000 and 2010 for the area within a sixty-mile radius of the Indian Point Nuclear Power Plant site at Buchanan, New York.

The area encompassed by the sixy-mile radius circle was divided into 16 sectors of 22° - 30'. The north oriented sector is formed by two radii, 11° -15' on either side or the true north as shown in Figure 1. This sector is referred to as sector "A" and succeeding sectors, B through P are drawn in the clockwise direction.

The area within the sixty-mile radius was further divided by 13 rings drawn about the Indian Point Nuclear Power Plant site as follows:

Two rings, each at a half-mile interval, for the first mile from the site.

Four rings, each at a one-mile interval, from one mile to five miles from the site.

Three rings, each at a five-mile interval, from five miles to twenty miles from the site.

Four rings, each at a ten-mile interval, from twenty to sixty miles from the site.

The population estimates for the year 1970 and the population projection for the years 1980, 1990, 2000 and 2010 for each of the 208 zones formed by the sectors and the rings are given in this report.

The summary of cumulative ring population estimations for the years 1970, 1980, 2990, 200 and 2010 is given in Table 1.

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Figure 1. Ring and Sector Orientation

Table I

SUMMARY OF CUMULATIVE RING POPULATION ESTIMATES

RADIUS OF THE RING IN MILES — CUMULATIVE RING POPULATION ESTIMATES

	1970	1980	1990	2000	2010
Half	21	31	45	65	88
One	745	1,008	1,375	1,891	2,453
Two	9,255	11,981	15,673	20,698	26,016
Three	20,318	25,747	33.045	42,926	53,349
Four	34,553	44,338	57,544	75,482	94,451
Five	52,683	70,053	94,512	129,397	168,164
Ten	218,398	297,459	408,198	564,220	734,682
Fifteen	450,207	603,035	814,078	1,107,195	1,423,387
Twenty	888,163	1,179,611	1,577,851	2,125,429	2,711,048
Thirty	3,984,844	4,637,627	5,480,207	6,584,630	7,724,505
Forty	11,659,574	12,882,240	14,403,268	16,333,563	18,276,655
Fifty	17,471,479	18,991,980	20,923,966	23,400,331	25,899,727
Sixty	19,510,656	21,383,172	23,821,556	26,997,743	30,235,074

II. 1970 POPULATION ESTIMATION METHODS

The area within a 60-mile radius is divided by rings and sectors, into 208 zones. For the purpose of estimating 1970 population, the zones were divided into three categories. The first category included the 32 zones within the initial one-mile radius. The second consisted of the 64 zones between the one and five-mile radii. The third category consisted of the remaining 112 zones between the five and 60-mile radii.

The zones in the first category are relatively small. Those within a half-mile radius of the Indian Point have an area of approximately 0.05 square miles. The land area of the zones between the one-half mile and one-mile radii is approximately 0.15 square miles. There is a substantial possibility of error in estimating population for these small zones because census data on such a fine scale is not always available. For this reason, Consolidated Edison made a door-to-door survey to determine the exact population within a one-half mile radius of the site. In addition, a field observation of the area within a one-mile radius, including an actual count of dwelling units, was made on January 26, 1972. The population within one mile of the site was estimated on the basis of the data collected by Consolidated Edison, the field observations, and 1970 census tracts shown in the New York-Northeastern New Jersey Metropolitan Map Series.

The zones in the second category are somewhat larger. Their land area ranges from approximately 0.6 square miles, for the zones between the one and two mile radii, to approximately 1.8 miles for the zones between the four and five mile radii. Where census data was available for tracts or communities within zones, these data were used. Elsewhere within the second category zones, population was estimated by use of maps and field inspection. In localities where large areas are fully developed with single-family dwellings, the total length of residential streets was measured by the use of a “map reader” The street length was then divided by 100 feet, which was the average plot width observed during the field survey, and multiplied by 3.5 persons per household, a figure obtained from the Bureau of the Census.

The zones in the third category are five to 60 miles form the plant site. The land area of each zone ranges from approximately 15 square miles, for zones between the 5 and 10 mile radii, to 216 square miles for zones between the 50 and 60 mile radii.

Because the outermost zones are so large, some villages, towns, cities, etc. are entirely located within a single zone. Therefore, the entire population of these communities, taken from the census population tables, could be ascribed to these zones.

For communities or census tracts located in more than one zone, the population was assumed to be distributed uniformly. The portion of the community or tract lying within each zone was determined by the use of a planimeter and a corresponding portion of its population was then attributed to that zone.

It should be emphasized that, for any given zone of the third category, the bulk of the population estimate is based on whole-tract or whole-community figures taken directly from the 1970 census tables. The component of the population estimate based on a real measurement is relatively minor. Therefore, the margin of error in these population estimates is considered to be small.

III. POPULATION ESTIMATES FOR THE 1970

The population estimates for the 13 rings are presented in Table 2 and the population estimates for the 208 specified zones are presented in Table 3.

Table 2

Ring Population Estimates for the Year 1970

Radius of the Ring in Miles	1970 Population Estimates
Half	21
One	724
Two	8,460
Three	11,063
Four	14,235
Five	18,130
Ten	165,715
Fifteen	231,809
Twenty	437,956
Thirty	3,096,681
Forty	7,674,730
Fifty	5,811,905
Sixty	2,039,177

Table 3

SPECIFIED ZONE POPULATION

BASED ON 1970 CENSUS

RING	SECTOR A	SECTOR B	SECTOR C	SECTOR D
0- ½ MILE	0	0	0	0
½- 1 MILE	0	0	0	18
1 - 2 MILES	0	0	1,050	1,470
2 - 3 MILES	158	840	2,880	1,890
3 - 4 MILES	280	875	2,135	1,855
4 - 5 MILES	525	2,240	2,660	2,100
5- 10 MILES	7,451	2,072	4,372	14,880
10 - 15 MILES	6,598	2,775	6,714	5,560
15 - 20 MILES	25,952	4,349	9,110	10,821
20 - 30 MILES	59,527	29,306	13,369	19,730
30 - 40 MILES	78,736	17,647	22,693	86,058
40 - 50 MILES	44,339	14,303	11,763	58,647
50 - 60 MILES	41,954	9,115	55,709	352,482
SECTOR TOTALS	265,520	83,522	132,455	555,511

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Table 3 (Cont'd)

SPECIFIED ZONE POPULATION

BASED ON 1970 CENSUS

RING	SECTOR E	SECTOR F	SECTOR G	SECTOR H
0- ½ MILE	0	14	7	0
½- 1 MILE	280	62	140	210
1 - 2 MILES	630	630	910	1,470
2 - 3 MILES	263	298	683	1,068
3 - 4 MILES	980	1,085	1,190	595
4 - 5 MILES	875	385	1,190	385
5- 10 MILES	9.453	8,356	28,570	3,744
10 - 15 MILES	9,167	20,394	30,102	20,221
15 - 20 MILES	12,206	36,683	46,182	86,421
20 - 30 MILES	34,037	305,998	83,399	786,120
30 - 40 MILES	263,030	78,656	74,226	1,170,810
40 - 50 MILES	425,957	25,822	505,957	1,209,558
50 - 60 MILES	485,949	103,058	238,010	9,938
SECTOR TOTALS	1,242,827	581,441	1,010,566	3,290,540

Table 3 (Cont'd)

SPECIFIED ZONE POPULATION

BASED ON 1970 CENSUS

RING	SECTOR I	SECTOR J	SECTOR K	SECTOR L
0- ½ MILE	0	0	0	0
½- 1 MILE	14	0	0	0
1 - 2 MILES	680	910	10	300
2 - 3 MILES	53	263	840	350
3 - 4 MILES	105	1,610	1,505	560
4 - 5 MILES	245	2,415	945	560
5- 10 MILES	23,003	32,853	10,329	4,508
10 - 15 MILES	35,324	39,018	25,566	8,116
15 - 20 MILES	55,912	80,640	27,058	5,766
20 - 30 MILES	985,563	539,709	121,568	19,934
30 - 40 MILES	3,612,246	2,039,326	147,590	39,326
40 - 50 MILES	2,477,415	727,826	207,086	57,451
50 - 60 MILES	60,320	531,848	97,847	23,871
SECTOR TOTALS	7,255,880	3,996,418	640,344	160,742

2.4.P-11

Table 3 (Cont'd)

SPECIFIED ZONE POPULATION

BASED ON 1970 CENSUS

RING	SECTOR M	SECTOR N	SECTOR O	SECTOR P
0- ½ MILE	0	0	0	0
½- 1 MILE	0	0	0	0
1 - 2 MILES	420	0	0	30
2 - 3 MILES	504	98	595	280
3 - 4 MILES	0	0	1,155	305
4 - 5 MILES	630	875	840	1,260
5- 10 MILES	421	2,289	6,138	7,276
10 - 15 MILES	2,469	8,939	3,017	7,829
15 - 20 MILES	6,545	3,878	6,293	20,140
20 - 30 MILES	10,527	40,661	15,053	32,180
30 - 40 MILES	11,445	8,574	12,553	11,814
40 - 50 MILES	9,290	4,287	19,665	12,539
50 - 60 MILES	146	6,330	11,601	5,999
SECTOR TOTALS	42,397	75,931	76,910	99,652

IV. METHODS OF POPULATION PROJECTIONS

In May, 1970, Regional Economic Development Institute, Inc. (REDI) prepared a report entitled, “Population Estimates for 1960 and 2000 for the Specified Zones in a 60-Mile Area Around Indian Point New York”, for the Consolidated Edison Company of New York, Inc. The various methods of population projection used by REDI, Inc. are presented in the appendix of this report.

For this report, it is assumed that the population growth between 1970 and the year 2010 will be the same as the growth projected by REDI, Inc. for the forty-year period between 1960 and the year 2000. Hence, the ratio of the population of the year 2000 to the population of the year 1960 was calculated for each specified zone by using REDI, Inc. data. These ratios were multiplied by the 1970 population to project the population for the year 2010 in each specified zone.

The population for the years 1980, 1990 and 2000 was projected by assuming a linear growth rate between the years 1970 and 2010.

V. POPULATION PROJECTIONS FOR THE YEAR 1980

The population projections for the 13 rings is presented in Table 4 and the population projections for the 208 specified zones is presented in Table 5.

Table 4

Ring Population Projections for the Year 1980

Radius of the Ring in Miles	Population Projected for the Year 1980
Half	31
One	977
Two	10,973
Three	13,766
Four	18,591
Five	25,715
Ten	227,406
Fifteen	305,576
Twenty	576,576
Thirty	3,458,016
Forty	8,244,613
Fifty	6,109,740
Sixty	2,391,193

2.4.P-14

Table 5

SPECIFIED ZONE POPULATION

PROJECTED FOR THE YEAR 1980

RING	SECTOR A	SECTOR B	SECTOR C	SECTOR D
0- ½ MILE	0	0	0	0
½- 1 MILE	0	0	0	21
1 - 2 MILES	0	0	1,149	1,608
2 - 3 MILES	210	1,121	3,140	2,057
3 - 4 MILES	408	1,275	2,333	2,278
4 - 5 MILES	700	3,265	4,690	3,061
5- 10 MILES	10,407	3,205	6,569	22,169
10 - 15 MILES	9,420	3,781	9,855	8,283
15 - 20 MILES	34,219	6,965	13,765	15,766
20 - 30 MILES	77,309	46,669	20,419	25,993
30 - 40 MILES	108,058	25,776	31,217	129,318
40 - 50 MILES	54,497	19,933	15,110	76,898
50 - 60 MILES	57,349	12,419	37,400	428,480
SECTOR TOTALS	352,577	124,409	145,647	715,932

Table 5 (Cont'd)

SPECIFIED ZONE POPULATION

PROJECTED FOR THE YEAR 1980

RING	SECTOR E	SECTOR F	SECTOR G	SECTOR H
0- ½ MILE	0	20	10	0
½- 1 MILE	335	89	204	306
1 - 2 MILES	821	862	1,314	1,962
2 - 3 MILES	383	434	1,025	1,426
3 - 4 MILES	1,198	1,581	1,635	794
4 - 5 MILES	1,275	551	1,517	514
5- 10 MILES	14,124	12,220	34,059	5,020
10 - 15 MILES	13,167	27,158	40,157	23,322
15 - 20 MILES	17,427	53,074	56,279	103,985
20 - 30 MILES	50,326	357,186	98,194	840,603
30 - 40 MILES	296,179	94,020	99,455	1,214,673
40 - 50 MILES	504,338	38,066	390,456	1,249,092
50 - 60 MILES	543,492	153,458	173,378	11,301
SECTOR TOTALS	1,443,065	738,719	897,683	3,452,998

Table 5 Cont'd)

SPECIFIED ZONE POPULATION

PROJECTED FOR THE YEAR 1980

RING	SECTOR I	SECTOR J	SECTOR K	SECTOR L
0- ½ MILE	0	0	0	0
½- 1 MILE	20	0	0	0
1 - 2 MILES	907	1,282	14	422
2 - 3 MILES	74	370	1,017	493
3 - 4 MILES	124	1,994	2,117	790
4 - 5 MILES	258	2,989	1,327	747
5- 10 MILES	30,901	44,980	14,625	6,019
10 - 15 MILES	42,392	52,052	32,874	10,779
15 - 20 MILES	70,628	104,318	37,016	8,401
20 - 30 MILES	1,027,136	586,325	160,145	27,571
30 - 40 MILES	3,781,447	2,139,556	210,150	57,226
40 - 50 MILES	2,540,338	790,690	291,572	79,766
50 - 60 MILES	99,923	687,334	122,455	33,942
SECTOR TOTALS	7,594,148	4,411,890	873,312	226,156

Table 5 Cont'd)

SPECIFIED ZONE POPULATION

PROJECTED FOR THE YEAR 1980

RING	SECTOR M	SECTOR N	SECTOR O	SECTOR P
0- ½ MILE	0	0	0	0
½- 1 MILE	0	0	0	0
1 - 2 MILES	583	0	0	42
2 - 3 MILES	711	130	794	373
3 - 4 MILES	0	0	1,627	430
4 - 5 MILES	841	1,168	1,121	1,682
5- 10 MILES	562	3,056	8,914	10,567
10 - 15 MILES	2,084	13,580	4,813	11,849
15 - 20 MILES	10,074	5,675	9,503	29,473
20 – 30 MILES	15,490	55,759	23,063	45,820
30 – 40 MILES	15,793	11,315	15,647	14,775
40 – 50 MILES	11,592	5,652	24,759	16,973
50 – 60 MILES	173	7,626	14,189	8,265
SECTOR TOTALS	57,903	103,961	104,430	140,249

2.4.P-18

V. POPULATION PROJECTIONS FOR THE YEAR 1990

The population projections for the 13 rings is presented in Table 6 and the population projections for the 208 specified zones is presented in Table 7.

Table 6

Ring Population Projections for the Year 1990

Radius of the Ring in Miles	Population Projected for the Year 1980
Half	45
One	1,330
Two	14,298
Three	17,372
Four	24,499
Five	36,968
Ten	313,686
Fifteen	405,880
Twenty	763,773
Thirty	3,902,356
Forty	8,923,061
Fifty	6,520,698
Sixty	2,897,591

2.4.P-19

Table 7

SPECIFIED ZONE POPULATION

PROJECTED FOR THE YEAR 1990

RING	SECTOR A	SECTOR B	SECTOR C	SECTOR D
0- ½ MILE	0	0	0	0
½- 1 MILE	0	0	0	25
1 - 2 MILES	0	0	1,258	1,761
2 - 3 MILES	281	1,497	3,424	2,239
3 - 4 MILES	595	1,859	2,550	2,798
4 - 5 MILES	935	4,761	8,270	4,463
5- 10 MILES	14,536	4,959	9,872	33,031
10 - 15 MILES	13,450	5,151	14,467	12,342
15 - 20 MILES	45,119	11,157	20,800	22,972
20 – 30 MILES	100,403	74,321	31,187	34,244
30 – 40 MILES	148,301	37,651	42,944	194,324
40 – 50 MILES	66,983	27,779	19,410	100,830
50 – 60 MILES	78,394	16,922	25,109	520,865
SECTOR TOTALS	468,997	186,057	179,291	929,894

Table 7 (Cont'd)

SPECIFIED ZONE POPULATION

PROJECTED FOR THE YEAR 1990

RING	SECTOR E	SECTOR F	SECTOR G	SECTOR H
0- ½ MILE	0	29	14	0
½- 1 MILE	402	128	297	446
1 - 2 MILES	1,070	1,180	1,900	2,620
2 - 3 MILES	559	633	1,538	1,904
3 - 4 MILES	1,465	2,306	2,248	1,060
4 - 5 MILES	1,859	791	1,935	686
5- 10 MILES	21,105	17,872	40,603	6,733
10 - 15 MILES	18,914	36,167	53,573	26,900
15 - 20 MILES	24,882	76,789	68,584	125,120
20 – 30 MILES	74,412	416,938	115,614	898,863
30 – 40 MILES	333,506	112,386	133,260	1,260,179
40 – 50 MILES	597,142	56,115	301,323	1,289,919
50 – 60 MILES	607,850	228,507	126,297	12,851
SECTOR TOTALS	1,683,166	949,841	847,186	3,627,281

Table 7 (Cont'd)

SPECIFIED ZONE POPULATION

PROJECTED FOR THE YEAR 1990

RING	SECTOR I	SECTOR J	SECTOR K	SECTOR L
0- ½ MILE	0	0	0	0
½- 1 MILE	29	0	0	0
1 - 2 MILES	1,212	1,806	20	595
2 - 3 MILES	105	521	1,233	694
3 - 4 MILES	148	2,470	2,979	1,114
4 - 5 MILES	273	3,700	1,863	998
5- 10 MILES	41,511	61,585	20,709	8,037
10 - 15 MILES	50,875	59,441	42,273	14,316
15 - 20 MILES	89,218	134,949	50,640	12,240
20 – 30 MILES	1,070,462	636,968	210,965	38,135
30 – 40 MILES	3,958,574	2,244,713	299,229	83,275
40 – 50 MILES	2,604,850	858,985	410,528	110,748
50 – 60 MILES	152,859	888,277	153,251	48,263
SECTOR TOTALS	7,970,125	4,893,415	1,193,690	318,415

Table 7 (Cont'd)

SPECIFIED ZONE POPULATION

PROJECTED FOR THE YEAR 1990

RING	SECTOR M	SECTOR N	SECTOR O	SECTOR P
0- ½ MILE	0	0	0	0
½- 1 MILE	0	0	0	0
1 - 2 MILES	810	0	0	59
2 - 3 MILES	1,003	174	1,060	499
3 - 4 MILES	0	0	2,292	607
4 - 5 MILES	1,123	1,559	1,497	2,246
5- 10 MILES	750	4,080	12,948	15,348
10 - 15 MILES	1,760	20,632	7,678	17,934
15 - 20 MILES	15,507	8,304	14,352	43,131
20 - 30 MILES	22,794	76,464	35,336	65,242
30 - 40 MILES	21,794	14,933	19,503	18,480
40 - 50 MILES	14,466	7,452	31,174	22,974
50 - 60 MILES	206	9,188	17,256	11,388
SECTOR TOTALS	80,213	142,786	143,196	197,908

VII. POPULATION PROJECTIONS FOR THE YEAR 2000

The population projections for the 13 rings is presented in Table 8 and the population projections for the 208 specified zones is presented in Table 9.

Table 8

Ring Population Projections for the Year 2000

Radius of the Ring in Miles	Population Projected for the Year 2000
Half	65
One	1,826
Two	18,807
Three	22,228
Four	32,556
Five	53,915
Ten	434,823
Fifteen	542,975
Twenty	1,018,234
Thirty	4,459,201
Forty	9,748,933
Fifty	7,066,768
Sixty	3,597,413

Table 9

SPECIFIED ZONE POPULATION

PROJECTED FOR THE YEAR 2000

RING	SECTOR A	SECTOR B	SECTOR C	SECTOR D
0- ½ MILE	0	0	0	0
½- 1 MILE	0	0	0	31
1 – 2 MILES	0	0	1,377	1,927
2 – 3 MILES	376	1,999	3,734	2,437
3 – 4 MILES	867	2,711	2,787	3,438
4 – 5 MILES	1,249	6,941	14,582	6,507
5- 10 MILES	20,303	7,672	14,834	49,213
10 – 15 MILES	19,203	7,019	21,236	18,388
15 – 20 MILES	59,492	17,870	31,429	33,471
20 – 30 MILES	130,396	118,356	47,634	45,115
30 – 40 MILES	203,530	54,996	59,077	292,008
40 – 50 MILES	82,329	38,715	24,934	132,210
50 – 60 MILES	107,162	23,058	16,857	633,170
SECTOR TOTALS	624,907	279,337	238,481	1,217,915

Table 9 (Cont'd)

SPECIFIED ZONE POPULATION

PROJECTED FOR THE YEAR 2000

RING	SECTOR E	SECTOR F	SECTOR G	SECTOR H
0- ½ MILE	0	43	21	0
½- 1 MILE	481	184	434	650
1 – 2 MILES	1,396	1,616	2,745	3,499
2 – 3 MILES	815	923	2,309	2,542
3 – 4 MILES	1,792	3,362	3,091	1,416
4 – 5 MILES	2,711	1,134	2,468	916
5- 10 MILES	31,535	26,138	48,404	9,029
10 – 15 MILES	27,169	48,164	71,470	31,027
15 – 20 MILES	35,526	111,102	83,580	150,551
20 – 30 MILES	110,026	468,686	136,125	961,161
30 – 40 MILES	375,538	134,339	178,554	1,307,391
40 – 50 MILES	707,024	82,724	232,536	1,332,080
50 – 60 MILES	679,829	340,259	92,001	14,614
SECTOR TOTALS	1,973,842	1,236,674	853,738	3,814,876

Table 9 (Cont'd)

SPECIFIED ZONE POPULATION

PROJECTED FOR THE YEAR 2000

RING	SECTOR I	SECTOR J	SECTOR K	SECTOR L
0- ½ MILE	0	0	0	0
½- 1 MILE	43	0	0	0
1 – 2 MILES	1,618	2,545	29	839
2 – 3 MILES	148	735	1,494	979
3 – 4 MILES	176	3,060	4,191	1,573
4 – 5 MILES	288	4,581	2,617	1,333
5- 10 MILES	55,765	84,318	29,324	10,731
10 – 15 MILES	61,055	92,638	54,358	19,014
15 – 20 MILES	112,701	174,575	69,279	17,835
20 – 30 MILES	1,115,616	691,985	277,911	52,746
30 – 40 MILES	4,143,998	2,355,039	426,066	121,182
40 – 50 MILES	2,681,019	933,179	578,015	153,765
50 – 60 MILES	233,837	1,147,965	191,794	68,626
SECTOR TOTALS	8,406,264	5,490,620	1,635,078	448,623

2.4.P-27

Table 9 (Cont'd)

SPECIFIED ZONE POPULATION

PROJECTED FOR THE YEAR 2000

RING	SECTOR M	SECTOR N	SECTOR O	SECTOR P
0- ½ MILE	0	0	0	0
½- 1 MILE	0	0	0	0
1 - 2 MILES	1,126	0	0	84
2 - 3 MILES	1,415	230	1,416	666
3 - 4 MILES	0	0	3,230	857
4 - 5 MILES	1,499	2,082	1,999	2,999
5- 10 MILES	1,002	5,449	18,806	22,292
10 - 15 MILES	1,486	31,345	12,250	27,143
15 - 20 MILES	23,870	12,153	21,675	63,118
20 - 30 MILES	33,541	104,858	54,141	92,987
30 - 40 MILES	30,074	19,708	24,311	23,113
40 - 50 MILES	18,052	9,826	39,251	31,099
50 - 60 MILES	244	11,070	21,229	15,690
SECTOR TOTALS	112,309	196,721	198,308	279,958

VIII. POPULATION PROJECTIONS FOR THE YEAR 2010

The population projections for the 13 rings is presented in Table 10 and the population projections for the 208 specified zones is presented in Table 11.

Table 10

Ring Population Projections for the Year 2010

Radius of the Ring in Miles	Population Projected for the Year 1980
Half	88
One	2,365
Two	23,563
Three	27,333
Four	41,102
Five	73,713
Ten	566,518
Fifteen	688,705
Twenty	1,287,661
Thirty	5,013,456
Forty	10,552,150
Fifty	7,623,072
Sixty	4,335,347

Table 11

SPECIFIED ZONE POPULATION

PROJECTED FOR THE YEAR 2010

RING	SECTOR A	SECTOR B	SECTOR C	SECTOR D
0- ½ MILE	0	0	0	0
½- 1 MILE	0	0	0	36
1 – 2 MILES	0	0	1,481	2,072
2 – 3 MILES	474	2,520	4,003	2,608
3 – 4 MILES	1,173	3,666	2,993	4,053
4 – 5 MILES	1,575	9,386	22,955	8,799
5- 10 MILES	26,526	10,878	20,548	67,704
10 – 15 MILES	25,534	8,991	28,870	25,298
15 – 20 MILES	74,223	26,050	43,728	45,232
20 – 30 MILES	160,723	171,733	66,845	56,248
30 – 40 MILES	262,191	74,470	76,248	404,473
40 – 50 MILES	97,102	50,490	30,466	164,212
50 – 60 MILES	137,609	29,533	12,256	740,212
SECTOR TOTALS	787,130	387,717	310,393	1,520,947

Table 11 (Cont'd)

SPECIFIED ZONE POPULATION

PROJECTED FOR THE YEAR 2010

RING	SECTOR E	SECTOR F	SECTOR G	SECTOR H
0- ½ MILE	0	59	29	0
½- 1 MILE	557	246	587	880
1 – 2 MILES	1,726	2,079	3,686	4,410
2 – 3 MILES	1,102	1,249	3,196	3,204
3 – 4 MILES	2,106	4,546	3,987	1,785
4 – 5 MILES	3,666	1,513	2,999	1,155
5- 10 MILES	43,483	35,429	55,712	11,419
10 – 15 MILES	36,301	60,570	90,005	34,780
15 – 20 MILES	47,237	149,300	97,906	174,570
20 – 30 MILES	150,444	550,796	155,124	1,014,094
30 – 40 MILES	412,947	154,952	225,647	1,346,431
40 – 50 MILES	809,318	112,842	188,999	1,366,800
50 – 60 MILES	743,502	467,883	71,403	16,198
SECTOR TOTALS	2,252,389	1,541,464	899,280	3,975,726

Table 11 (Cont'd)

SPECIFIED ZONE POPULATION

PROJECTED FOR THE YEAR 2010

RING	SECTOR I	SECTOR J	SECTOR K	SECTOR L
0- ½ MILE	0	0	0	0
½- 1 MILE	59	0	0	31
1 – 2 MILES	2,040	3,349	39	1,104
2 – 3 MILES	196	967	1,742	1,288
3 – 4 MILES	202	3,632	5,508	2,072
4 – 5 MILES	301	5,434	3,435	1,680
5- 10 MILES	70,619	108,414	38,733	13,524
10 – 15 MILES	70,648	116,663	66,471	23,861
15 – 20 MILES	135,866	214,502	89,020	24,102
20 – 30 MILES	1,153,108	739,401	346,468	68,374
30 – 40 MILES	4,298,572	2,447,191	565,270	163,597
40 – 50 MILES	2,725,156	997,122	760,005	199,929
50 – 60 MILES	328,559	1,409,397	229,498	90,948
SECTOR TOTALS	8,785,326	6,046,072	2,106,189	590,479

Table 11 (Cont'd)

SPECIFIED ZONE POPULATION

PROJECTED FOR THE YEAR 2010

RING	SECTOR M	SECTOR N	SECTOR O	SECTOR P
0- ½ MILE	0	0	0	0
½- 1 MILE	0	0	0	0
1 – 2 MILES	1,466	0	0	111
2 – 3 MILES	1,865	294	1,785	840
3 – 4 MILES	0	0	4,250	1,129
4 – 5 MILES	1,890	2,625	2,520	3,780
5- 10 MILES	1,263	6,867	25,350	30,049
10 – 15 MILES	1,298	43,801	17,800	37,814
15 – 20 MILES	33,706	16,482	30,143	85,595
20 – 30 MILES	45,687	134,995	76,168	123,249
30 – 40 MILES	38,913	24,607	28,997	27,644
40 – 50 MILES	21,552	12,260	47,196	39,623
50 – 60 MILES	281	12,850	24,942	20,276
SECTOR TOTALS	147,921	254,780	259,151	370,110

APPENDIX A

Methods of population projections used by

R.E.D.I., Inc. for their May 1970 Report

"Population Estimates for 1960 and 2000

for Specified Zones

in a 60-Mile Area Around Indian Point"

2.4.P-34

METHOD OF POPULATION PROJECTIONS USED BY R.E.D.I., INC.

In summary, the estimating method consisted of these two major steps:

> Projections of total population in the year 2000, by counties, or parts of counties located within 60 milesfrom the site, were disaggregated by municipality or other minor civil divisions comprising the county, by two different equations, from which a compromise estimate was deter-mined, and Population of municipalities or other minor civil divisions as reported in the 1960 Census of Population and the 2000 projected compromise estimate derived as above were allocated to the specified zone on the basis of the area and land use of the portion of the one or more municipalities lying in part in the give zone.

PROJECTING MUNICIPALITY POPULATION

Since the projection of population requires long and costly re-search, it was decided to utilize larger area projections made by a responsible agency in the area. Among projections of this nature reviewed were those made by the Regional Plan Association of New York, National Planning Association of Washington, D.C. and the Tri-State Transportation Commission. In terms of area coverage and time span, those made by the Regional Plan Association (RPA) were found to be most suitable to the purposes of this study.

These RPA projections are the result of over fifteen years of study following the initial findings in the New York Metropolitan Region Study by Harvard University. The set used actually represents the estimates presently being utilized by the Association.

This RPA set required additional development of population estimates and projections for counties, or at least parts of counties, lying in the 60-mile area around the site but not covered by the RPA projections. The latest RPA set consists of New York City and 26 counties around New York City (5 of these counties were divided into two sub-areas to reflect major differences in density). Two of these are entirely outside of the 60-mile area around the site, but this area includes parts of 4 other counties not included in the RPA set. In summary, the 60-mile circle is made up of 36 areas, which may be classified into following sets: a) 23 full RPA-areas (representing 19 full counties); b) 9 areas that are portions of RPA-areas or counties: c) 4 areas that are parts of counties not belonging to the RPA Set.

The 2000 projections for those of these RPA counties which lie on the periphery of the 60-mile area around the site, were then scaled down to totals for only the municipalities or other minor civil divisions lying within or partially in this site-circumscribed area on the basis of their 1960 population as a per cent of the county total then.

For the four parts of counties in the site-circumscribed area but not in the RPA set, the 1960-2000 growth rates estimated for adjacent counties in the RPA set were used. The counties involved were,

Sullivan, New York
Columbia, New York
Pike, Pennsylvania
Hartford, Connecticut

It may be noted that since the parts of counties within the site-circumscribed area generally involved only a small strip of municipalities or other minor civial divisions, it is not unreasonable to assume that their growth patterns will approximate those of adjacent counties within the area.

The 1960 population and the 2000 projected estimates for the 36 counties and parts of counties used as control totals in the subsequent projections of municipality population, are reported in Table 3*. The ordering of areas in this table is that in the latest RPA population projections study and is centered on Manhattan. This order was merely adopted for convenience and has no bearing on the subsequent development of estimates by zones within the site-circumscribed area.

*Six counties or parts of counties consist of a single municipality. Therefore, municipal disaggregation of county projections were not needed in these instances. Note that the population totals for the entire area reported in Table 3 are greater than those reported in Tables 1 and 2 because of municipalities on the fringe falling only partially within the 60-mile radius.

Using these county control totals, population in the constituent 638 municipalities or other minor civil divisions were projected to 2000 by the techniques developed in our 1965 study for the Indian Point site. The following repeats the description of these techniques, drawn from that study, with revision to fit the present exercise.

A. Extrapolative Projection. Our first set of projections is based on the assumption that places which grew faster than their counties in the 1950's will continue to do so, and that those which lagged behind their counties' growth in the 1950's will continue to lag for the balance of this century. Specifically, each municipality's population was first projected linearly to 2000, simply extending the 1950-1960 growth rate for another 40 years. This extrapolation was done on an arithmetic basis (using the 1950-1960 increase rate in persons per annum)*. The extrapolated figures for all the muncipalities in each county or part of county were then totaled, and adjusted up or down pro rata to make the total conform to the RPA or other total already established for the county or part, as reported in Table 3.

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* Geometric extrapolations would result in obviously unrealistic projections.

B. Density-Based Projection. One quite obvious shortcoming of the extrapolative technique just described is that it takes no account of restraints upon growth arising from the filling up of developable space. We sought, therefore, to develop an alternative set of projections which would incorporate the hypothesis that percentage rates of growth of individual communities slacken off with higher population densities per square mile.

In our earlier study, investigation disclosed a marked relationship of the expected sort in the 1950-1960 growth and density rate for muncipalities. It appeared also that a straight-line regression relation between the logarithms of (1) the growth ratio and (2) the density of population at the beginning of the time interval provided a more appropriate formulation than a single linear relation.

Accordingly, the data on 1950 and 1960 population and land area for all muncipalities were processed so as to yield a statistically fitted regression formula for muncipalities in each county, relating rate of population growth to density of population. In fitting the equation, the data for individual municipalities were weighted according to population, giving larger places a proportionately greater influence on the formula.

These regression equations were generally associated with a high degree of correlation so as to leave little doubt as to the usefulness of this approach as a guide to relative rates of expected population growth.

The projections produced by this method for individual municipalities were then totaled for each county and adjusted to make the county totals conform to the RPA or other total already established for that county, or part of county, just as was done with the first or extrapolative set of projections.

C. Compromise Projection. The two alternative sets of projections just described rest on entirely different principles, each of which (the continuity of growth differentials, and the inverse relation of growth rate to density) has demonstrable validity but falls short of complete adequacy. Consequently, it seemed appropriate to combine the two types of projections into a third, incorporating both the continuity and the density effects.

To get this third set, labeled “com promise” projections, we took for each municipality the geometric mean between the adjusted extrapolative and the adjusted density-based projection. Then the “compromise” projections for the municipalities of each county were added up and adjusted to make the county total conform, as in the two previous cases, to the RPA or other total already established for the county or part of county.

The averaging procedure allows each of the two effects (growth-continuity and density) to exert an effect on the compromise projections. Where the extrapolative and the density-based projections were in close agreement, they reinforce each other in projecting differentiation of growth rates in different parts of a county; where the extrapolative and density-based projections give sharply differing answers, they tend to cancel one another out in terms of such differentiation, leading to compromise projections which show relatively little dissimilarity in growth rates among the part of a county. This seems appropriate – where the two approaches we have tried give very different results, we are well advised to take both of them less seriously and have less occasion for diverging very far from the simple assumption that all parts of any given county will grow at equal rates.

The geometric mean was chosen in preference to the arithmetic as a way of still further toning-down the most extreme variations of growth rates. It seems to us that the compromise projections are the “best” of the three so far as anyone can judge in advance. However, all three sets of projections are available for inspection should users of these reported results desire to make their own evaluations. and decisions.

REFERENCES

1970 Census of Population of New York, New Jersey, Connecticut and Pennsylvania

Vicinity Map Sheets of Poughkeepsie, Duchess County, New York and Danbury, Fairfield County, Connecticut

Census Tract Outline Maps for Patterson, New Jersey, New York City, New York, Stamford, Norwalk and Bridgeport, Connecticut

County Map Sheets for Westchester, Putnam, Orange, Rockland, Ulster County, New York and Fairfield County, Connecticut

Selected Sheets of New York, Northeast New Jersey Metropolitan Map Series

100-Percent Tract Tables for the Areas Within the Sixty-Mile Radius of Indian Point Nuclear Power Plant

1970 Census of Population and Hosing, Master Enumeration District – List

R.E.D.I., Incl., Population Estimates for 1960 and 2000 for Specified Zones in a 60-Mile Area Around Indian Point New York, May, 1970

ENVIRONMENTAL ANALYSTS, INC. 226 Seventh Street, Garden City, New York 11530 • (516) 741-3061

2.5 HYDROLOGY

The hydrological features of the Indian Point site are relevant to the analysis of radioactive liquid and gaseous discharges from the plant. During normal plant operation, liquid wastes are discharged to the Hudson River through the Indian Point 3 circulating water discharge tunnel into the common discharge canal for Units 1, 2, and 3. Discharges are limited in accordance with the State Pollutant Discharge Elimination System Permit. The sources of ground water will not be susceptible to contamination from accidental ground seepage or leakage from the plant because of the permeability of the bedrock and the higher elevation of the plant relative to the river. Gaseous releases from the plant following a hypothetical accident have been studied for possible deposition of contaminants into surrounding surface water reservoirs. Therefore, the hydrological features are categorized by the Hudson River, ground water and wells, and surface water reservoirs. Two consultants have studied the hydrology of the Indian Point site. In 1955, prior to construction of Unit No. 1, Mr. Karl R. Kennison reported the flow characteristics of the river at the site. In 1965, the firm of Metcalf and Eddy reviewed Mr. Kennison's report and further reported the ground water hydrology and surface water reservoirs. The report by Metcalf and Eddy is included in this section, appended by the Kennison report (see pages S-1 to S-35).

Flow in the Hudson River is controlled more by the tides than by the runoff from the tributary watershed. Opposite the plant, the width of the river is 4500 to 5000 feet with a depth of 55 to 75 feet less than 1000 feet offshore. Total flow past the plant during the peak tidal flow is about 80,000,000 gallons per minute about 80 percent of the time, and it has been estimated that about 500 feet off the shore line, flow is at least 9,000,000 gallons per minute in a section 500-600 feet wide. This large flow assures adequate dilution and complete mixing of the discharge from the plant. The plant was designed and operates such that discharges into the river do not prevent using the river water for drinking purposes.

The net mean downstream flow due to runoff is as follows:

11,700,000 gpm may be expected to be exceeded 20 percent of the time;

4,710,000 gpm may be expected to be exceeded 60 percent of the time;

1,800,000 gpm may be expected to the exceeded 98 percent of the time.

The history of river flow for a 17-year period, presented in tables in both the Kennison and the Metcalf and Eddy reports, is for the net river runoff flow.

Within a five-mile radius of the plant only one municipal water supply utilizes ground water. Other wells are for industrial and commercial usage. The rock formations in the area, and the 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. This subjects is discussed further in Section 2.7.

Only two reservoirs within a five-mile radius of the site are used for municipal water supplies. The Camp Field Reservoir is the raw-water receiving basin for the City of Peekskill with the Catskill Aqueduct and Montrose Water District as alternate supplies. 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 alternate source of supply. A third reservoir within five miles of the plant, Queensboro Lake, supplies water to a state park area only. The location of these reservoirs and those within a fifteen-mile radius are shown on Figure 2.5-1.

The City of New York's Chelsea Pumping Station is located about one mile north of Chelsea, New York, on the east bank of the Hudson River. Water can be pumped from intakes in the river at the rate of 100 million gallons per day into the city reservoir system, as required to supplement the primary supply from watersheds. The pumping station is 22 miles upriver from Indian Point measured along the centerline of the river as shown on Figure 2.5-2.

Discharge of any contaminant to a tidal estuary will result in its distribution throughout the estuary. Factors affecting this distribution include: tidal amplitude and current, river geometry, salinity distribution, and fresh water discharge. Quirk, Lawler and Matusky Engineers, Environmental Science and Engineering Consultants, of New York City, made extensive studies of the influence of these factors and assisted in the study of contaminant transport in the river. A report of this study is included in this section (see pagesI-1 to D-3). This study served to determine the effect of radioactive discharges on overall river contamination, and specifically conditions at Chelsea pumping station, as discussed in detail in Section 14.3.2. During normal operation, discharges do not result in river concentrations that exceed Maximum Permissible Concentrations (MPC) at the Indian Point discharge canal.

Flooding at the site has been nonexistent. The highest recorded water elevation at the site was 7.4 feet above mean sea level during an exceptionally severe hurricane in November, 1950.

Since the river water elevation would have to reach 15'-3" above mean sea level before it would seep into the lowest floor elevation of any of the Indian Point buildings, the potential for any flooding damage at the site appears to be extremely remote.

However, in order to determine the maximum probable elevation that the Hudson River could reach, the firm of Quirk, Lawler and Matusky was commisioned to make an in-depth study of the river under various flooding conditions. A report of this study, dated April, 1970, is included in this section (see pages I-1 to D-3).

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

a) Flooding resulting from runoff generated by a Probable Maximum Precipitation over the entire Hudson River drainage basin upstream of the site.

b) Flooding caused by the concurrent of an upstream dam failure concurrent with heavy runoff generated by a Standard Project Flood.

c) 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 is a result of the simultaneous occurrence of a Standard Project Flood, a failure of the Askokan 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 feet above mean sea level. Local wave action due to wind effects has been determined to add 1 foot to the river elevation, producing a maximum water elevation of 15 feet above mean sea level at the Indian Point site.

Since this maximum water elevation is 3 inches lower than the critical elevation of 15'-3" 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 3.

Additional Hudson River Hydrodynamic Studies

During the mid- and late-1970's, many hydrological studies of the Hudson River were conducted by electric utilities both in the vicinity of Indian Point and over the entire tidal portion of the estuary from the George Washington Bridge to the Albany dam. Consolidated Edison and the Power Authority have transmitted the results of these studies to the Nuclear Regulatory Commission over the years, some as part of the Environmental Technical Specifications report requirements, and others as part of applications for license amendments. Because the results of both the data acquisition and the hydro-dynamic modeling studies generally support the earlier studies already described, these results will be reviewed very briefly:

The La Salle Hydraulic Model Study of Hudson River Flows Around Cooling Water Intakes, 1976

The purpose of the La Salle study was to estimate the range of the proportion of cooling water which recirculated from the discharge back into the intakes of Indian Point 2 and 3. Data were acquired by physical modelling and by use of neutral buoyancy drogues released in front of the intakes and the discharge. In addition, observation of patterns of turbulence and flow direction and strength were also made during all tidal stages and aided in interpretation of the quantitative data.

The study concluded that, over the whole tidal cycle, an average of 7.7 percent of the discharge water was recirculated. When the flood tide is running, the effluent creates a large eddy in front of the intakes, such that water drawn in actually comes from the upstream direction. The effluent itself, during a flood, goes around the outside of the eddy in front of the intakes, continuing upstream in a relatively wide, turbulent, diffused zone reaching up to 700 feet out from the east shore before arriving at Peekskill Bay. During ebb tide, all water comes to the intakes from a narrow 200 to 250 foot wide zone along the east shore. The effluent goes downstream in a zone 400 to 500 feet out from the east shore. During slack water, water is drawn into the intakes from the area just in front of the plants or from upstream because the effluent cuts off the flow from downstream. Various mixing processes during the different tidal stages result in fairly complete mixing of water in front of the intakes, minimizing saline and thermal stratification.

Influence of Indian Point Unit 2 and Other Stream Electric Generating Plants on the Hudson River Estuary, 1977

This report, edited by James T. McFadden, includes contributions by Texas Instruments and Lawler, Matusky and Shelly Engineers on the Hydrodynamics of the Hudson Estuary. Over a 57-year period, the long-term annual average freshwater flow at Green Island was estimated to be 13,268 cubic feet/second. However, during the early-to-mid 1970's, the annual average ranged from a low of 14,547 in 1971 to a high of 19,077 in 1973. Monthly average flows ranged from a low of 5,591 cubic feet per second in August 1973 to a high of 40,520 in May 1972. Tidal flows determine the overall flow regime of the Indian Point reach of the estuary, reaching 100,000 to 200,000 cubic feet per second. Only extreme freshwater inflows such as those which occurred in storms on March 28, 1913 and March 19, 1936 have suppressed tidal flows below the Albany dam. This suppression extended 48 Km below the dam.

During times of low freshwater flow, the salt front ascends the estuary to as far north as Newburgh. In drought, it may even reach Poughkeepsie. The presence of saline and fresh waters in the same river reach induces density- dependent turbulences and mixing. During these periods, tidal and vertical turbulences act to produce mixing and water mass transport motions. The ratio of tidal flow (Qt) to freshwater flow update (Qf) yields the vertical stratification factor, which at Indian Point ranges from a low of 3.3 in spring to a high of 23 in summer.

Conclusion

In the event of a radiological discharge into the Hudson River estuary, the turbulence caused by the discharge velocity itself plus the natural mixing processes described by the La Salle and the McFadden reports would determine the rapidity of mixing, of dilution, and of transport and diffusion. Mixing processes are at work at all times. The two reports, the major contributors in near-field and far-field hydrodynamics for Indian Point, support the findings of the earlier Indian Point hydrodynamic studies.

HYDROLOGY OF INDIAN POINT SITE

AND SURROUNDING AREA

METCALF & EDDY ENGINEERS

OCTOBER, 1965

REPORT PREPARED BY GEORGE P. FULTON
UNDER DIRECTION OF HARRY L. KINSEL, P. E.

S-1

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-1-

INTRODUCTION

The hydrological features of the Indian Point site have been studied in three categories; the Hudson River, ground water and surface water reservoirs. Flow data and the flood history of the Hudson River in the vicinity of the Indian Point plant are discussed. Ground water sources within the area are generally used for industrial or commercial purposes with some limited residential usage on the west side of the river. The surface water reservoirs in the surrounding area that are used for water supplies and sources of alternate water supplies are also described.

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HUDSON RIVER

General

The Consolidated Edison Indian Point plant is situated on the east bank of the Hudson River below Peekskill, just above Verplancks Point. In the general area of the plant, water from the Hudson River is used only for industrial cooling purposes. The nearest community utilizing the Hudson River for a public water supply at the present time is Poughkeepsie, some 30 miles upstream from the plant site.

Flow

Flow data for the Hudson River were abstracted from a previous report of Mr. K. Kennison, submitted to Consolidated Edison on November 18, 1958 (included as an appendix to the section on hydrology). Flood data were obtained from the Survey Division of the Corps of Engineers in New York City.

In the vicinity of Indian Point, the width of the Hudson River ranges from 4,500 to 5,000 feet with maximum depths of from 55 to 75 feet. Cross sectional areas of the river from a point three quarters of a mile upstream from the plant site to a mile downstream are in the order of from 165,000 to 170,000 square feet.

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-3-

Flow duration records of the Hudson River for a 17-year period preceding 1930 show the following:

Rate of Flow c.f.s.	Percent of Time Exceeded

26,000	20%
15,250	40%
10,500	60%
7,000	80%
4,000	98%

It is evident that even the highest rates of flow expected will influence depth of flow in the river to only a small degree in the vicinity of the plant. This is due to the relatively high available flow section and the width of the river. River depth is affected more by the tidal influence than it can be by any anticipated flood flows.

The Hudson River is tidal as far upstream as Troy, some 100 miles from Indian Point. The elevation of the water surface in the vicinity of the plant is so responsive to the tidal cycle that average rate of flow has little effect on depth of flow or velocity of flow.

Flood History

Tide elevations vary both daily and seasonally and, in addition, can be affected by atmospheric conditions such as can exist during extreme storms or hurricanes. The atmospheric conditions can cause a surge which, added to the normal tide, establishes water elevation.

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The highest water elevation at the U.S.G.S. station at Verplancks Point, one-half mile below Indian Point, was 7.4 feet above MSL (mean sea level) recoreded in the year 1950. A higher surge occurred in 1960, but the normal tide stage was such that actual water elevation was somewhat less than the 1950 record.

in an earlier period, before 1935, the highest recorded elevation was 4.75 ft. above MSL at Verplancks Point on August 24, 1933.

Mean water elevations at Verplancks Point are just below 1.0 (MSL). The mean range of water depth stages is about 3.0 ft.. With high runoff in the Hudson River Basin, the mean range at times averages a half a foot higher during the spring period.

The highest river elevation, recorded in 1950, was about 6.5 feet higher than average river levels, or some 5.0 feet higher than average high river stages. Considering past flood history and the fact that flood stages are primarily the effect of tidal influence, flooding of the Indian Point plant site appears to be a highly unlikely possibility.

Contamination Potential

The hazards of contamination of water supplies by discharge of water borne wastes from the Consolidated Edison Indian point plant are almost minimal. In the reach of the Hudson River that could be affected, river water is used only for industrial cooling.

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It should be mentioned that the City of New York is now in the process of constructing a river water pumping station at Chelsea in Putnam County below Poughkeepsie. The intent is to pump Hudson River water into the City system.

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WELLS AND GROUND WATER

General

Within a five-mile radius of the plant the only public water supply using ground water is the Stony Point system of Utilities and Industries located in Rockland County across the river from Indian Point. Reports on ground water resources within this five-mile radius indicate the existence of numerous other wells. These wells are for industrial and commercial usage and for individual water supplies for private residences. Residential usage, however, is almost entirely confined to the area on the west side of the Hudson River.

Ground Water Geology

Water bearing strata in the area within a five-mile radius of Indian Point can be divided into unconsolidated surface deposits and consolidated bedrock. Unconsolidated deposits cover most of the bedrock in this area and range in thickness from a few feet in the hills to several hundred feet in the larger\ valleys. Unconsolidated deposits range from clays, which produce only meager quantities of water, to coarse sand and gravel capable of yielding several hundred gallons per minute to a well.

The bedrock underlies the unconsolidated deposits and, where these are absent, crops out at the surface. Ground water in bedrock occurs principally in fractures and solution channels.

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Thus, the water bearing characteristics are generally similar, although the rocks differ widely in mineral composition and water yield.

Bedrock in Westchester County is, for the most part, metamorphic in character and includes schist and gneiss, with smaller amounts of limestone, quartzite and slate. Small injections of granite can also be found. Only minimal yields of ground water can be obtained from bedrock formations in Westchester County.

Consolidated rocks are the chief source of water in Rockland County. Principal rock units include the following:

a) Newark Group; - sandstone, shale and conglomerate.
b) Palisade Diabase – diabase with some baslat.
c) Cambrian and Ordovician Rocks – quartzite, limestone and dolomite.
d) Precambrian Rocks – granite, gneiss, with some schist and diorite.

The Newark group provides the greatest source of ground water supply in Rockland County. The other units of bedrock yield only minimal quantities, as in Westchester County.

A small area of Orange County lies within the 5-mile radius being considered. Wells in this area have been drilled in bed-rock formations similar to those in Westchester County where the water yield is small.

S-9 -8-

Well Supplies

As mentioned before, the only public water supply served by wells in the 5-mile radius of Indian Point is the Stony Point System. This system serves the Villages of Haverstraw and West Haverstraw as well as portions of the Towns of Haverstraw and Stonly Point. The Stony Point supply wells are located in strati-fied drift, an unconsolidated formation. These wells are rela-tively shallow, the greatest depth about 35 ft. Total yield of the wells to the system averages about 550 gpm.

Other wells in Rockland County, in the area being considered, include some wells for commerical and industrial use and many private wells serving individual residences. These wells are located in bedrock for the msot part and range from 100 to 300 ft. in depth. Consumption of water from wells serving private homes will vary from 100 to 1,000 gpd (gallons per day), depending on the number of persons using the supply and the facilities using water.

There are only a few wells still in use in Westchester County within the 5-mile radius. Almost all the wells within 2 to 3 miles of Indian Point have been abandoned and connections have been made to public water systems for suppy. At the fringes of the area a few private wells are used for individual residences. These wells are mostly in unconsolidated deposits with depths less than 50 ft. Some wells exist in bedrock with depths varying up to several hundred feet.

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A small portion of the community of Fort Montgomery in Orange County lies within 5 miles of the plant. Homes in this community are served entirely by individual private wells in bedrock. Depth of the wells vary up to several hundreds of feet.

Contamination Potential

The bedrock formation is such that it is highly unlikely that wastes percolating into the ground from the Indian Point site will reach the water bearing formations used for water supply on the west side of the river in Rockland and Orange Counties. Most of the wells in Westchester County are shallow, in unconsolidated formations with ground surface elevations considerably higher than at the plant site. This situation would preclude the possibility of contamination of the supply through ground water flow. Bedrock wells in Westchester County are similarly at higher elevations and, for the most part, are drilled in different rock formations than exists at the plant site.

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SURFACE WATER RESERVOIRS

General

The major sources of water supply in the Indian Point area are lakes and surface water reservoirs. The reservoirs within a 15-mile radius of the plant site are tabulated in Tables 1-7 along with the users, capacities and distances from Indian Point. A detailed analysis of the reservoirs within 5 miles of the plant describes alternate sources of supply to those communities served by the reservoirs.

City of Peekskill-Camp Field Reservoir

The 54-million gallon Camp Field Reservoir of the City of Peekskill system, located 2.9 miles from Indian Point, is a raw-water receiving basin for the water treatment plant. Water is pumped into this basin from Peekskill Hollow Brook. For the most part, the water supply is the continuous flow of this brook. At times of low flow the supply can be supplemented by releasing water into the stream from holding reservoirs in Wicopee (Putnam County) some 11.7 miles from Indian Point or from the Catskill Aqueduct of the City of New York, located a short distance upstream from the pump intake.

The City of Peekskill system is divided into two service pressure areas. Water for the low-pressure area flows by gravity from Camp Field Reservoir through a bank of slow-sand filters

into the system. No additional storage is provided for this section of the system. Water for the high-service area flows from the reservoir through two diatomaceous earth filters by gravity and then is pumped to a pair of elevated storage tanks with a total capacity of 800,00 gallons. The high-service system serves approximately 25 percent of the peekskill area. The remaining area, including Standard Brands and most of the other industrial consumers, is served by the low-pressure system.

Total water consumption in Peekskill averages about 5 mgd. The largest single user is Standard Brands, at an average rate of 1.5 mgd. All water is supplied from Peekskill Hollow Brook. Two connections to other systems are available for emergency conditions. One is the above-mentioned Catskill Aqueduct connection which discharges into Peekskill Hollow Brook. This flow must be processed through the two treatment facilities for use. The other emergency connection is to the Montrose Water District system which can supply between 1.0 and 1.25 mgd from the Catskill Aqueduct to the low-service section of the Peekskill system.

Since no piping is installed to bypass Camp Field Reservoir, contamination of this basin would deprive Peekskill of its normal source of supply. Installation of a bypass would involve some 800 lin. ft. of 24-in. pipe between the inlet force mains and the outlet lines to the two filter facilities. With such a

S-13 -12-

bypass, it would be possible to take water directly to the filters from Peekskill Hollow Brook after the passage of contaminated water in the event of prolonged contamination of Camp Field Reservoir. It might be necessary to accelerate flushing out of the brook and the impoundment at the pumping station in such a situation by releasing water from either the Catskill Aqueduct or the Wicopee reservoirs.

Peekskill most likely could not depend on the Montrose connection alone. This can supply less than one-hale the normal demands of the low-service system even with the assumption that Standard Brands would not operate during the emergency. The high-service system has only 800,00-gallon storage, which would last less than 24 hours after shutting down the Peekskill Hollow Brook supply.

As presently arranged, the City of Peekskill would be practically deprived of a water supply with elimination of Peekskill Hollow Brook as a source. A study will soon be made under the auspices of the Westchester County Water Agency and the State of New York to determine the feasibility of connecting the Peekskill system to a proposed transmission main crossing northern Westchester County from the Delaware Aqueduct of the City of New York. This proposal could furnish an independent source of water in sufficient supply to serve all the needs of the City of Peekskill in the vent of an emergency.

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<u>Palisades Interstate Park Commision – Queensboro Lake</u>

Queensboro Lake, some 5 miles from Indian Point,k serves as the year-round water supply for Bear Mountain Inn. The inn facilities include the offices of the Palisades Interstate Park commission as well as a hotel and restaurant. Three other lakes feed into Queesboro Lake through stream flow or by pip connection. Only Queensboro Lake is connected directly to the water system and no bypass is available to route water around the lake from a more distant location.

In case of contamination of Queensboro Lake, Bear Mountain Inn would be deprived of its water supply. A neighboring community, Fort Montgomery, is served entirely by individual private wells. This would seem to indicate that installation of an emergency well supply for Bear Mountain Inn would be feasible.

Stony Point Water System – Utilities and Industries

The Stony Point supply of Utilities and Industries, and investor-owned water company, serves the towns of Stony Point and Haverstraw as well as the villages of Haverstraw and West Haverstraw. Total average consumption is about 1.l8 mgd with 1.0 mgd from a surface supply and 0.8 mgd from wells.

The impounding reservoir of the surface supply of 4.5 million gallon capacity os located some 3.5 miles from Indian Point 3 Point. With contamination of this supply, the system would be left with only the wells which furnish about 45 percent of total consumptions.

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Negotiations are now under way for purchase of the Stony Point supply by the Spring Valley Water Company, an investor-owned utility serving most of the remaining areas of Rockland County. This company derives water from a well system of 13 to 15 mgd capacity and up to 7 mgd from De Forest Lake outflow some 10.8 miles from Indian Point. Plans have been completed for construction this fall of a connection between the Spring Valley Water Company system and the Stony Point system. This connection will furnish well water form the Spring Valley supply to the Stony Point network.

As far as can be ascertained from public records, the above three systems comprise the only surface water usage within a 5-mile radius of the Indian Point power plant except for industrial cooling water usage of the Hudson River. All other supplies are reported as originating in wells or from surface storage outside the 5-mile limit.

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TABLE 1

WATER STORAGE RESERVOIRS

WITHIN 15 MILE RADIUS OF INDIAN PONT

WESTCHESTER COUNTY

Code	Reservoir	User	Capacity Million Gallons	Distance Miles	Surface Acres
W-8	Indian Brook	Ossining WB.	101	6.5	17
W-18	Pocantico Lake	New Rochelle Wat. Co.	200	11.9	63
W-14	Ferguson Lake	Pocantico Hills Est.	40 *	13.5	28
W-13	Tarrytown Res.	Tarrytown	313	14.0	85
W-13	Open Res. – 2	Tarrytown	1.75 & 1.10	14.0	1
W-1	Croton Res.	New York City (See List)	65,300 (Inside 15 mi.)		4059
W-10	Whippoorwill La.	New Castle Wat. Co.	25 *	13.3	8
W-11	Byram Lake	Mt. Kisco	950	15.0	133
W-11	Open Res.	Mt. Kisco	10 *	14.0	2
W-5	Lake Shenorock	Amawalk-Shenorock WD.	90 *	11.1	16
W-6	Open Res.	Lincoln Hall School	25 *	11.9	6
W-1A	Amawalk	NYC (See List)	10,000 (Included in W-1)	11.6	588
W-4	Camp Field Res.	Peekskill	54	2.9	11

* Estimated

TABLE 2

WATER STORAGE RESERVOIRS

WITHIN 15 MILE RADIUS OF INDIAN PONT

PUTNAM COUNTY

Code	Reservoir	User	Capacity Million Gallons	Distance Miles	Surface Acres
P-20	Lake Mahopac	See List	5,000 *	12.7	577
P-10	Oscawanna Lake	See List	3,500 *	9.5	362
P-21	Pelton Pond	N.Y.S. Fahnestock Park	125 *	14.0	11
P-6	Cold Spring	Cold Spring	150 *	13.0	25
B-3	Cargill Res.	Beacon	160	15.0	22
B-2	Mt. Beacon Res.	Beacon	180	14.5	17
B-1	Melzingah Res.	Beacon	60	13.3	8
W-4	Wicopee	Peekskill	1,200	11.7	166
P-5	Lake Secor	Carmel WD #5	350 *	10.8	50

*Estimated

TABLE 3

WATER STORAGE RESERVOIRS

WITHIN 15 MILE RADIUS OF INDIAN PONT

ORANGE COUNTY

Code	Reservoir	User	Capacity Million Gallons	Distance Miles	Surface Acres
0-11	Lusk Res.	U.S. M.A.	50 *	7.5	16
0-4	Intake Res.	Highland Falls	2.5	6.5	
	Bog Meadow	" "	80	8.3	43
	Little Bog	" "	4.5	7.5	2
	Jims Pond	" "	40	8.4	16
0-12	Turkey Hill La.	Palisades Int. Park	150	5.9	58
	Nawahunta La.	" "	22	6.7	16
0-20	Silvermine La.	Palisades Int. Park	456	6.0	84
	Queensboro La.	" "	56	5.0	37
0-16	Lake Stahahe	Palisades Int. Park	230	11.1	90
0-16	Summit Lake	Palisades Int. Park	110	8.3	34
	Barnes La.	Pal. Int. Pk. & U.S.M.A.	24	8.0	18
	Te'ata La.	Pal. Int. Pk.	77	7.7	32
	Upper Twin La.	" " "	105	7.7	24
	Lower Twin La.	" " "	88	7.6	26
	Massawiepa La.	" " "	104	7.7	29
0-17	Lake Tiorati	Pal. Int. Pk., Tiorati	1,500	6.7	296
0-10	Crowmwell Lake	Woodbury	80	11.2	55
0-2	Walton Lake	Chester	300	14.6	129

0-5	Lake Mombasha	Monroe	1,750	13.0	324
0-1	Echo Lake	Arden Farms	40 *	9.5	30
0-7	Op Res.	Sterling Forest	60 *	13.7	42

ORANGE COUNTY (CONT'D.) TABLE 3 (CONT'D)

Code	Reservoir	User	Capacity Million Gallons	Distance Miles	Surface Acres
0-8&9	Tuxedo Lake	Tuxedo & Tuxedo Pk.	2,500	14.5	294
0-3	Alex Meadow	Cornwall	23	9.2	9
	Arthur's Pond	Cornwall	115	9.2	20

* Estimated

TABLE 4

WATER STORAGE RESERVOIRS

WITHIN 15 MILE RADIUS OF INDIAN PONT

ROCKLAND COUNTY

Code	Reservoir	User	Capacity Million Gallons	Distance Miles	Surface Acres
R-14	Lake Sebago	Sebago Lake, Pal. Int. Pk	1,100	10.8	300
R-18	Lake Welch	Welch Lake	1,000	7.2	209
R-13	Breakneck Pond	Breakneck Lake, Pal. Int. Pk.	100	9.2	63
R-3	Sec. & Third Res.	Letchworth Vill.	100	8.5	40
R-1	Open Res.	Utilities & Ind.	4.5	.3.5	5
R-7	Hillburn Res.	Hillburn	1.0	14.7	4
R-6	Deforest Lake	Hackensack Wat. Co. Spring Val. Wat. Co.	5,500	10.8	960

TABLE 5

MULTIPLE USERS OF WATER SUPPLY SYSTEMS
WITHIN 15 MILE RADIUS OF INDIAN POINT
WESTCHESTER COUNTY

New Croton Aqueduct (New York City)
- Ossining Water Board
- Sing Sing Prison
- Village of North Tarrytown
- New Rochelle Water Company
  - Village of Bronxville
  - Town of Eastchester
  - Village of North Pelham
  - Village of Pelham
  - Village of Pelham Manor
  - Village of Tuckahoe
- Village of Irvington
- Village of Briarcliff Manor
- New Castle Water District #1
- Village of Tarrytown

Old Croton Aqueduct (New York City)
- Ossining Water Board
  - Village of Ossining
  - Town of Ossining
  - Sing Sing Prison

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Kensico Reservoir (New York City)

City of White Plains

North Castle District #1

Westchester Joint Water Works No. 1

Village of Mamaroneck

Town of Harrison

Town of Mamaroneck

City of Rye

City of New Rochelle

Village of Larchmont

Village of Scarsdale

Village of Pelham Manor

Harrison District #1

Catskill Aqueduct (New York City)

Grasslands (Westchester Co.)

Hawthorne Improvement District

Hawthorne

Town of Mt. Pleasant

Valhalla W D

Valhalla

Town of Mt. Pleasant

City of Yonkers

Village of Scarsdale

New Rochelle Wat. Co (same as Pocantico Lake)

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Amawalk Reservoir (New York City)

Yorktown W S D D

Amawalk Heights W D

Town of Somers

Town of Yorktown (13 Water Districts)

Peekskill System (City of Peekskill)

City of Peekskill

Village of Buchanan

Town of Cortlandt

Indian Brook Reservoir (Ossining Water Board)

Village of Ossining

Town of Ossining

Sing Sing Prison

Whippoorwill Lake (New Castle Water Co.)

Town of New Castle (Part)

Town of North Castle (Part)

Pocantico Lake (New Rochelle Water Co.)

Village of Ardsley

Village of Dobbs Ferry

Town of Greenburgh

Village of Hastings

Village of Scarsdale

Village of Eastchester

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Tarrytown Reservoir

Village of Tarrytown

Glenville W D

Town of Greenburgh

Eastview

Town of Mount Pleasant

Village of North Tarrytown

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TABLE 6

MULTIPLE USERS OF WATER SUPPLY SYSTEMS

WITHIN 15 MILE RADIUS OF INDIAN POINT

PUTNAM COUNTY

Lake Oscawanna

Hiawatha Improvement Co.

Hilltop W D

Wildwood Knolls W D

Oscawanna Lake (Private Homes)

Lake Mahopac

Lake Gardens

Lake Mahopac Woods

Mahopac Hills

Mahopac Old Village

Lake Mahopac (Private Homes)

Lake Mahopac Ridge

Lake View Park

Mahopac School

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TABLE 7

MULTIPLE USERS OF WATER SUPPLY SYSTEMS
WITHIN 15 MILE RADIUS OF INDIAN POINT
ROCKLAND COUNTY

De Forest Lake

- Hackensack Water Co.
- Spring Valley Water Co.
  - Town of Clarkstown (Part)
  - Town or Ramapo
  - Town of Orangetown
- Nyack
  - Village of Nyack
  - Village of South Nyack
  - Upper Nyack
  - Town of Clarkstown (Part)

Stony Point Supply (Utilities and Industries)

- Town of Stony Point
- Town of Haverstraw
- Village of West Haverstraw

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TRANSPORT OF CONTAMINANTS

IN THE HUDSON RIVER

ABOVE INDIAN POINT STATION

QUIRK, LAWLER AND MATUSKY ENGINEERS
Environmental Science and Engineering Consultants
New York, New York

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BASIS FOR ANALYSIS

Transport of any substance in a tidal estuary is governed by the Law of Conservation of Mass. Figure 1 illustrates the application of this law in an estuary. After discharge to the estuary, waste particles are carried down-stream, in the movement of upland runoff toward the ocean. This phenomenon is known as convection. The rate of convective mass transport across any river section is equal to the product of fresh water runoff, Q, and contaminant concentration, c.

Besides convection, particles are transported in an estuary by longitudinal mixing. Longitudinal mixing, or dispersion, is a complex function of reversing tidal currents and salinity-induced circulation. Dispersive transport occurs only in the presence of a concentration gradient of the material being transported. The rate of dispersive transport is equal to the product of the dispersion coefficient, E, and the negative of the longitudinal concentration gradient, dc/dx. The dispersion coefficient, E, is a measure of the estuary's ability to transport material in the presence of a concentration gradient, and is a quantitative function of tidal current and salinity-induced circulation.

The concentration profile in Figure 1 indicates how convection and dispersion distribute estuarine contaminants. Since only contaminants that decay or, at best, are conserved, are being considered, the maximum containment concentration must occur at the point of introduction of the contaminant to the estuary. In the case of saline contamination, the salt is introduced at the mouth of the estuary so that the maximum salinity occurs here; in the case of discharge of radioactive contaminants at Indian Point, the maximum concentration of radioactivity will exist in this vicinity, as shown on Figure 1.

The concentration in the region downstream of the point of discharge, (x = 0), decays less rapidly than does its counterpart in the upstream region. This is so because in the downstream region, dispersion, in moving material in the direction of decreasing concentration, aids convection. More material is transported downstream than upstream, so, at the same absolute distance from the point of discharge, the upstream concentration is lower than the downstream concentration.

A mass balance over the incremental volume, $A\Delta x$, in Figures 1 is written:

Inflow – Outflow + Production = Accumulation . . . . (1)

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S-38 Figure 1

Algebraic summation of the individual contributions, shown in Figure 1 to Equation 1 gives:

$$\left[Qc - EA\frac{dc}{dx}\right]_x - \left[Qc - EA\frac{dc}{dx}\right]_{x+\Delta x} - KCA\Delta x = \frac{d}{dt}\left[CA\Delta x\right] \qquad \text{. . . . . . . (2)}$$

in which:

c = contaminant concentration, ML^{-3}

x = distance along longitudinal axis of the estuary, L

t = time, T

A = cross-sectional area of the estuary, L^2

Q = fresh water flow (upland runoff), L^3T^{-1}
E = longitudinal dispersion coefficient, L^2T^{-1}
K = first order decay constant, T^{-1}

The production, or in this case, decay, term is the rate at which material is produced or consumed by physical, chemical, biochemical or nuclear reaction within the volume element.

For decay according to first order kinetics, the usual kinetics of radioactive decay, this rate of consumption of contaminant is equal to the product of the unit rate, Kc, times the volume, AΔx, within which the reaction is taking place.

The accumulation term completes the inventory by accounting for the net rate of increase or decrease of material upon summation of the rate of inflow, outflow and production. This is equal to the time rate of change of total contamination mass within the reactor volume, AΔx.

The parameters, Q, A, E and K, in most estuaries are functions of space and time. To avoid tenuous mathematical complexity, these parameters are often considered to be constants. This approach, justification of which appears in a later section of the report, has been selected for the analysis used in this report. For the case of constant Q, E, A and K, Equation 2 rearranges to:

$$E\left[\frac{\left.\frac{dc}{dx}\right|_{x+\Delta x}-\left.\frac{dc}{dx}\right|_{x}}{\Delta x}\right]-\frac{Q}{A}\left[\frac{c|_{x+\Delta x}-c|_{x}}{\Delta x}\right]-Kc=\frac{dc}{dt} \qquad \ldots\ldots 3$$

The bracketed terms are average rates of change with respect to x. The limit of Equation 3, as Δx approaches zero, is as follows:

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$$E\frac{d^2c}{dx^2}-U\frac{dc}{dx}-Kc=\frac{dc}{dt} \qquad \ldots\ldots (4)$$

U is equal to Q/A and is the average fresh water velocity. Equation 4 is a linear partial differential equation in x an t and is often referred to as the convection-diffusion equation for non-conservative substances. It has been selected as the defining equation for all subsequent analyses presented in this report.

At this juncture, it is important to note that the concentration, c, is actually a tidal smoothed, area averaged concentration. This means that rather than attempt to define local behavior at any point within a cross-section and during a tidal cycle, the analyst looks at the average concentration over an entire cross-section over a full tidal cycle. Justification of this procedure is given by Kent (1), Harleman and Holley (2), and Lawler (3).

This justification proceeds by starting with the equation of continuity of a single chemical specie (4), in which contaminant concentration is a function of three space dimensions and real time. Dependence on the lateral and vertical space coordinates is replaced by dependence on total cross-sectional area by integrating over the total width and depth. The resulting equation is then integrated over a tidal cycle and change with respect to real time replaced by change with respect to tidal cycle units of time.

In the course of these integrations, several new terms are generated, all of which contribute to the dispersion phenomenon. These are eventually replaced by the overall dispersion flux, $E \frac{dc}{dx}$.

Once contaminants are dispersed over the river channel, the various concentrations at specific points within the cross-section and tidal cycle can be expected to be less than 20% of the tidal smoothed, area averaged value. Figures 2 through 7 illustrate this for salt. The actual variation of salinity across various cross-sections within the reach between Indian Point and Chelsea is Show on Figures, 2, 4 and 6. Figures 3, 5 and 7 show the sinusoidal variation of the area averaged salinity at these sections over a tidal cycle, as well as a linearized plot of this variation.

S-40	3
S-41	Figure 2
S-42	Figure 3
S-43	Figure 4
S-44	Figure 5
S-45	Figure 6
S-46	Figure 7

II. SELECTION OF NUMERICAL VALUES OF PARAMETERS

Numerical values of the parameters E, U and K, which appear in the defining differential equation and therefore control the distribution of any contamination in the estuary, must be chosen for the Hudson River.

1. FRESH WATER DISCHARGE

Fresh water velocity, U, is obtained by dividing fresh water discharge by the river cross-sectional area, A. Fresh water flow into the Hudson is measured at Green Island, at mile point 152, where the tributary drainage area totals 8090 square miles. The drainage area of the Hudson Basin, tributary to the entire River, is approximately 13,370 square miles. Over 95% of this area is located north of Indian Point. Because of the inability to measure directly fresh water flow in tidal waters, the Green Island gage issued to establish lower River discharges. The ratio of tributary drainage areas between Indian Point and the gage is 1.57. Analysis of data developed by the United States Geological Survey (USGS) indicates a most probable value for yield factor of 1.22. All values of lower River flow referred to in this report were established using this ratio, i.e., lower River flow is equal to Green Island gaged flow times 1.22.

The pattern of the long-term monthly flows, shown in Figure 8, is indicative of the general variation of River discharge. During the months of March through May, the flow averaged 29,000 cfs or almost 3.5 times the average discharge during the months from June through October. This is equivalent to the statement that the volume of fresh water discharged during the spring months is in exces of twice the volume discharge during the subsequent five-month period.

Figure 1 and Equation 4 indicate that as fresh water velocity decreases, given a fixed value of the longitudinal dispersion coefficient, the dispersion effect increases. Therefore, contaminant concentration values in the region above Indian Point can be expected to increase as flow decreses. Furthermore, due to increased salinity intrusion during periods of low fresh water flow, the longitudinal dispersion coefficient, which is strongly dependent on salinity-induced circulation, can be expected to increase in the upper region of the River. For these reasons, analysis of the effect of pollutants on the River require that drought flows by selected in assigning values of U.

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S-48 Figure 8

Figure 9 shows a statistical analysis of Hudson River drought flows for the years 1918 through 1964. For drought durations of one week (seven consecutive days), and one month, a plot of flow versus the percent of the time such flow can be expected to occur is given. For example, Figure 9 indicates, for a duration of one week, a flow of 2630 cfs can be expected to occur 5% of the time or once in 20 years.

It should be noted that the response of the Hudson to area-wide droughts is significantly different from that of individual, smaller-sized basings in the region. The difference can be attributed to the size and number of sub-drainage areas within the overall basin and the degree of regulation obtained from up-River storage facilities, such as the Sacandaga and Indian Lake reservoirs.

2. <u>CROSS-SECTIONAL AREA</u>

Figure 10 shows 10 shows the variation of cross-sectional area with distance above the Battery. Variation is erratic and as such is not amenable to simple mathematical description; i.e., as an elementary function of distance. Between Indian Point and Chelsea, the area varies from a minimum of 120,000 square feet just north of Bear Mountain Bridge to a maximum of 175,000 square feet at the mouth of Newburgh Bay. The average area over this 22 mile river reach is 140,000 square feet; this number has been selected as the value of the constant parameter, A, in Equation 3.

3. <u>LONGITUDINAL DISPERSION. COEFFICIENT</u>

The value of the longitudinal dispersion coefficient at any point within the salt-intruded reach of the River can be conveniently obtained by analysis of salinity profiles. The limiting form of Equation 2 for the case of a conservative substance such as salt, and non-constant values of Q, A and E, is:

$$\frac{1}{A}\frac{d}{dx}\left[EA\frac{dc}{dx}-Qc\right]=\frac{dc}{dt} \qquad \ldots\ldots (5)$$

If the variation of salinity with x and t is know, the derivatives $\frac{\partial c}{\partial x}$ and $\frac{\partial c}{\partial t}$ may be obtained graphically or numerically. Equation 5 can then be used to compute the value of E at any point within the saline reach of the River.

This procedure requires that a number of profiles be available so that the time derivative, $\frac{\partial c}{\partial t}$, can be computed and also requires that the value of Q, now a time and distance dependent function, controlling the intrusion, be known. This

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S-50 Figure 9

S-51 Figure 10

latter requirement poses some difficult in evaluating Hudson River dispersion. Fresh water flow can only be measured at Green Island, above the tidal region, and the attenuating effect of tidal mixing on time-variable flows is not known.

These difficulties have been avoided by recognizing that drought flows in the Hudson remain relatively constant for extended periods of time; Q, and therefore U, are known and the steady Q gives rise to steady salinity profiles during these periods. Under these conditions, the net flux of salt in the River must be zero since there is no sink or source of salt within the estuary. Equation 5 then reduces to:

$$E\frac{dc}{dx} - Uc = 0 \qquad \ldots\ldots (6)$$

Rearrangement of Equation 6 yields a solution for the dispersion coefficient.

$$E = U\left[2.303 \quad \frac{d \log c}{dx}\right]^{-1} \qquad \ldots\ldots (7)$$

Numerical values of $\frac{d \log c}{dx}$ may be obtained by graphical differentiation of a semi-logarithmic plot of salinity versus distance. U(x) is equal to the flow associated with that profile, divided by the area, A(x), at the point in question. Typical steady state salinity profiles are shown in Figure 11. Values of E, computed as described above, are shown in Figure 12 for these and several other drought profiles.

Figure 12 indicates that the dispersion coefficient at some points may increase as flow decreases whereas, at other points, the reverse may occur. For example, at mile point 20, the value of E, during the 1964 drought flow of 4100 cfs, was 12,000 sf/sec and, during the 1959 drought flow of 8700 cfs, was 6000 sf/sec. On the other hand, at mile point 50, E in 1964 was 4200 sf/sec and, in 1959, was 5000 sf/sec.

These phenomena can be explained in terms of the mechanisms contributing to longitudinal dispersion. In the lower part of the saline region, under drought conditions (less than 12,000 cfs), salinity-induced circulation, which depends strongly on the salt concentration, is the predominating mechanism, whereas, toward the end of the intrusion, this saline effect is less predominant and also less variable. The relative contribution of fresh water flow to the dispersion characteristics of the River increase as the absolute contribution of the salinity

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S-53 Figure 11

S-54 Figure 12

decreases. Thus, increases in fresh water flow can, under some conditions, outweigh the corresponding decrease in salinity, the net effect being an increase in the dispersion coefficient.

Under other conditions, the reverse is true and a decrease in the dispersion coefficient in the presence of an increased flow will be observed. Details for these phenomena and a quantitative method for the prediction of E(x) in the Hudson River as a function of flow are more fully discussed in previous reports (5), (6).

The determination of E as a function of x has been presented to justify the use and selection of constant values of E in this report. A choice of E equal to the maximum value of E(x) within the reach between Indian Point and Chelsea will result in a conservative analysis for the following reasons:

(1) As Chelsea is approached, the true value of E will fall below this maximum, causing the actual contaminant concentration to be lower than that predicted by constant parameter analysis.

(2) The predicted downstream flux will be less than the actual downstream flux because the true E values, in this region, are larger than the constant E. Thus, the predicted value of the fraction of total contamination discharge moving upstream will be greater than the actual value of this fraction.

These qualitative statements can be seen more clearly by reference to Figure 1.

Figure 12 indicates that maximum E in the reach between Indian Point and Chelsea occurs between mile points 45 and 50. Accordingly, the values of E for this analysis have been selected by obtaining the average E between mile points 45 and 50 for any given flow. A second choice of E has been made by obtaining the average between mile point 43 and 65 (Indian Point and Chelsea).

The average value of E over a finite length of River is obtained by application of the mean value theorem for derivatives to Equation 7. This yields:

$$\left[\frac{E}{U}\right]_{AVG} = \left[2.303\,\frac{\Delta \log c}{\Delta k}\right]^{-1} \qquad \text{. . . . . . (8)}$$

$$E_{AVG}\;8\;U_{AVG}\left[2.303\,\frac{\Delta \log c}{\Delta k}\right]^{-1} \qquad \text{. . . . . . (9)}$$

A correlation of all available Hudson River salinity and flow data is shown on Figure 13. Values of E used in this report have been computed by application

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S-56 Figure 13

of Equation 9 to these data. For example, at a flow of 4000 cfs, the computation for average E between mile points 43 and 65 is:

$$E = \left[\frac{4000}{141{,}300}\right] \cdot \left[\frac{2.303\,(\log 7000 - \log 2200)}{[-43(-65)]\,5280}\right]^{-1}$$

= 2830 sq. ft/sec
= 8.74 sq. mile/day

Correspondingly, for the same flow, the average E between mile points 45 and 50 is:

$$E = \left[\frac{4000}{123{,}500}\right] \cdot \left[\frac{2.303\,(\log 6500 - \log 5400)}{-45-(-50)}\right]^{-1}$$

= 4640 sq. ft/sec
= 14.3 sq. mile/day

Figure 14 shows the variation, with flow, of average E, computed by Equation 9 as shown above.

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S-58 Figure 14

III. EFFECT OF CONTINUOUS DISCHARGE ON CHELSEA INTAKE

This section analyzes the effect of a continuous discharge from Indian Point on water drawoff at Chelsea and is subdivided as follows:

1. A steady state of equilibrium analysis
2. A transient analysis or approach to steady state

1. ANALYTICAL DEVELOPMENT FOR STEADY STATE CONDITION

Figure 15 depicts the problem. The defining differential equation is given by Equation 4. Since this equation does not include discharge at Indian Point or drawoff at Chelsea, it will not define behavior across these two planes. For these reasons, the Hudson is divided into three regions, one above Chelsea, one between Chelsea and Indian Point and one below Indian Point. A solution for each region is obtained by application of proper boundary conditions to the general solution of Equation 4.

The steady state form of Equation 4 is:

$$\frac{E d^2 c}{d x^2} - \frac{U dc}{dx} - Kc = 0 \qquad \text{. . . . . . (10)}$$

The general solution of this second order, linear, ordinary differential equation is:

in whic h

$$c = C_1 e^{jx} + C_2 e^{kx} \qquad \text{. . . . . . (11)}$$

$$j = \frac{U + \sqrt{U^2 + 4KE}}{2E}$$

$$k = \frac{U - \sqrt{U^2 + 4KE}}{2E}$$

C_1, C_2 = arbitrary constants

Equation 11 is the form of the general solution for each of the three regions. Designating River velocity above Chelsea as U_1 and below Chelsea as U_2, the general solution in each of the three reaches is written:

$$c_I = C_1 e^{j_1 x} + C_2 e^{k_1 x} \qquad \text{. . . . . . (11a)}$$

$$c_{II} = C_3 e^{j_2 x} + C_4 e^{k_2 x} \qquad \text{. . . . . . (11b)}$$

$$c_{III} = C_5 e^{j_2 x} + C_6 e^{k_2 x} \qquad \text{. . . . . . (11c)}$$

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S-60 Figure 15

in which

$$\left.\begin{matrix} j_1 \\ k_1 \end{matrix}\right\} = \frac{U_1 \pm \sqrt{U_1^2 + 4KE}}{2E}$$

$$\left.\begin{matrix} j_2 \\ k_2 \end{matrix}\right\} = \frac{U_2 \pm \sqrt{U_2^2 + 4KE}}{2E}$$

To evaluate the six arbitrary constants, six boundary conditions are necessary. These are developed as follows:

1. The contaminant can be expected to reach negligible concentrations before passing out of the estuary into the ocean. This is not due to any diluting effect of the ocean, but rather because the distance between Indian Point and New York Harbor is sufficiently long to permit virtually complete disappearance of contaminant originating at Indian Point by the time this contaminant reaches the Harbor. This means that the downstream end of the estuary has no influence on contaminant distribution in the estuary. The estuary may therefore be considered to be infinitely long and the first boundary condition may be written:

$$C_{III}\Big|_{x=\infty} = 0 \qquad \text{BC \#1}$$

2. In the upsteam region, convection opposes dispersion and the distance from Indian Point to the upper end of the estuary is even greater than the distance from Indian Point to the lower end. For these reasons, the statements concerning BC #1 are even more applicable here and the second boundary condition is written:

$$C_I\Big|_{x=-\infty} = 0 \qquad \text{BC \#2}$$

3. Although Equation 10 does not define behavior across sections at Indian Point and Chelsea, and discontinuity in some derivative will occur at these points, the contaminant concentration itself is continuous and therefore single-valued at all points. This fact gives rise to the third and forth boundary conditions:

$$C_I\Big|_{x=a} = C_{II}\Big|_{x=a} \qquad \text{BC \#3}$$

$$C_{II}\Big|_{x=0} = C_{III}\Big|_{x=0} \qquad \text{BC \#4}$$

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4. To describe the behavior at the boundary between regions II and III, a material balance about the plane of discharge is constructed as shown on Figure 15. The steady state material balance is written:

$$\left[Q_2 c_{II} - EA\frac{dc_{II}}{dx}\right]_{x=\frac{\Delta x}{2}} + {}^{qr}c_{II} + {}^{q}IP\ {}^{c}IP - {}^{qr}c_{II} - \left[Q_3 c_{III} - EA\frac{dc_{III}}{dx}\right]_{x=\frac{\Delta x}{2}} = K\bar{c}A\Delta x \qquad \text{. . . . . . (12)}$$

in which Q_2 = River flow above Indian Point

^{q}IP = volumetric discharge from plant

Q_3 = Q_2 + ^{q}IP = net River flow below Indian Point

^{c}IP = concentration of plant contaminants prior to introduction to recirculating flow

qr = recirculating River flow through plant

Simplifying Equation 12 and taking the limit as $\Delta x \rightarrow 0$ yields:

$$^{q}IP\left[{}^{c}IP - {}^{c}II\right]_{x=0} - EA\left[\frac{{}^{dc}II}{dx} - \frac{{}^{dc}III}{dx}\right]_{x=0} \qquad \text{. . . . . . (13)}$$

In reality, virtually all of the flow from Indian Point is recirculated from the River. Therefore $^{q}IP <<< Q_2$, and for all practical purposes $Q_2 = Q_3$. Call ($^{q}IP \cdot {}^{c}IP$), W, the continuous load on the River, take the limit of Equation 12 and obtain for the fifth boundary condition:

$$W = EA\left[\frac{{}^{dc}II}{dx} - \frac{{}^{dc}III}{dx}\right] \qquad \text{BC \#5}$$

Notice that the first derivatives of the contaminant concentration are discontinuous at the point of discharge. This behavior is shown clearly by the contaminant profile in Figure 1.

5. The behavior at the boundary between regions I and II is developed similarly. A material balance about the plane of drawoff is constructed in Figure 15 and is written:

$$\left[Q_1 c_I - EA\frac{dc_I}{dx}\right]_{a-\frac{\Delta x}{2}} - q_c c_a - \left[Q_2 c_{II} - EA\frac{{}^{dc}II}{dx}\right]_{a+\frac{\Delta x}{2}} - K\bar{c}A\Delta x = 0 \qquad \text{. . . . . . (14)}$$

in which Q_1 = River flow above Chelsea

q_c = drawoff at Chelsea

Q_2 = Q_1 - qc River flow below Chelsea

c_a = contaminant concentration at Chelsea

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As Δx approaches zero, $^{c}I = {}^{c}II = {}^{c}a$ and Equation 14 becomes:

$$\left.\frac{dc_I}{d_x}\right|_{x=a} = \left.\frac{dc_{II}}{dx}\right|_{x=a} \qquad \text{BC \#6}$$

Notice, in the case of drawoff from the River, the concentration of contaminant in the withdrawn flow is identical to the concentration of contaminant in the River at the point of drawoff. In the case of discharge to the River, the contaminant concentration in the discharged flow is much larger than in the River at this point. Thus, in the case of drawoff, the defining differential equation does not hold across the point of drawoff because the River flow is changed, while in the case of discharge, it does not hold because of the imposition of a net load on the River.

Substitution of Equations 11a, b, c into these six boundary conditions yields values for the six arbitrary constants. The explicit solutions for contaminant concentration becomes:

$$c_I = \frac{W e(j_2 - j_1)}{AE(j_1 - k_2)} a + j_1 x \qquad \text{. . . . . . (15)}$$

$$c_{II} = \frac{W}{AE}\left[\frac{e^{j_2 x}}{j_2 - k_2} + \frac{(j_2 - j_1)\, e^{(j_2 - k_2)} \quad a + k_2 x}{(j_2 - k_2) \quad (j_1 - k_2)}\right] \qquad \text{. . . . . . (16)}$$

$$c_{III} = \frac{W}{AE}\left[\frac{(j_1 - k_2) + (j_2 - j_1)\, e^{(j_2 - k_2)a}}{(j_2 - k_2) \quad (j_1 - k_2)}\right] e^{k_2 x} \qquad \text{. . . . . . (17)}$$

For the case of no decay, K = 0, and:

$$j_1 = \frac{U_1}{E}$$

$$j_2 = \frac{U_2}{E}$$

$$k_2 = 0$$

For this case, the concentrations at x = 0 (Indian Point) and at x = a (Chelsea) are, respectively:

$$C_O = \frac{W}{Q_1}\left[\frac{U_1}{U_2}\left(1 - e\frac{U_2.a}{E}\right) + e\frac{U_2.a}{E}\right] \qquad \text{. . . . . . . (18)}$$

$$C_a = \frac{W}{Q_1} e \frac{U_2.a}{E} \qquad \text{. . . . . . . (19)}$$

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For no drawoff at Chelsea, Equation 18 and 19 reduce to the simple case of discharge of a conservative contaminant at x = 0; i.e., $U_1 = U_2$, $Q_1 = Q_2 = Q$ and:

$$C_O = \frac{W}{Q} \qquad \text{. . . . . . . (20)}$$

$$C_a = \frac{W}{Q} e^{\frac{U}{E}a} \qquad \text{. . . . . . . (21)}$$

The ratio of concentration at Chelsea to concentration at Indian Point is:

$$\frac{{}^{C}\text{Chelsea}}{{}^{C}\text{Indian Point}} = \frac{C_a}{C_o} = \frac{1}{1 + \frac{U_1}{U_2}\left(e^{\frac{U_2 . a}{E}} - 1\right)} \qquad \text{. . . . . . . (22)}$$

For the case of no drawoff at Chelsea, this reduces to $e^{\frac{U_a}{E}}$.

2. TRANSIENT CONDITION

Subsequent to commencement of a steady, continuous discharge, a time lag occurs before steady state profiles, described by Equation 15 through 21, are established. To determine concentration build-up as a function of time as well as of space, an unsteady state analysis of Equation 4 must be made. Such an analysis has been judged necessary in this study, not only to establish the rapidity of approach to steady state, but also to serve as a basis for a computer solution of the maximum permissible continuous release when radioactive decay is taken into account (7).

Analysis shows that the 100 mgd Chelsea draw has only a slight effect on equilibrium concentration at Chelsea. The same can be expected during the approach to equilibrium so that transient analysis without consideration of draw-off was used. This has been developed previously in considerable detail (8). The final equation for the distribution of contaminant upstream of the point of waste discharge is:

$$
\begin{aligned}
{}^{c}I(x,t) = \frac{W}{2Q\sqrt{1+\frac{4KE}{U^2}}} & \left[E\,x\,P\left[\frac{U}{2E}\left(1+\sqrt{1+\frac{4KE}{U^2}}\right)x\right] \;.\; ERFC\left(\frac{-x}{\sqrt{4Et}} - \sqrt{\frac{U^2+4KE}{4E}t}\right) \right. \\
& \left. - EXP\left[\frac{U}{2E}\left(1-\sqrt{1+\frac{4KE}{U^2}}\right)X\right] \;.\; ERFC\left(\frac{-x}{\sqrt{4Et}} + \sqrt{\frac{U^2+4KE}{4E}t}\right) t \right]
\end{aligned}
\qquad \text{. . . . . . (23)}
$$

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The corresponding steady state solution, given by Equation 16 when ʲ1 = ʲ2 (no drawoff, $U_1 = U_2$), is:

$$^{c}I\ (x) \quad = \quad \frac{W}{Q\sqrt{1+\frac{4KE}{U^2}}}\, e^{\frac{U}{2E}\left[1+1\sqrt{+\frac{4KE}{U^2}}\right]x} \qquad \ldots\ldots\ (24)$$

The ratio of the transient response to the equilibrium response is:

$$\frac{^{c}I(x,t)}{^{c}I(x,\infty)} = \frac{1}{2}\left[ERFC\left[\frac{-x}{\sqrt{4Et}} - \sqrt{\frac{U^2+4KE}{4e}}\,t\right] - EXP\left[-\frac{U}{E}\sqrt{1+\frac{4KE}{U^2}}\,x\right].\,ERFC\left[\frac{-x}{\sqrt{4Et}} + \sqrt{\frac{U^2+4KE}{4E}}\,t\right]\right] \qquad \ldots\ldots\ (25)$$

For the case of no decay, Equation 25 reduces to:

$$\frac{^{c}I(x,t)}{^{c}I(x,\infty)} = \frac{1}{2}\left[ERFC\left[\frac{-x}{\sqrt{4Et}} - \sqrt{\frac{U^2 t}{4E}}\right] - EXP\left[-\frac{U}{E}x\right]\ .\ ERFC\left[\frac{-x}{\sqrt{4Et}} + \sqrt{\frac{U^2 t}{4E}}\right]\right] \qquad \ldots\ldots\ (26)$$

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IV. INSTANTANEOUS RELEASE

This case represents the condition of an accidental spill of radioactive contaminant to the River. A slug of material is released over a short time interval, which for practical purposes can be assumed to be instantaneous. The object is to determine the time of appearance of and the value of maximum concentration at Chelsea.

1. PREVIOUS STUDIES

Studies of the effect of instantaneous release of conservative substances at Indian Point were conducted on the Hudson River Model at the Waterways Experiment Station, Vicksburg, Mississippi, circa 1962 (9). Figure 16 is a reproduction of Plate 30, reference 9, and shows the distribution of conservative dye, released over a single tidal cycle at Indian Point, for a River flow of 12,000 cfs. Notice that the spread is asymmetrical, favoring the downstream direction. This documents the variable nature of the dispersion coefficient and the fact that it increases in the downstream direction, as shown previously in Figure 12. A more detailed analysis of these data, in terms of the mechanisms which cause E to vary, may be found in references 5 and 6.

The occurrence of maximum upstream E values between mile points 45 and 50 is demonstrated by Figure 16. Within this reach a decreasing slope, particularly for tidal cycles 15 through 30, can be seen, indicative of greater spreading or longitudinal dispersion. For a flow of 12,000 cfs, salinity is well below mile point 55, the approximate location of the mouth of Newburgh Bay, and therefore not available to induce circulation, i.e., increase E. Below this point the channel narrows, the velocity is higher, and the downstream-directed convection strong. However, the rate of tidal energy dissipation, besides salinity-induced circulation, the other major cause of dispersion, is relatively high and dispersion is enhanced and the dye moves up this far.

Tidal energy dissipation in the larger expanse of the bay is relatively low; without salinity-induced circulation present, dispersion becomes negligible and is overpowered by downstream-directed convection. Thus, at the flow of 12,000 cfs, dye does not appear above mile point 55.

At drought flows, of course, salinity is present for north of this point; significant dispersion, at these times, can be expected in the vicinity of Chelsea.

S-66	15
S-67	Figure 16

2. <u>CONSTANT PARAMETER ANALYSIS</u>

Draw off at Chelsea is not considered; the results of the continuous analysis indicate this is not a serious omission. Detailed analysis of the instantaneous release for constant River characteristics has been developed previously (8); a brief outline of the development is given here.

The defining differential equation is Equation 4. The initial and boundary conditions are developed as shown on Page 9 and are:

Initial Condition: $C\big|_{t=0} = 0, -\infty \le x \le \infty$

Boundary Conditions #1, #2: $C\big|_{x=\pm\infty} = 0$, all t

Boundary Conditions #3 $C\big|_{x_- \to 0} = C\big|_{x_+ \to 0}$ all t

Boundary Conditions #4 $AE\left[\frac{dc}{dx}\Big|_{x-\to 0} - \frac{dc}{dx}\Big|_{x+\to 0}\right] = f(t),\ t>0$

f(t) in B. C. #4 is the delta function and is written:

$$f(t) = \begin{cases} \frac{M}{\Delta t}, & 0 < t < \Delta t \\ 0, & \Delta t < t \le \infty \end{cases}$$

in which M = Mass of contaminant released

The Laplace Transform Solution of Equation 4, subject to the above conditions, yields:

$$C(x,t) = \frac{M}{2A\sqrt{\pi E t}} e^{-\frac{(x-Ut)^2}{4Et} - Kt} \qquad \text{. . . . . . (27)}$$

To compute the dilution effect only, set K – 0. Equation 27 becomes:

$$C(x,t) = \frac{M}{2A\sqrt{\pi Et}} e - \frac{(x-Ut)^2}{4Et} \qquad \text{. . . . . . (28)}$$

The maximum value of C(x,t) at a given x is desired. Differentiate Equation 28 with respect to t and equate the results to zero to determine the time at which the maximum concentration occurs. This procedure yields:

$$t_{critical} = \frac{E}{U^2}\left[-1 + \sqrt{1 + \frac{U^2 x^2}{E^2}}\right] \qquad \text{. . . . . . . (29)}$$

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VIII. REFERENCES

1. Kent, R. “Diffusion in a Sectionally Homogeneous Estuary.” Paper No. 3392, Transactions, American Society of Civil Engineers, Vol. 128, Part III (1963).

2. Holley, E. R., Jr. and Harleman, D. R. F. “Dispersion of Pollutants in Estuary Type Flows.” Report No 74, Hydrodynamics Laboratory, Massachusetts Institute of Technology, Cambridge, Mass. (January 1965).

3. Lawler, J. P. “Mathematical Analysis of Estuarine Pollution,” in “Engineering Aspects of Marine Waste Disposal.” Environmental Health Sciences and Engineering Training Course Manual, R. A. Taft Sanitary Engineering Center, Cincinnati, Ohio (1965).

4. Bird, R. B., Stewart, W. E. and Lightfoot, E. N. “Transport Phenomena.” John Wiley and Sons, New York, N. Y. (1960).

5. “Evaluation of Waste Diffusion in the Hudson River at Peekskill.” Report to Standard Brands, Incorporated by Quirk, Lawler and Matusky, Water Resources Engineers, New York, N. Y. (December 1964).

6. “Hudson River Dispersion Characteristics.” Progress Report to Consolidated Edison Company of New York, Incorporated by Quirk, Lawer and Matusky, Water Resources Engineers, New York, N. Y. (October 1965)

7. “Indian Point Station – River Diffusion Study.” Consolidated Edison Company of New York, Incorporated – Memorandum from C. A. Larson to W. J. Cahill, Jr. (April 26, 1966).

8. Lawler, J. P. “Differential Equations” in “Stream and Estuarine Analysis.” Training Manual, Tenth Summer Institute in Water Pollution Control, Manhattan College, New York, N. Y. (June 1965).

9. "Joint Pollution Studies in the New York Harbor Model." Miscellaneous Paper No. 2-558, U. S. Army Engineer Waterways Experiment Station, Vicksburg, Miss. (February 1963).

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K A R L R. K E N N I S O N

CIVIL AND HYDRAULIC ENGINEER
361 CLINTON AVE., BROOKLYN, N.Y.

Mr. G. R. Milne Nov. 18, 1955
Mechanical Engineer
Cons. Edison Co. of N.Y.
4 Irving Place
New York 3, N. Y.

Dear Sir :

You have described to me the general features of the atomic-energy power plant which you are planning to construct on the east bank of the Hudson River below Peekskill. I under-stand that you wish me to report on such hydrologic features of the site as may effect your plans.

From the information that you have made available to me I conclude that the most useful information I can give you is that which relates to the amount and character of the flow in the river. At the proposed site the river has a width of about 4500 to 5000 feet, a maximum depth of 55 to 75 feet at less than 1000 feet off shore, and a cross-sectional area of about 165,000 to 170,000 square feet. Sheet 1 shows a number of cross sections of the river, plotted from the U.S.C. &G.S. charts, at intervals of 1500 feet, from 3750 feet upstream to 5250 feet downstream from the proposed plant.

At this site the effect of the tides is all important and so far outweighs any other consideration that, at least for present purposes, the information already available on the day-by-day variation of the runoff from the tributary watershed is adequate.

On Sheet 2 I have plotted an approximate flow-duration curve from data I had already calculated covering a period of

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- 2 -

17 years.

An average rate of about 26000 cfs may be expected to be exceeded 20% of the time

"	"	"	"	"	15250	"	"	"	40%	"	"	"
"	"	"	"	"	10500	"	"	"	60%	"	"	"
"	"	"	"	"	7000	"	"	"	80%	"	"	"

for say 2 % of the time the rate may be as low as 4000 cfs

However as above indicated the ebb and flow of the tide is the all important consideration. The river is tidal to as far upstream as Troy. Its hourly behavior in the tidal range varies throughout its length. The U. S. Coast & Geodetic Survey has tabulated a great deal of information from which a general picture of conditions off the shore at the proposed site can be obtained.

On Sheet 3 I have plotted the data, as they are applicable to this particular site. This indicates that the elevation of the water surface is so responsive to the tidal cycle that the average rate of flow, or runoff from the tributary watershed, has relatively little effect on the velocity past the site. I conclude that it is this velocity and the resulting volume of flow available for mixing and dilution in which you are primarily interested. In the limited time at my disposal I can only draw general conclusions. These may be adequate for present purposes. You could obtain better information by running a series of tests on surface and sub-surface floats, at varying distances off shore, throughout the tidal cycle.

The velocity recorded by the U.S.C.&G.S. is that in midstream at or near the surface. In order to be on the safe side in drawing conclusions, I have assumed that 80 % of this velocity represents the average vertically from surface to bottom, and that 80 % also represents the average horizontally from side to

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side, hence that roughly 64 % represents the average over the entire cross section. I have also assumed that 15 % of the total cross section, or a stretch about five or six hundred feet wide off shore, is all that should be used in considering the initial mixing or diluting effect. In making this assumption I am governed to some extent by Hazen's studies relative to the off-shore distance of Poughkeepsie's water intake to avoid direct contamination by its sewage. I have further assumed that the velocity in this off-shore stretch is only 60% of the midstream velocity, hence that roughly 48% represents the average over the cross section of this off-shore stretch.

On Sheet 4 I have shown the result of these assumptions, which, as above stated, are believed to be on the safe side in considering the direct effect of mixing or dilution of your wastes. This emphasizes the all-important effect of the tides, the quantity available for dilution varying in about three hours from a maximum of eight or ten million gallons per minute to nothing.

Although you will have to put up with this variation as far as your continuous cooling water circulation is concerned, it does point to the desirability of incorporating in your design a method of controlling the time for the discharge into the cooling water outlet of any and all waste that is to any extent radio-active. I would say that this should be done in any event for the drainage from your routine and emergency demineralizers, and it might well be done also for drainage from all areas liable to accidental contamination.

From your estimate of the extent of dilution already

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accomplished in the demineralizer waste overflow, I trust you can get an approximate figure for the dilution that may result in the river off shore, and can compare this with what you may find necessary or desirable for adequate protection of fish life or of the fish eating public.

As far as the effect on public water supplies is concerned, the use of the Hudson River for water supply, other than condenser cooling, it ver limited. The nearest municipality involved is Poughkeepsie, 30 miles or more upstream, and even at that distance threatened at times with the problem of salinity. There is no likelihood that in the future any nearer municipality will take its domestic water supply from the Hudson. In fact the tendency is the other way, and the more remote muncipalities of Catskill and Hudson have abandoned earlier supplies taken from the river.

As far as the effect on ground water is concerned, you have acquired an ample area of surrounding land. I can see no possibility of any deletorious effect.

I trust that this information which I have assembled in the limited time available will be helpful to you. If from these approximate figures there appears to be any question as to the adequacy of the safety factor in dilution, you may, as above stated, require additional information from float test.

From what you have told me about your proposed designs and methods of operation, I suspect that there is no real question of safety but only one of public relations – avoidance of even the appearance of danger.

Very truly yours,

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[Historical Information]

Consolidated Edison Company of New York, Inc.

EVALUATION OF FLOODING CONDITIONS AT INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

April, 1970
Revision of Report of February, 1969

Quirk, Lawler & Matusky Engineers
Environmental Science & Engineering Consultants
505 Fifth Avenue
New York, New York 10017

QUIRK, LAWLER & MATUSKY ENGINEERS

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the investigation was conducted by Quirk, Lawler & Matusky Engineers under the direction of Dr. John P. Lawler. Mr. Karim A. Abood of Q.L & M served in the capacity of project engineer and wrote this report. Dr. Karel Konrad of Q.L. & M. contributed materially to this study and performed any of the calculations reported herein.
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-S-1-

SUMMARY OF FINDINGS, CONCLUSIONS & RECOMMENDATIONS

1. Consolidated Edison Company of New York, Incorporated, (Con Ed) plans to build a third nuclear generating unit at its Indian Point site. The site is located in the town of Buchanan, Westchester County, New York and lies along the east bank of the Hudson River, some 43 river miles above New York City Harbor. The proposed facility will have a guaranteed output of 965 megawatts electric.

2. The Atomic Energy Commision (AEC), by virtue of the Atomic Energy Act, is empowered to review and issue licenses to construct and operate nuclear power plants. The AEC licensing regulations include submission of a Preliminary Safety Analysis Report (PSAR). site safety criteria for this report requires an analysis of the

area's hydrology.

3. Quick, Lawler & Matislu Engineers (Q/L. & M.), Environmental Science & Engineering Consultants were retained by Con Ed to study the hydrology of the Indian Point site and to determine the maximum water-surface elevations that can occur as a result of possible flooding conditions at the site. The establishment of such flood levels is necessary to provide adequate protective measures during flood conditions.

Three previous studies had been performed prior to this investigation. The results of those studies were submitted to the AEC. The first appears in Supplement 10, Docket 50-286; the second is Q. L. & M's report of February, 1969 and the third is Q.L. & M's Summary of March, 1970.

The present study was conducted along the lines of a number of specific guidelines developed during several meetings with AEC staff personnel. The study is of a comprehensive and detailed nature involving careful analysis and accepted methodology and follows closely the procedures utilized by the U.S. Corps of Engineers for estimating the hydraulic events under consideration.

4. Several flooding conditions governing the maximum water elevation at the site, were investigated including:

a. Flood resulting from runoff generated by a Hudson River Probable maximum Precipitation (PMP).

b. Flooding caused by the occurrence of a dam failure concurrent with heavy runoff generated by a Hudson River

Standard Project Flood (SPF). This condition was considered because the previous condition did not result in a dam failure.

c. Flooding due to the occurrence of a Probable maximum Hurricane (PMH) concurrent with spring high tide.

5. The determination of the most severe water surface elevations at the site was based upon the simultaneous occurrence of the above-delineated flooding conditions with several critical boundary conditions in the Hudson River including:

a. Mean Water Elevation
b. High tide Water Elevation
c. Low tide water Elevation
d. Standard Project Hurricane (SPH)
e. Probable maximum Hurricane (PMH)

6. The work required to achieve these objective included:

a. Determination of the Hudson River hydraulic and Hydrodynamic characteristics, which include channel

Geometry, flow resistance, tidal dynamics and basin Hydrology.

b. Derivation of the Maximum Probable Flood (PMF) and Standard Project Flood (SPF) for the Hudson River Basin upstream of the site.

c. Structural and hydraulic analysis of the major dams in the basin to determine their stability and failure possibility under PMP conditions.

d. Determination of the Probable Maximum Hurricane (PMH) and Standard Project Hurricane (SPH) for the New York Harbor area and resulting peak storm surge heights in the Hudson River.

The specific details of these items are given in Chapters II, III, IV and Appendix A respectively. The results of the determination of the maximum water-surface elevations resulting from the flooding conditions of Item 4 and boundary conditions of Item 5 above are presented in Chapter V.

The procedures employed and major findings are summarized below in the following six items (#7 through 12).

7. A PMP of 14 inches in 72 hours and a probable maximum storm (PMS) pattern similar to the one derived by the U.S. Westher Bureau for the adjacent Susquehanna Basin at Danville were used in this study. These were adopted on the basis of similarity in topography,

shape and size between the two neighboring basins.

The 6-hour incremental rainfall depths and related runoffs for the entire basin arranged in a critical time sequence documented in Figure S-1.

The Hudson Basin was divided into 28 subbasins and a flood hydrograph for each subbasin was established using the PMP values corresponding to the selected PMS together with the appropriate rainfall losses and unit hydrograph. To illustrate the procedures employed, a brief description of the development of a typical subbasin flood hydrograph is presented below.

Figure S-2 summarizes the basic flood data for the Catskill Creek Basin used in this study. An isohyetal map of the subbasin appears in the upper right hand corner. The isohyets, shown as broken lines, represent the subbasin portion of the total Hudson Basin PMS. this total PMS is shown in Figure III-8. The rainfall values are for the total storm duration of 72 hours. Twelve sets of such values corresponding to twelve 6-hour subdurations were established.

For each time increment, average rainfall depth over the subbasin was computed. These increments were then rearranged in accordance with a recommended critical time sequence. The appropriate initial and infiltration losses (1" and .05"/hr. respectively) were subtraced to obtain the subassin rainfall excesses (runoff) for each subduration. The resulting rainfall and runoff distributions are shown in the upper left hand corner.

The generaged runoff was applied to the 6-hour unit hydrograph to develop the subbasin flood hydrograph.

with base flows added the flood hydrographs of the 28 subbasins were combined in their proper time sequence and/or routed downstream to give the inflows into the Hudson River from its tributaries.

An example of the flood hydrograph combination and routing is depicted in Figure S-3. the example considers subbasins No. 15, 16 and 17 of the Mohawk River Basin. The flood hyrographs of subbasins No. 15 and 16 were combined to obtain the Mohawk River flood hydrograph at the mouth of Schoharie Creek. The resultant was routed through the river using the classical Muskingum coefficient method to Cohoes. This routed hydrograph was then combined with subbasin No. 17 flood

I-6 Figure S-1

I-7 Figure S-2

I-8 Figure S-3

I-9 -S-4-

hydrograph to establish the total Mohawk River hydrograph at the confluence of the Mohawk River with the Hudson.

the main river inflows were then routed downstream in a similar manner until the flood hydrograph at Indian point was determined.

The routed hydrographs at several locations along the Hudson River are depicted in Figure S-4.

The above-listed steps were verified by predicting flows using actual rainfall-runoff records and then comparing the results with published values at Cohoes, Green Island and the Wallkill River.

The peak discharge thus obtained at Indian Point amounts to 1.1

million cubic feet per second and represents the Probable Maximum Flood (PMF) at the site. The corresponding Standard Project Flood was selected as equal to 50 per cent of this value.

In developing the PMF, every effort was made to make all the necessary assumptions as conservative as possible and the results were closely checked with the U.S. Army Corps of Engineers generalized curves.

The computed PMF agrees reasonably well with its Susquehanna, Delaware and Potomac Rivers counterparts. It is more than 60 per cent higher than the statistically evaluated 10,000 year flood at Indian Point.

8. To obtain the most severe flooding conditions, the heavy rainfall centers of the selected PMS were deliberatedly placed over the basins of the largest two reservoirs in the Hudson River Basin. These reservoirs are the largest regulating feature in the basin (Sacandaga) and New York City's reservoir at Ashokan. These reservoirs are located some 180 and 75 miles upstream of the site respectively

Inflows to Sacandaga and Ashokan Reservoirs resulting from placing the heavy rainfall centers of the PMS over their basins were routed through the reservoirs. The results clearly indicate that both dams have sufficient storage to pass this flood without danger of overtopping or piping, i.e. errosion of the surface of the downstream face. The specific details follow.

9. The Sacandaga Reservoir Dam, know as the Conklingville Dam, is extremely stable and has proven its ability to withstand sever

I-10

Figure S-4

I-11 -S-5-

earthquakes and floods. It has an ample cross-section with a very broad base and a substantial freeboard of 24 feet above the crest of spillway at elevation 771.0.

The dam is relatively impervious, made of excellent material consisting of glacial drift, has an impermeable earth core and is founded on rock. It has an unusually large toe, sizable cover of riprap and sufficient outlet works and spillway capacity.

when routing the probable maximum inflow through the reservoir, the maximum stage reached was elevation less than elevation 784.0, more than 11 feet below the crest of the dam.

Computation of the location of the free water surface within the

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dam showed that the seepage water caused by the maximum possible water elevation would collect in the rock toe and would be safely discharged without carrying soil grains with it. This was the case even when a highly conservative and hypothetical case consisting of the maximum possible stage upstream and a 20 foot water depth downstream of the dam.

These estimates are based upon several conservative assumptions including:

a. Outlet works would start operation when the reservoir level reaches elevation 771.0, the crest of spillway elevation.
b. High initial reservoir level at elevation 768.0 prior to the beginning of the PMS.
c. Inoperative power house pumps.
d. Homogeneous material, i.e. no advantage was taken of the deflection in seepage curve caused by the relatively impermeable core.
e. Presence of water downstream of the dam.

10. The study of the effect of routing the Esopus Creek flood hydrograph (upstream of the Ashokan Dam) resulting from the Hudson River PMS through the Ashokan Reservoir indicates that the dam would not fail from overtopping or seepage. The maximum possible reservoir elevation would be below the crest of the masonry core in the dikes and more than 11 to 15 feet below the crest of the east and west basin reservoirs respectively. The dam has proven its ability to withstand such high elevation on March 31, 1951 when the elevation in the east basin reached a record high of 592.23 feet, about 1.5 feet higher than the elevation resulting from the Hudson River PMP.

I-12 -S-6-

11. Since the results of the previous item indicated that the Ashokan Dam would not fail, a more critical condition resulting in a dam failure was investigated. The reason behind consideration of a more severe Ashokan Dam condition stems from the fact that the Ashokan Reservoir contains the largest volume of stored water in the basin within less than 100 miles from the Indian Point site.

This flooding condition consists of the simultaneous occurrence of a probable maximum flood over the Esopus Creek Basin upstream of the Ashokan Dam and a Standard Project Flood over the rest of the Hudson River Basin.

this condition resulted in a peak reservoir inflow of 200,000 cfs causing a dam failure as a result of a result of a reservoir elevation higher than the crest of the masonry wall in the west dike and earth part of the main dam.

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However, the structural analysis of the masonry part of the main dam showed that this part would not fail from overturning and sliding with minimum factors of safety of about 1.1. and 1.9 respectively.

The failure of the Ashokan Dam would cause a maximum reservoir outflow of 2.6 million cfs and create a seven billion cubic feet lake extending from the dam to Glenerie Falls some 5 miles west of the Hudson River. The lake would have a total length of 14 miles and a surface width ranging form 4,000 to 1,800 feet. It would also have a control section at Glenerie Falls. The outflow of this lake would be about 620,000 cf.s

This outflow was combined with the flood hydrographs resulting from Esopus Creek PMP and Hudson River PMP and routed downstream to Indian Point. Much of the discussion on the determination of the PMF at Indian Point summarized in Item 7 above applies here, mutatis mutandis.

the maximum discharge at Indian Point resulting from this condition would be 705,000 cfs.

12. The maximum water-surface elevations at the site resulting from the flooding conditions summarized in item 4 concurrent with the initial conditions of Item 5 above were determined. These conditions were conservatively grouped in seven different ways on the basis of a simultaneous occurrence of an appropriate set of flooding, hurricane and tidal conditions.

I-13

-S-7-

As in the case of flooding from dam failure, the tidal variation in the Hudson Estuary was treated as an integral part of the system and its influence was simultaneously coupled with the other hydraulic occurrences. The results of this study suggest that the river above the Tappan Zee Bridge, some 27 miles above the Battery, would experience relatively small tidal variations under the PMP conditions. However, to examine the effect on Indian Point the upstream extent of tidal influence was extended to the site.

In the evaluation of the maxium water surface elevation at the site, the wind produced local oscillatory short period waves were also included. The computed stages were increased by one foot to account for this effect.

The standard Step Method was employed in this investigation to determine the flow profiles and stages at the site. The computations were initiated at the Battery, the mouth of the Hudson River. The results were checked with a more refined and newly developed method.

The seven groups of flooding, hurricane and tidal conditions and the corresponding stages at Indian Point are summarized in Table S-1

Cases No. 1 through 3 consider the Indian Point stage at peak discharge due to runoff generated by the PMP over the basin and mean, high and low tide stages at the Battery. The flow profiles corresponding to these cases are depicted in Figure S-5.

Case No. 4 considers the maximum river stage at the site resulting from the dam failure concurrent with the Hudson River SPF summarized in Item 11 above.

To carry the degree of severity a step further, a more critical boundary condition consisting of the peak storm surge at the Battery resulting from the SPH was imposed on the flooding condition of steps No. 5 and 6.

To maintain the selection of severe conditions and in accordance with the high degree of conservatism adopted in this report, the SPH based on the transposition of the September 1944 hurricane rather than the less critical SPH of the 1938 hurricane was used. The 1944 value is 75 per cent higher than the maximum storm surge during hurricane "Donna" which struck on September 1, 2960 and which is the storm having the greatest effect since 1821.

Case No. 6 considers an even more critical set of occurrences con-

I-14.

I-15 Figure S-5
I-16 -S-8-

sisting of the simultaneous occurrence of the following three severe conditions:

a. PMP over the Ashokan Reservoir resulting in a dam failure
b. runoff generated by the SPP over the Hudson Basin
c. Peak storm surge resulting from the SPH for the New York Harbor area

The simultaneous occurrence of these three conditions is extremely remote. Each one of these three conditions are in themselves unlikely events. The combination of all three decreases, therefore, the likelihood of occurrence significantly.

The flow profiles corresponding to cases No. 4 through 6 are depicted in Figures S-6.

For convenience, the results of routing the PMH through the Hudson River to the site presented in our report are also listed in Table S-1.

Conclusion – These analyses show clearly that the maxiumu elevation at Indian Point due to flooding and wave runup is 15 feet or less.

I-17 Figure S-6
I-18

I. INTRODUCTION

Consolidated Edison Company of New York, Incorporated, plans to build a third nuclear generating unit on the Hudson Rive at Indian Point. The new facility is called Indian Point No. 3 reactor since two other nuclear power units have been authorized for the same site.

The Indian Point site is located in the town of Buchanan, Westchester County, New York. the property lies along the east bank of the Hudson River, some 43 river miles above New York City Harbor.

The proposed facility will have a guaranteed output of 965 megawatts electric. On completion of Unit No 3, the total manufacturer's guaranteed rating of Unit Nos. 2 & 3 will be 2123 megawatts electric.

the Atomic Energy Commission (AEC), by virtue of the Atomic Energy Act, is empowered to review and issue licenses to construct and operate nuclear power plants. The AEC licensing regulations include submission of a Preliminary Safety Analysis Report (PSAR). Site safety criteria for this report requires an analysis of the area's hydrology.
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Quirk, Lawler & Matusky Engineers (QL&M), Environmental Science and Engineering Consultants were retained by Consolidated Edison to study the hydrology of the Indian Point site and to determine the maximum water elevations that can occur as a result of possible flooding conditions at the site of the Indian Point Nuclear Generating Unit No. 3. The establishment of such a flood level at the site is necessary to provide the necessary protective measures during flood conditions.

Preliminary analysis based upon highly conservative estimates were conducted in 1968. The results of that study are delineated in

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Supplement 10, Indian Point Unit No. 3, AEC Docket 50-286.

The following two flooding conditions, which the AEC indicated would govern the maximum water elevation, were investigated:

1. Flooding from maximum rainfall concurrent with dam failure at Sacandaga and Ashokan Reservoirs and flow at ebb tide.

2. Flooding resulting from the probable maximum hurricane concurrent with spring high tide.

In the 1968 report it was noted that a more refined flooding analysis would give more realistic results, and that the hurricane surge analysis was based upon a theory applicable to conditons of open

The 1968 preliminary values were then modified as a result of a new study which considered surrounding topography, land use, river geometry and frictional losses. A report documenting the results of the new study was submitted to the AEC in February, 1969.

Because the 1969 results indicated that the hurricane surge analysis proved to be controlling, specific attention was paid to determining the water elevation resulting from the occurrence of probable maximum hurricane.

After a review of the February, 1969 report, including conferences with AL & M and Con Ed by the AEC, the commission requested that the applicant restudy in depth the runoff resulting from heavy rainfall over the basin drainage area and revise the water elevation computation at Indian Point.

In particular, the AEC objected to the use of the so-called “rational formula” for the runoff computations and the Manning Open Channel Flow Equation for the determination of the water elevation

The AEC suggested a more refined approach including the use of the Corps of Engineers findings and procedures to determine the probable maximum flood and the backwater profile method to predict the water surface elevation resulting from flooding due to probable maximum precipitation over the basin.

A study along the lines of the suggested procedures was instituted in January of this year and involved a thorough evaluation of the maximum predictable stage of the Hudson River at the Indian Point site Major emphasis was placed on the determination of the probable maximum flood, dam failure analysis, influence of the tidal flow and the maximum water elevations for several flooding conditions. Two meetings were held on March 26 and April 21, 1970 between the AEC staff personnel, Con Ed and Q & M to discuss the proposed program and to delineate requirements for further refinements. At these meetings, as number of specific guidelines for the analyses of all pertinent parameters and accepted methodology were developed. The procedures and methods adopted for this study follow closely the developed guidelines and those utilized by the U.S. Corps of Engineers.

The purpose of this report is to present the results of this study. For convenience, the results of the revised probable maximum hurricane study of the February, 1969 report are also included in this report.

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the report is formulated as follows:

1. A general description of the Hudson River Basin and the channel characteristics is given in Chapter II.

2. The specific details and adopted methodology in computing the Hudson River probable maximum flood at Indian Point are presented in Chapter III. The results of the previous flood investigations are also documented in this chapter.

3. chapter IV presents the results of the flood routing and dam failure analysis of the Hudson River Basin Reservoirs.

4. A discussion of the procedures and results of the maximum water elevations for several flooding conditions resulting from a probable maximum flood, dam failure and tidal flow is given in Chapter V.

5. The results of the revised analysis of the water elevation at Indian Point resulting from the occurrence of a probable maximum hurricane concurrent with spring high tide are documented in Appendix A. The material included in this Appendix is essentially that of Chapter III of the February, 1969 report.

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II. DESCRIPTION OF HUDSON RIVER BASIN *

A. Description of the Basis

The Hudson River Basin is located in the eastern part of New York State, draining an area of 12,650 square miles. Most of this area lies in the east central part of New York State, with small portions in Vermont, Massachusetts, Connecticut and new Jersey. A plan view of the Basin is shown in Figure II-1. Several locations of interest to this study are underlined in Figure 11-2. The distance in river miles above the mouth of the river is also shown in Figure II-2.

The Hudson River Basin is bounded on the north by the St. Lawrence and the Lake Champlain Drainage basins; on the east by the Connecticut River, the Housatonic River Basins and the Connecticut Coastal Area; on the west by the Delaware, Susquehanna, Oswego and Black River Basins; and on the south by the basins of small streams tributary to the Hudson River in New York Harbor. The Hudson River watershed extends 128 miles east-to-west and 238 miles north-to-south.

The Hudson River has its source in Henderson Lake in the Adirondack

Mountains in northern New York State and flows generally south for 315 river miles to its mouth at the Battery where it discharges into New

* Some of the material covered in ther first two sections of this chapter is based on several Federal and State publications on the Hudson River 8-14

II-1

II-2 FIGURE II-2
II-3 -7-

York Upper Bay. Upstream of Henderson lake, the stream is known as the Opalescent River and its headwaters are in Lake Tear-of-the-Clouds on the Southwest slop of Mount Marcy in Essex County.

The major tributaries entering the main stream are mohawk River, Hoosic River, Kinderhook Creek, Indian River, Sacandaga River, Esopus Creek and Rondout Creek. The drainage areas of these and other principle tributaries are listed in Table II-1. For convenience, the entire Hudson River Basin has been separated into three principal drainage areas; the Upper Hudson, the Mohawk River, and the Lower Hudson subbasins.

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The division between the Upper and Lower Hudson Basin is at the confluence of the Mohawk River with the Hudson at Green Island. The Federal Dam at Troy, some 154 river miles above the Battery, is the head of tidewater.

The Upper Hudson River flows generally south-southeast to the confluence with the Sacandaga River where it turns to the east. At Hudson Falls it turns again to the South Its total length to Green Island is about 150 miles. The river drains an area of some 4627 square miles. From its source to Troy Dam, the Hudson River drops 1810 feet, resulting in an average bottom slope of about 12 feet per mile.

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TABLE II-1

Drainage Areas Of

Hudson River Basin

Stream	Drainage Area (square miles)
Upper Hudson River	
Cedar River	164
Indian River	201
Boreas River	92
Schroon River	568
Sacandaga River	1,058
Batten Kill	441
Kayaderosseras – Fish Creek	252
Hoosic River	713
Minor streams and direct drainage	1,138
Sub-total	4,627
Mohawk River	
Oriskany Creek	146
West Canada Creek	562
East Canada Creek	291
Schoharie Creek	926
Minor streams and direct drainage	1,537
Sub-total	3,462
(Hudson River at Green Island)	8,090)
Lower Hudson River	
Kinderhook Creek, including Stockport Creek	512
Catskill Creek	417
Roeliff-Jansen Kill	208
Esopus Creek	425
Rondout Creek, including Wallkill River	1,197
wappinger Creek	208
Minor streams and direct drainage	1,594
Sub-total	4,561
Total of Hudson River Basin at the Battery	12,650

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The Mohawk River has its source in the hills near the boundary between Lewis and Oneida Counties, New York. It flows in a southerly direction to Rome, thence it follows a general east-southeast course to its junction with the Hudson River at Cohoes, New York. The total length of the river is about 155 miles and it drains some 3462 square miles. The Mohawk River falls irregularly from its source at elevation 1800 feet above mean sea level to elevation 14.3 feet where it joins the Hudson River at Cohoes.

The Lower Hudson River commences at the junction of the Mohawk and Upper Hudson Rivers at Troy and discharges into the Upper New York Bay. All of this section of the river is tidal. Because of its special nature, detailed description of the estuary portion of the Hudson River is presented in another section of this chapter.

The total length of the Lower Hudson River is about 154 miles asnd it drains an area of some 4561 square miles.

The average slope in the Lower Hudson, as represented by the half-tide level, is about 2 feet in 150 miles. The slope is greatest in the section of the river from troy to Catskill and least between Catskill and Tarrytown.

II-6 -9-

The major physiographic features of the Hudson River Basin are a mountainous terrain covering 48 per cent of the basis; cultivated lands covering 42 per cent of the basin; lakes and water bodies covering 2 per cent of the basin; and urban developments covering 8 per cent of the basin.

B. Basin Hydrology

The general climate of the Hudson River Basin may be considered as moist continental. The Upper Hudson Basin has comparatively long, cold and snowy winters sand short mild summers. In the Lower Hudson Basin, the climate is much milder due to the modifying influence exerted by the valley. For these areas there are usually longer summer periods and milder winters. The Mohawk River Basin has variable weather conditions with characteristics of both areas.

The average annual temperature within the basin ranges from 50°F in the southern portion to 40°F in the Adirondack Mountains. The corresponding average July temperature varies form 75°F to 65°F and the average January temperature is 30°F and 15°F respectively The maximum and minimum temperatures recorded in the basin are 106°F and -42°F respectively.

II-7 -10-

The average annual precipitation varies from 34 inches in the center of the basin to more than 50 inches in the Adirondacks. Most of the basin receives an average of about 40 inches. In general, the precipitation is distributed evenly through the year with a slight rise during the summer. Figure II-3 depicts the normal annual precipitation for the entire Hudson River Basin.

The average annual snowfall for the basin ranges from about 30 inches in New York City to over 130 inches in the Adirondack mountains.

A summary of major basin characteristics including rainfall and runoff data for the three subbasins i.e. the Upper Hudson, Mohawk and Lower Hudson Rivers, is given in Table II-2.

The U.S. Geological Survey (U.S.G.S.) maintains stream gages at some 62 locations in the basin; 21 in the Upper Hudson, 11 in the Mohawk and 30 in the Lower Hudson Basin. Because of tidal oscillation, it is not possible to measure the fresh water flow in the Lower Hudson River directly. Flow histograms in the tidal portion of the river are usually constructed from a knowledge of the Green Island time-discharge relation (the most downstream gaging station above tidewater). The ration of the drainage areas tributary to Green Island and the Battery is 1.65. However, analysis of data developed by the U.S.G.S. indicates that a yield factor of 1.225

II-8
II-9

represents the statistically probably value of the ration of the Lower Hudson discharge to Green Island discharge.

A summary of runoff data at several stations representative of the three subbasins, i.e. the Upper Hudson, Mohawk and Lower Hudson Rivers, is given in Table II-3.

A comparison of long term Lower Hudson monthly average flows with the 1964 histogram is shown on Figure II-4. These histograms were prepared from the Upper Hudson flow measurements at the Mechanicville gage and the Mohawk River measurements at the Cohoes gage for the period from 1918 to 1947 and the Green Island gage for later years. The Lower Hudson River values referred to in this section were established using the yield factor of 1.225.

Flood records indicate that in the past several major floods in the Hudson River Basin have occurred during the spring as well as during the fall or early winter. General storms covering the entire watershed are usually of the transcontinental (cyclonic) or tropical types.

The greatest flood of record over most of the basin occurred in March, 1913 as the result of a period of rapid thaw followed by five days of heavy rainfall. The March, 1936 storm, which followed a sudden rise

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in temperature after a winter of heavy snowfall was the second greatest

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TABLE II-2

HUDSON RIVER BASIN CHARACTERISTICS

Subbasin	Drainage Area (sq.mi.)	Channel Length (miles)	Mean Annual Temp °F	Mean Annual precipitation (inches)	Mean Annual Evaporation & Trans Precipitation	Mean Annual Runoff cfs	Mean Annual Runoff cfs sq.mi.	Mean Annual Runoff in	Usable Storage Capacity acre – ft.
Upper Hudson River Basin	4627	150	40	40	16	7400	1.60	24	863,000
Mohawk River Basin	3462	155	45	46	24	5600	1.61	22	203,900
Lower Hudson River Basin	4561	154	48	42	20	8700*	1.90	22	551,000

* Estimated

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TABLE II-3

STREAMGAGE SUMMARY AT REPRESENTATIVE
STATIONS IN THE HUDSON RIVER BASIN

Station	Area (sq.mi.)	Mean Flow (cfs)	Mean (csm)	Runoff (inches)	Years of Record
Mohawk River Basin					
Mohawk River					
near Rome	150	379	2.53	34.4	47
near Little Falls	1,348	2,708	2.01	27.1	41
at Cohoes	3,453	5,537	1.61	21.8	43
Upper Hudson River Basin					
Hudson River					
near Newcomb	192	387	2.02	27.4	43
at Gooley	419	830	1.98	27.0	52
at North Creek	792	1,534	1.94	26.3	61
at Hadley	1,664	2,849	1.71	23.3	47
at Mechanicville	4,500	7,431	1.65	22.42	70
Sacandaga River					
near Hadley	1,055	2,106	2.00	27.2	61

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Hoosic River					
near Eagle Bridge	510	903	1.77	24.1	56
Schroon River					
at Riverbank	527	795	1.51	20.6	61
Lower Hudson River Basin					
Hudson River At Green Island	8,090	13,060	1.62	22.0	22
Kinderhook Creek at Rossman	329	454	1.38	18.8	44

II-12

TABLE II-3 (Continued)

STREAMGAGE SUMMARY AT REPRESENTATIVE
STATIONS IN THE HUDSON RIVER BASIN

Station	Area (sq.mi.)	Mean Flow (cfs)	Mean (csm)	Runoff (inches)	Years of Record
Catskill Creek at Oak Hill	98	126	1.29	17.5	58
Walkill River at Gardiner	711	1,045	1.47	20.0	44

II-13
FIGURE II-4

-12-

Several tropical type storms occurred in October, 1869; November, 1927; and September, 1938.

Table II-4 summarizes the peak discharges and stages of the Hudson River at Troy and Albany during the eight largest floods since 1846.

Several peak discharge frequency curves based upon a number of statistical evaluation methods are shown in Figure II-5. Further discussion of these curves is presented in the next chapter.

Most of the major floods occurred prior to construction of the Sacandaga Reservoir in 1930 which greatly moderated downstream flows. This reservoir and several other reservoirs control about 20 per cent of the entire Hudson River watershed, they have been built to control floods, augment the natural river flows, and for municipal and industrial supply. The major seven reservoirs in the Basin are listed in Table II-5. Detailed description of these reservoirs appear in Chapter IV of this report.

C. Hudson River Channel Characteristics

The major Hudson River Channel characteristics used in flood routing of the probable maximum flood and backwater computations are summarized below.

TABLE II-4

Flood Stages and Discharges, Hudson River

(Hudson River Basin)

Flood	Troy, N. Y. Drainage Area 8,090 sq.mi.		Albany, N. Y. Drainage Area 8,270 sq.mi.	
	Peak Stage (ft.m.s.1.)	Peak Discharge (c.f.s.)	Peak Stage (ft.m.s.1.)	Peak Discharge (c.f.s.)
October 5, 1869 1/ 2/			18.98	
April 10, 1895 1/ 2/			16.42	
March 28, 1913 1/ 2/	29.4	223,000	21.45	228,000
April 12, 1922 1/	25.7	154,000	15.98	158,000
November 4, 1927 1/	24.8	150,000	15.96	153,000

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March 19, 1936	19.17	215,000	17.5	220,000
September 22, 1938	27.1	183,000	16.5	187,000
December 31, 1948	26.74	181,000	17.46 3/	185,000 3/
Bank full stage	24.0		11.0	

1/ Before completion of Sacandaga Reservoir and 27 ft. navigation channel to Albany
2/ Before construction of Federal lock and dam at Troy.
3/ January 1, 1949.

II-16
Figure II-5

II-17

TABLE II-5

MAJOR HUDSON RIVER BASIN RESERVOIR SUMMARY

No.	Reservoir	Year Placed in Distance Area	Stream Area	Usable Capacity Service	Drainage from Indian Point	Purposes (ac-ft)	(miles)
	MOHAWK RIVER BASIN						
1	Delta Reservoir	1912	Mohawk River	64,500	140	Canal Regulation	225
2	Hinckley Reservoir	1914	West Canada Creek	78,000	374	Canal Regulation and Utica Water Supply	217
3	Schoharie Reservoir	1926	Schoharie Creek	61,400	314	New York City Water Supply	190
	UPPER HUDSON RIVER BASIN						
4	Indian Lake	1898	Indian River	103,000	131	River Regulations, Power	250
5	Sacandaga Reservoir	1931	Sacandaga River	760,000	1,044	River Regulation, Flood Control, Power	180
	LOWER HUDSON RIVER BASIN						
	Ashokan Reservoir	1915	Esophs Creek	383,000	257	New York City Water Supply	75

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Rondout Reservoir	Rondout Creek	158,000	95	New York City Water Supply	75

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The Lower Hudson River Channel is a relatively deep and straight channel. The aviation in the cross sectional area, at mean low water, between the ocean entrance and the head at the Federal Dam at Troy is shown in Figure II-6. The variation is significant and erratic and cannot be accurately described by a simple mathematical model. The total area in the Lower Hudson ranges between over 250,000 sq. ft. in Haverstraw Bay and less than 50,000 sq. ft. at Troy with an average of about 120,000 sq. ft.

The top width at mean low water is shown in Figure II-7. The variation in the surface width is even more erratic than the cross-sectional area. This is due to the presence of several bays and shoals in the Lower Hudson. Two major bays are located on Figure II-6. These are Haverstraw Bay and Newburgh Bay.

The mean depth of the river is shown in Figure II-8. The mean depth is defined as the cross-sectional area divided by the top width. From the Battery to the head of Haverstraw Bay, the mean depth generally decreases from some 38 feet to about 16 feet. Upstream of Haverstraw Bay the depth abruptly increases, in an erratic manner, reaching a maximum of approximately 100 feet in the vicinity of West Point. Point values approach some 200 feet. This abrupt increase in depth has a significant effect on salt water intrusion in the river.

II-19 FIGURE II-6
II-20 FIGURE II-7
II-21 FIGURE II-8
II-22 -14-

As indicated before, the Lower Hudson River is a tidal estuary between the Federal Dam at Troy and the New York Harbor. It is also classified as a dampened, reflected tidal wave regimen.[15] Dampening occurs by dissipation of tidal energy via channel friction as the oceanic tidal wave progresses upstream. Reflection includes secondary wave propagations caused by channel obstructions. Complete reflection occurs at the Federal Dam at Troy. Additional wave reflections occur due to significant changes in channel width. As widths increase, wave amplitudes tend to decrease, whereas a decrease in channel width causes an increase in wave amplitudes. Tidal behavior at any section is the composite effect of ocean tide, channel friction and wave reflection. The primary ocean tides are also variable with maximum amplitudes occurring during spring tide and minimum amplitude during neap tide. Variations in fresh water discharge and the barometric conditions also contribute to changes in amplitude.

Figure II-9 illustrates the principal tidal characteristics along The stretch of river between the ocean entrance and Poughkeepsie.

High and low water above mean low water at Sandy Hook, New Jersey are shown on the upper figure. The half-tide level indicates the average slope in the river. The total fall from Troy to the sea is about 2 feet.

II-23 FIGURE II-9

II-24 -15-

The variation in the tidal range along the river is shown in the middle figure. It will be noted that, in moving upstream, the range of tide diminishes from about 4.4 feet at the Battery to a minimum of about 2.6 feet at a point near Storm King (mile point 56). The tidal range then reaches a maximum of 4.1 feet at a point near Catskill and then increases to 4.8 feet at Troy.

The high water and low water lunar intervals, as referred to the transit of the moon over the meridian of Greenwich, are shown on the lower figure.

Figure II-10 shows the variation of mean sectional, tidal velocity along the river. The raw data for the ebb and flood strengths were obtained from the 1929 U.S.G.S. study. Ebb and flood strengths were each averaged across the river cross section. Mean absolute velocity over a tidal cycle ws then obtained by averaging section averaged ebb and flood strengths.

The variation of the mean tidal "flow" along the river is also shown on Figure II-10. It will be noticed that the mean tidal flow decreases from a maximum of 425,000 cfs at the Battery to zero at the Federal Dam at Troy (about mile point 153).

II-25

FIGURE II-10

II-26 -16-

Hudson River salt intrusions at equilibrium, or near equilibrium conditons, for selected flows are shown on Figure II-11 Data are shown for both the Hudson River model and the prototype conditions. Prototype profiles were obtained by chemical and electtrical measurements sand are representative of equilibrium conditions. Model profiles represent steady state equilibrium conditions and were obtained by chemical measurement on the Hudson River model at the Waterways Experiment Station, U.S. Army Corps of Engineers, Vicksbury, Mississippi. Salinities shown are tidal average values over a tidal cycle and a full channel cross-seciton. The Vicksbury model is operated for the mean tide.

Figure II-12 shows the relationship between the length of salt water intrusion at equilibrium in miles and fresh water flow in the lower

Hudson. this length is defined as the point on the intrusion curve at which the salinity is 100 ppm.

These results indicate that the salinity intrusion is influenced by the oceanic tidal action which causes the ocean-derived salt to advance landward and the up river inflow which tends to flush the estuary.

II-27

FIGURE II-11

II-28

FIGURE 11-12

II-29 -17-

III. HUDSON RIVER BASIN PROBABLE MAXIMUM FLOOD AT INDIAN POINT

A. Introduction

The probable maximum flood (PMF) has been defined as a hypothetical flood produced by the most severe, but reasonably possible, rainfall and related runoff, at a particular area. A rainfall producing a probable maximum flood is ofter called “Probalbe Maximum Precipitation” (PMP). The probable maximum flood has also been described as the boundary between possible floods and impossible floods, i.e. a flood having a probability of occurance approaching zero or a return period of infinity.

No detailed investigation had been performed prior to this study to determine the maximum probable flood for the entire Hudson River Basin.

A semi-empirical approach was followed by Q L & M in early 1970 to estimate such a flood at Indian Point. This approach was based upon several Corps of Engineers studies of a number of neighboring river basins as well as a detailed study of the Upper Hudson River Basin. The probable maximum flood for the Upper Hudson River Basin was conducted by Stone & Webster Engineering Corporation in 1969, The shaded area in Figure III-1 represents the area studied by Stone & Webster.

III-1 Figure III-1

III-2 -18-

A more comprehensive study of a formal or detailed nature involving careful analyses and accepted methodology was conducted in 1970 by Q L & M. The details of that study are given in this chapter. However, the procedures followed are delineated below.

1. A pattern and a time distribution of a probable maximum pre-

cipitation were developed for the Hudson River Drainage Basin at Indian Point.
2. The Hudson River Drainage Basin was divided into twenty-eight subbasins and a unit hydrograph was developed for each subbasin. The twenty-eight subbasins are located on Figure II-1.
3. The probable maximum precipitation of Item 1 was applied to the developed unit hydrographs of Item 2 with the appropriate initial and infiltration losses to establish the subbasin flood hydrographs.
4. The subbasin flood hydrographs of Item 3 were combined in their proper time sequence to give a main river hydrograph. The main river hydrograph was then routed downstream. This process started at the uppermost subbasin and continued in the downstream direction until the resultant probable maximum flood hydrograph at Indian Point was obtained.

The results of the sem-empirical study are presented first followed by the delineation of the detailed study. A copy of the Stone & Webster Upper Hudson River study is appended to this report.

III-3 -19-

B. Previous Investigations

As indicated in Chapter I, three flood studies had been performed prior to this investigation to determine the maximum flow for the Hudson River at Indian Point. A brief description of these studies follows.

The first study was of a preliminary nature and was based upon highly conservative estimates. The results of that study which was conducted in 1968 are delineated in Supplement 10, Indian Point unit No. 3, AEC Docket 50-286.

The 1968 preliminary estimates were then modified as a result of a new study in the early part of 1969. This study considered surrounding topography, land use, river geometry and frictional losses. The so-called “rational formula” was used for the runoff computations. The results of the 1969 study are summarized in Table III-1.

Another investigation based upon several Corps of Engineers studies of a number of adjacent river basins and on a detailed study of the Upper Hudson River Basin was conducted in the early part of 1970. A brief discussion of this study is presented below.

Six different methods were used to determine the probable maximum

III-4 -20-

food for the Hudson River at Indian Point. These methods and the corresponding results are listed in Table III-2.

The PMF values at Indian Point were predicted on the basis of a thorough comparison of the Hudson River Basin with the following basins:

1. Susquehanna River at Harrisburg
2. Potomac River at Washington D.C.
3. Delaware River at Montague
4. Hudson River at Stillwater

The following four basins parameters were used for this purpose:

1. Size of the drainage area (A)
2. Available storage capacity (S)
3. Maximum observed peak discharge (Q)
4. Probable maximum flood (PMF)

The values corresponding to the above listed basins are compared with their Hudson River at Indian Point counterparts in Table III-3. Several interesting conclusions may be drawn from this comparison. On the basis of drainage area size, the Hudson River has the highest available storage capacity and the lowest maximum observed peak. These results indicate that the Hudson River is a highly regulated

III-6 -21-

basin. Therefore, prediction of its PMF on the basis of the drainage area size alone may be misleading.

The relationship between the drainage area and computed probable maximum flood for the adjacent basins is shown in Figure III-2. The Upper Hudson River value at Stillwater is also shown and is well below the majority of the points. For comparison purposes, the value corresponding to the present study is also shown.

The fact that the Hudson River Basin is regulated appreciably is clearly seen in Figure III-3 which correlates the maximum observed peak flow with the drainage area size for the basins of interest. The Hudson River maximum observed peak is 60 per cent lower than that of an adjacent river of equivalent size.

A prediction of the Hudson River PMF on the basis of the maximum observed peaks was also attempted. The results are depicted in Figure III-4. The result of the formal PMF study of this report is also shown on this Figure.

A more realistic approach taking into consideration the combined influence of drainage area size (A), available storage capacity (S), and maximum observed peak flow (Q) on the probable maximum flood (PMF) was then investigated. The classical dimensional analysis

III-9

FIGURE III-2 PROBABLE MAXIMUM FLOOD – DRAINAGE AREA RELATIONSHIP

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Figure III-3

II-11 Figure III-4

III-12 -22-

method was employed for this purpose. This tool recognizes the fact that the physical factors influencing a physical phenomenon should be related in a mathematical model which is dimensionally homogeneous. The parameters A, S and Q were treated as the independent variables representing the physical factors and the PMF as the dependent variable defining the physical phenomenon. The results of this approach are shown in Figure III-5. The value corresponding to the present study is also shown.

Another simple estimate equating the probable maximum flood to a statistically evaluated 1:10,000 year peak flood is also shown in Table III-2. A discharge-frequency curve based on a statistical evaluation of the available 50 U.S.G.S. data points (1918-1969) were prepared as shown in Figure II-5 and extended to 10,000 years. The data points corresponding to discharges prior to the completion of the Sacandaga Reservoir in 1930 were not adjusted for discharges from the Sacandaga river which would have been impounded by the Conklingville Dam.

III-13 Figure III-5
III-14 -23-

C. Probable Maximum Precipitation

The Hudson River Flood records at Green Island show that in the past most major floods have occured in the spring months as a result of an early spring storm combined with melting snow. The hydrologic history of the Hudson River at Green island was discussed in Chapter II. The probable maximum precipitation for the spring months, however, is about 50 per cent of the all season probable maximum precipitation[2]. Moreover, the adjacent Connecticut River Drainage Basin study[3] indicated that the maximum storm would be at least equal to the maximum combination of rainfall and snow melt. Therefore, a summer or fall storm producing a runoff at least as great as a spring storm with melting snow was used in this study.

The probable maximum precipitation for this study was developed from the 72-hour precipitation and depth-duration-drainage area (DDA) curves prepared by the U.S. Weather Bureau for the adjacent Susquehanna River basin[1]. A copy of the Susquehanna DDA Curves is reproduced in Figure III-6.

The PMP curves of Figure III-6 were obtained by transposing storms of record into the Susquehanna Basin and adjusting them for maximum moisture at the transposed location.

III-14a Figure III-6
III-15 -24-

The general level of these curves has been found to be consistent with that of recent estimates for the Potomac & Delaware River Basins[1].

On the basis of these curves, a probable maximum precipitation depth of 14 inches associated with a duration of 72 hours was used for the entire 12,650 square miles of the Hudson River Basin above Indian Point.

1. Selection of Pattern & Areal Distribution of the Probable Maximum Precipitation Storm

The storm selected for application to the Hudson River Basin is similar in areal distribution to that of the Susquehanna River at Danville. The Danville pattern was developed by the U.S. Weather Bureau[1] and is reproduced in Figure III-7.

This pattern was considered in this study due to the similarity in shape and drainage area size between the Susquehanna River at Danville (11,220 Sq. Miles) and the Hudson River at Indian Point (12,650 Sq. Miles). The two rain centers of the Danville pattern were shifted and rotated so as to give best fit to the Hudson River Basin. The resulting pattern is shown in Figure III-8. The letters in Figure III-8 represent different values of the pattern storm Isohyets as will be shown later. One of the selected storm centers

III-16 Figure III-7
III-17 Figure III-8

III-18 -25-

was located on the Sacandaga River Basin (Subbasin 8) in the Upper Hudson River Basin and the other one on the Catskill Creek Basin (Subbasin 18.3) in the Lower Hudson River Basin. The direct distance between these two centers is about 80 miles.

Because of the comparatively small differences between the average elevation of the Hudson River Basin and the elevations of the terrain covered by the selected storm isohyets, no adjustment was made for difference in elevation.

2. Computation of the Isohyet Values for the Total Duration of the Probable Maximum Storm

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The purpose of this step is to define rain concentration associated with the 72 hour total storm within the Hudson River Basin. The procedure employed in this step was also used to compute the isohyet values for incremental durations as discussed in subsequent sections of this chapter.

The computational procedure is described below. Due to the presence of two storm centers, the values of the lettered isohyets of Figure III-8 were determined as follows:

a. The two centers were treated as separate storms and the areas within Isohyets A_1, B_1, C_1, A_2, B_2, C_2, were measured.

III-19 -26-

b. The average depth of rainfall corresponding to each one of the areas of Step a was obtained from the appropriate DDA curves of Figure III-6. The 72 hour curve was used for the computation of the isohyet values for the total duration of the probable maximum storm.

c. To insure that the precipitation averaged along the individual isohyets must equal the basin average PMP, an auxilary isohyet C'_1 enveloping the A_1, B_1, C_1, center, extending down to the A_2, B_2, C_2, center, and forming a border line with the northern end of C_2, isohyet was established.

d. The area enclosed by C'_1 isohyet was conveniently selected equal to that of C_2 isohyet. The average depth of rainfall corresponding to C'_1 isohyet is therefore equal to its C_2 counterpart.

e. The average rainfall depth over an area equivilent to $C'_1 + C_2$ was computed using the appropri- DDA curves of Figure III-6.

f. The average rainfall depths of "Step b," were then adjusted by multiplying them by the ratio of the average depth computed in "Step e" to that of "Step d." As mentioned earlier, this was done to insure that the precipitation averaged along the individual

III-20 -27-

isohyets must equal the basin average PMP.

g. The average depths of rainfall over the areas bounded by isohyets D and E which envelop both centers were computed in a manner similar to that outlined in Steps a and b.

h. The individual values for the selected isohyets were determined graphically as shown in Figure III-9 and summarized below:

i. compute the average value of any two adjacent isohyets using the following equation which assumes linear depth-area relationship.

$$\frac{V_n + V_{n+1}}{2} = \frac{P_{n+1} A_{n+1} - P_n A_n}{A_{n+1} - A_n}$$

in which

$V_{n'}$ V_{n+1} = Values of isohyets number n and n+1. These numbers refer to the lettered isohyets shown in Figure III-8.

$A_{n'}$ A_{n+1} = Area bounded by isohyets number n and n +1. (Steps a and g above).

$P_{n'}$ P_{n+1} = Average depth of rainfall over the areas bounded by

III-21 Figure III-9

III-22 -28-

isohyets 1 and 2 (steps f and g above).

ii. Plot the computed average values of the isohyets versus the corresponding average areas, i.e. $\frac{V_n + V_{n+1}}{2}$ vs. $\frac{A_n + A_{n+1}}{2}$

iii. Locate the area enclosed by the individual isohyets on the plot of item ii above and draw straight lines between any two locations passing through the individual points and following the general shape of the curve of step ii above.

iv. Obtain the values of the isohyets corresponding to the straight lines of step iii.

v. Check the values obtained in step iv by computing the average values of any two isohyets

(step iv) and comparing it against the values

computed in step i.

vi. Repeat until the computed values of step iv

are the same as those obtained using equation 1.

This trial and error procedure was found to be

very stable and its convergance was remarkable.

III-23 -29-

The results of the areas (step a), separate storm depths of rainfall (step b), adjusted depths (step f and g), and isohyet values (step h) for the total duration of the probable maximum storm are summarized in Table III-4.

3. Time Distribution of PMP

The purpose of this step is to distribute the volume of the rain associated with the total duration of the probable maximum storm in time. A procedure similar to that outlined in Hydrometeorological Report No. 40 was employed. The procedure consists of the following two steps:

a. Time concentration of PMP: This expresses how much of the total rainfall in 72 hours is concentrated in the maximum (1^{st}) 6-hour increment, the next highest (2^{nd}) 6-hour increment, 3^{rd}, 4^{th}, etc. The total basin rain volume was distributed proportional to the 6-hour incremental values from the PMP depth-duration curve of Figure III-6. Thus the maximum (1^{st}) 6-hour increment is the 6-hour PMP at the area of the basin, the next highest (2^{nd}) 6-hour increment is the difference between 12-hour and 6-hour PMP etc.

This procedure is conservative in that it combines PMP for all durations in one storm event.[1]

III-24

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TABLE III-4

ISOHYET VALUES FOR THE TOTAL DURATION
OF THE PROBABLE MAXIMUM STORM

Isohyet[1] No.	Area Enclosed[2] by Isohyet (sq.mi.)	Two Storm[3] Ave. Rainfall Depth (in.)	Adjusted[4] Ave. Rainfall Depth (in.)	Isohyet[5] Value (in.)
A_1	56	28.6	25.8	22.6
B_1	330	24.2	21.8	19.4
C_1	1,060	21.2	19.1	16.8
C_1	2,130	19.2	17.3	14.2
A_2	140	26.2	23.6	20.8
B_2	710	22.3	20.1	17.6
C_2	2,130	19.2	17.3	14.2
D	11,020	14.7	14.7	12.0
E	17,210	13.3	13.3	9.6

1 See Figure III-8
2 Step a of the procedure
3 Step b and g of the procedure
4 Step f of the procedure
5 Step h of the procedure

III-25 -30-

b. Sequence of increments: The 6-hour incremental values of PMP over the entire Hudson Basin at Indian Point were rearranged in accordance with the following criteria recommended in Hydrometeorological Report No. 40:

i. Group the four highest 6-hour increments of the 72-hour PMP in a 24-hour period, the middle four increments in a 24-hour period, and the lowest four increments in a 24-hour period.

ii. Within each of these 24-hour periods, arrange the four increments in accordance with sequential requirements; that is, the second highest next to the highest, the third highest adjacent to these, and the fourth highest at either end.

iii. Arrange the three 24-hour periods in accordance with the sequential requirements; that is, the second highest 24-hour period next to the highest with the third at either end. Any possible sequence of three 24-hour periods is acceptable with the exception of placing the lowest 24-hour period in the middle.

III-26 -31-

4. Computation of 6-hour increments of PMP

The incremental average depth for the 12 6-hour subdurations were obtained using the DDA curves of Figure III-6 and within-basin depth-area curves. The within-basin curves were derived for several subdurations and sizes of areas in a manner similar to that recommended in Hydrometeorological Report No. 40. The within-basin curves gave results higher than those based upon the entire basin ratio.

The results of the incremental average depth for the first 4 6-hour increments and for the second and third days are summarized in Table III-5. For successive 6-hour values within the second and third days the recommended U.S. Weather Bureau[1] percentages shown in Table III-5 were used.

The incremental values of the isohyets for the 12 subdurations were computed using the procedure outlined in the preceding item. The results are given in Table III-6.

5. Computation of Subbasin Probable Maximum Precipitation & Runoff

The values of the probable maximum precipitation corresponding to the 12 subdurations for each one of the 28 subbasins shown in Figure II-1 were computed by planimetering the area between the isohyets within the subbasin boundaries.

III-27

INSERT TABLE III-5-10 (DOCUMENT TABLE III5-10)
-32-

The results of one of the subbasins (Catskill Creek – Subbasin 18.3) are summarized in Table III-7.

These increments were rearranged in accordance with the recommended sequence. The following rainfall losses, as recommended by Mr. Nunn of the AEC,[17] were then subtracted to obtain rainfall excesses for each subbasin and subduration:

Initial Losses = 1 in.
Infiltration Losses = .05 in./hour

The recommended rainfall losses agreed reasonably well with the October, 1955 storm and resulting stream flows at several U.S.G.S. gaging stations as will be discussed in the next section of this chapter.

The results of the probable maximum precipitation and runoff computations are given in Tables 8, 9, and 10 for the subbasins in the Upper Hudson, Mohawk and Lower Hudson Rivers respectively.

The incremental PMP values and related runoff for the entire Hudson River Basin above Indian Point are depicted in figure III-10. It is interesting to note that the runoff coefficient corresponding to these results in about 70 per cent more than twice the value used in the February, 1969 report.

III-35 Figure III-10
III-36 -33-

D. Unit Hydrograph Analysis

Six-hour unit hydrographs were established for each of the Hudson River subbasins shown in Figure II-1.

Unit hydrographs for 12 subbasins in the Upper Hudson were taken from the Stone & Webster Report[4] on the Upper Hudson. A copy of that report is appended to this report. A discussion of these unit hydrographs is reproduced from reference below: [4]

> "Unit hydrographs were determined from records of nine gaging stations located on the various streams and reservoirs within the basin having drainage areas varying in size from 90 to 1,044 square miles. In general, the unit hydrographs were developed from the hurricane storm of September 1938 and compared with the next largest storm for which data were available. The second storm used varied from area to area. The storm

of September 1938 produced the largest floods without snow melt for which adequate records are available. It was not possible to develop a unit hydrograph for the Indian Lake Reservoir and Batten Kill from the September 1938 storm because of inadequate rainfall data, and other storms were used. A computer program described by D. W. Newton and J. W. Vinyard, was used in developing the unit hydrographs for the gaged areas.

III-37 -34-

Unit hydrographs for approximately 70 percent of the total drainage area were developed from rainfall and runoff records. The unit hydrographs for the Sacandaga Reservoir and Indian Lake inflows and the Hudson River at the USGS gage at Gooley were used without modification. The unit hydrographs developed at the gaging stations on the Schroon River, Batten Kill and Kayaderosseras Creek were transposed to their respective mouths using Snyder's coefficients and the method outlined in Unit Hydrographs – Part I Principles and Determinations."

Synthetic unit hydrographs were derived for the three subbasins in the Mohawk Basin, Subbasins 9, 13, and 14 in the Upper Hudson and all the Lower Hudson Subbasins (excluding Subbasin 18.5). Taylor & Schwarz's method[5] was used for this purpose. This method was developed for basins in the north and middle Atlantic states. The basic data used in the development includes the Hoosic River, one of the major tributary streams of the Hudson River.

No unit hydrograph was derived for subbasin 18.5. This subbasin has a small drainage area and the runoff from the precipitation over this area was assumed to be instantaneous.

III-38 TABLE III-11
III-39 Figure III-11
III-40 TABLE III-12
III-41 Figure III-15
III-45 -35-

The basic equations used in the derivation of the synthetic unit hydrographs are listed in Table III-11.

The construction of the shape of the synthetic unit hydrographs was guided by the empirical Corps of Engineers[6] relationship between the peak discharge rate and the width of unit hydrographs at ordinates exceeding approximately 50 per cent of the maximum. This relationship corresponding to ordinates equal to 50 and 75 per cent of the peak ordinate is reproduced in Figure III-11.

The area under the synthetic unit hydrographs which should be equivalent to one inch of direct runoff provided another control.

The basic subbasin characteristics used in deriving the unit hydrographs are summarized in Table III-12. The computed unit hydrograph peak flows and lag time are also listed in Table III-12.

Three typical unit hydrographs for three subbasins in the Upper Hudson, Mohawk and Lower Hudson Basins are depicted in Figure III-12, III-13, and III-14 respectively.

Figure III-15 compares the unit hydrograph peaks used in this study with those prepared by the Corps of Engineers through September, 1961.

III-46 -36-

The Corps of Engineers results are represented by a mean, upper and lower limit lines which were drawn to envelop some 200 observed peak discharge values of 6-hour unit hydrographs.

The Hudson River Basin synthetic unit hydrograph values are represented by open circles. Notice that the synthetic peaks used in this study are higher than those represented by the Corps of Engineers mean line.

The Upper Hudson unit hydrograph peaks which were developed from the hurricane storm of September, 1938 are also shown in Figure III-15. These points fall within the upper and lower limit lines with the majority of them above the mean line. Only subbasin #5 (Schroon River) value is below the lower limit line. Since the Schroon River Subbasin has a very small slope ($6x10^{-4}$) and large natural storage capacity and since its unit hydrograph was developed from observed data, it was not found necessary to adjust its peak.

A more elaborate verification of the developed Hudson River Basin unit hydrographs based upon observed precipitation and runoff data was conducted. The flood of October 1955[18] was used for this purpose because of availability of adequate rainfall data in the Hudson River Basin.

III-47 -37-

Storms swept the New York and southern New England area in three periods; October 6-8, 13-17, 30-31. The worst floods occurred in mid-October, 1 or 2 days before or after October 16. The areas most affected were western Massachusetts, western Connecticut, southeastern New York, and a separated area in south-central New York and north-central Pennsylvania.

Maximum 4-day rainfall (October 14-17) at Esopus Creek and Schoharie Creek Subbasins was 17.72 and 16.77 inches respectively. October rainfall at individual stations ranged from 3.14 inches at Plattsburgh, New York to 25.27 inches at West Shokan. This last figure is about 60 per cent of the mean annual rainfall in New York.

The Mohawk River discharge observed at Cohoes reached a peak of 100,000 cfs on October 17. The corresponding Hudson River peak at Green Island and Wallkill River peak at Gardiner in the Lower Hudson Basin were 113, 000 cfs on October 17 and 30,800 cfs on October 16 respectively.

The unit hydrographs developed for the above-mentioned basins (all of the Upper Hudson, Mohawk and Wallkill River unit hydrographs) and the rainfall losses used in this study were verified using the October 1955 storm rainfall and runoff data in the following manner:

III-48 -38-

1. Isohyetal maps corresponding to 12 6-hour increments of rainfall commencing at 24:00 on October 13, 1955 were constructed. Hourly precipitation data observed at 25 weather bureau station in the Hudson River Basin were used for this purpose. A sample of these isohyetal maps is depicted in Figure III-16.
2. The average depth of rainfall for the 12 subdurations for each subbasin was obtained.
3. The rainfall losses used previously were then subtracted to obtain the rainfall excesses for each subbasin; i.e. initial losses of 1 inch and infiltration losses of 0.05 inch/hour. The rainfall and runoff results for the Upper Hudson, Mohawk and Wallkill River Basins are listed in Tables III-13, III-14 and III-15 respectively.
4. Using the beginning of the storm (24:00 October 13) as a common time base, the runoffs were applied to the developed 6-hour unit hydrographs to produce successive 6-hour flood hydrographs for each subbasin.
5. The subbasin flood hydrographs were then combined and routed downstream to the gaging stations; i.e. Cohoes, Green Island and Gardiner. The routing procedure employed for this purpose will be presented in the next section of this chapter.
6. The generated floods of "Step 5" above are compared to the observed Hudson, Mohawk and Wallkill River data in Figures III-17, III-18 and III-19 respectively. The observed data were obtained from the U.S.G.S. hourly records. The agreement between the

III-49 Figure III-16
Figure III-17

III-55 -39-

computed Mohawk and Upper Hudson River floods and their measured counterparts is remarkable. The computed values are somewhat higher than the observed data.

The agreement between the observed and computed results for the Wallkill River is reasonable. The generated peak is about 10 per cent highter than its measured counterpart. This and the other minor differences in the Upper Hudson and Mohawk River results may be due to the non-uniform aerial distribution over large subbasins, base flow and losses estimates. The non-uniform aerial distribution over the long and narrow Wallkill River Subbasin (100 miles x 15 miles) was significant and maybe the reason behind the 10 per cent disagreement.

The good agreement between the measured and computed results indicate that the use of the developed unit hydrographs and recommended rainfall losses is justified.

E. Development of the Hudson River Probable Maximum Flood at Indian Point

1 Description of Procedure: The procedure followed for determination of the probable maximum flood at Indian Point was delineated earlier and may be summarized as follows:

a. Flood hydrographs for all of the Hudson River Subbasins were established using the probable maximum

precipitation (PMP) amounts from the selected probable maximum storm (PMS) together with rainfall losses (Section C above) and the derived unit hydrographs (Section D above).

b. With base flows added the flood hydrographs were combined and/or routed downstream to give the inflows into the Hudson River from its tributaries. Inflows to the Sacandaga and Ashokan reservoirs were routed through the reservoirs. The reservoirs' outflows were then combined with the appropriate subbasin flood hydrographs to define the inflows in the main channel.

c. The main river inflows were then combined with the Hudson River bank subbasin flood hydrographs and routed downstream to Indian Point using the beginning of the probable maximum precipitation as the time origin.

The specific details of this procedure, excluding reservoir flood routing, are presented in this section. When possible, the Mohawk River Basin computations will be used in a detailed manner to illustrate the adopted procedure.

The results of the reservoir outflows are used in this section. However, the details of the reservoir flood routing procedure, are discussed in the next chapter in connection with a thorough treatment of dam failure analysis.

2. Mohawk River Flood Development

a. Mohawk River Subbasins

As indicated earlier, the Mohawk River Basin was divided into three subbasins. These are designated as subbasins numbers 15, 16 and 17 in Figure II-1.

Subbasin No. 15 envelops an area of 2170 square miles and includes the Mohawk River drainage area above the mouth of Schoharie Creek.

Subbasin No. 16 includes Schoharie Creek basin which drains an area of 926 square miles.

Subbasin No. 17 includes the remaining 366 square miles of the Mohawk River Basin and extends from the mouth of Schoharie Creek to Cohoes at the junction of the Mohawk and Upper Hudson Rivers.

b. Subbasin Flood Hydrographs

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To illustrate the procedures employed in this study, a detailed description of the basic flood data and results of a typical subbasin is presented below. The Schoharie Creek Subbasin was arbitrarily selected for this purpose.

Figure III-20 summarizes the necessary basic data for Schoharie Creek. An isohyetal map of the subbasin appears in the upper right hand corner. Isohyets D and C^2 of the selected PMS (Figure III-8) are shown in Figure III-20. Several values interpolated between the pattern isohyets are also shown. The values shown in the Figure correspond to the 72-hour duration. The values of the 12 6-hour subdurations are listed in Table III-6.

The subbasin precipitation, determined by planimetering the area between the isohyets within the subduration is depicted in the upper left hand hyetograph. The rainfall losses (1” initial and .05”/hr. infiltration) as well as the rainfall excesses (shaded area) are also shown. The rainfall excesses were obtained by subtracting the rainfall losses from the incremental rainfall

III-61 FIGURE III-20
III-62 -43-

values. This procedure is outlined in Tables III-9 through III-12.

The synthetic 6-hour unit hydrograph for this sub-basin is represented by the broken line in the lower left hand side of the figure. The method used to derive this as well as the other synthetic unit hydrographs is outlined in Table III-11. The results are summarized in Table III-12. The Schoharie basin characteristics required for the derivation of the unit hydrograph are listed in Figure III-20.

The Schoharie Creek flood hydrograph is also given in the figure. This hydrograph was obtained by applying the rainfall excesses to the 6-hour unit hydrograph.

c. Flood Hydrograph Combination and River Channel Flood Routing

A base flow of 1 cfs/square mile was added to the product of the rainfall excesses and unit hydrograph values to obtain the subbasin outflow. This base flow value which represents the stream flow main-

III-63 -44-

tained by ground water return and subsurface storm flow represents a long term average fore the Hudson River Basin. For the Upper Hudson River Subbasins, base flows obtained from observed flood hydrographs were used. These base flows are listed in Table 4 of the Stone & Webster Report[4].

The generated flood curve in Figure III-20 includes the appropriate base flow value for the Schoharie Creek Subbasin, i.e. 926 cfs.

The generated flood hydrographs for the three Mohawk River Subbasins were combined and routed downstream to the Hudson River. The procedure used for this purpose is depicted in Figure III-21.

The individual flood hydrographs for the three sub-basins are represented by the solid curves in Figure III-21. The Mohawk River Flood downstream of its conjunction with the Schoharie Creek was obtained by combining the flood hydrographs of subbasins 15 and 16. The resultant is represented by (15 + 16) curve in Figure III-21.

The resultant flood hydrograph was then routed through the Mohawk River to Cohoes. The output is represented by the broken curve in Figure III-21. The classical Muskingum coefficient method for routing[7] was used for this purpose. This method will be described later.

The routed hydrograph was then combined with the flood hydrograph of subbasin #7 to obtain the total Mohawk River flood hydrograph at Cohoes. The resultant hydrograph is shown in Figure III-21

The foregoing procedure and the one outlined in the previous section of this chapter were programmed for solution of RAPIDATA time-sharing facilities. A listing of the Mohawk River Program is given in Plate III-1.

Plate III-2 is a listing of the input or data file for the Mohawk River Program which includes rainfall excess (Table III-9), unit hydrograph ordinates (Figure III-21) above flows for the three subbasins.

The solution printouts including the flood hydro-

graph for each subbasin and the results used to construct Figure III-20 are shown in Plate III-3. A more detailed description of the unit hydrographs for all subbasins is also included in Plate III-3.

A description of the flood routing procedure follows.

The Muskingum Method expresses the outflow as the summation of products of routing constants and inflows. The relationship is expressed in equation III-2

$$o_2 - o_1 = \left[\frac{2\ \Delta t}{2k\ (1-x) + \Delta t}\right] \left[I_1 - o_1\right] + \left[\frac{\Delta t - 2kx}{2k\ (1-x) + \Delta t}\right] \left[I_2 - I_1\right] \qquad \text{III-2}$$

in which
o_1, o_2 = Outflow from reach 1 and 2
I_1, I_2 = Inflow at upstream end of reach 1 and 2
Δt = Length of the routing period having a maximum value of 2kx and may be taken as equal to k. The routed hydrograph is relatively insensitive to the value of Δt.
k = Time of travel of flood wave and also

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the change of storage per unit change of discharge.
x = A dimension less constant representing an index of the wedge storage in a routing reach.

The routing coefficients x and k used in this study are summarized in Table III-16. The Upper Hudson River coefficients were determined by Stone & Webster[4] from available flood records.

The October, 1955 storm flood records described in the previous section of this chapter were analyzed to determine the Mohawk River routing coefficients.

The Lower Hudson River routing coefficients were evaluated using a step backwater calculation as will be shown in the next section of this chapter.

Except for two subreaches in the Upper Hudson a value of 0.3 was used for the dimensionless constant x. This value is conservative and is representative of wide rectangular channels.[7] The influence of variation in x on the Hudson

III-81 -48-

River routed hydrographs is discussed in the last section of this chapter.

3. Upper Hudson River Flood Development

A procedure similar to the one outlined in the preceding section was used to derive the Upper Hudson River flood hydrograph at Green Island.

Because of the presence of the Sacandaga Reservoir which controls nearly 30 per cent of the total Upper Hudson basin area, the basin was divided into two segments. The first portion extends from the source to the mouth of Sacandaga River and the second includes the remaining portion of the basin, i.e. from Sacandaga River to Mechanicville.

A computer program, data file, and printouts similar to their Mohawk River counterparts were developed for both segments and are presented in Plates III-4 through III-9. The results are presented graphically in Figure III-22.

The upper portion of this basin includes subbasins 1 through 8. the outflow from subbasin 8 (Sacandaga River Basin) was obtained by routing the flood hydrograph through Sacandaga Reservoir using

III-83 -49-

a detailed reservoir flood routing procedure. The procedure is described in the next chapter.

The flood hydrographs of subbasins 1 through 7 were combined and/or routed downstream to Sacandaga River. The Sacandaga Reservoir outflow was then combined with the resultant of the upstream subbasins. This new flood hydrograph was combined and/or routed with the hydrographs of subbasin 9 through 14 to obtain the resultant Upper Hudson River flood hydrograph at Mechanicville.

The Upper Hudson unit hydrographs at Mechanicville, upstream and downstream of Sacandaga River are shown in Figure III-22. The outflow from Sacandaga Reservoir is also depicted.

The routing coefficients used were obtained from the Stone & Webster Report. A description of the method used to determine these coefficients is reproduced from reference #4 below:

> "The interrelated constants X and K, used to determine the coefficients for routing by the coefficient me method, were evaluated on the section of the river above Thurman Station, because of lack of adequate flood data, evaluation of these constants was made from the stage-discharge-volume relationship for each reach. This

III-111 -50-

> procedure required determination first of K, which is the travel time for an elemental discharge wave to traverse the reach. Then an x, which is a measure of the wedge storage in the reach, was assumed and the flood was routed.
>
> Finally, to check the correctness of the assumed X, the actual wedge storage was calculated using a step backwater calculation with varying flow within the reach corresponding to the inflow and outflow hydrographs. Trials were continued until the assumed and

calculated X's were substantially in agreement."

The Mechanicville flood hydrograph was then combined with the Cohoes hydrograph to obtain the flood hydrograph at Green Island. The Mechanicville hydrograph was not routed to Cohoes since Subbasin 14 of the Upper Hudson Basin extends all the way down to the junction of the Mohawk and Hudson Rivers near Troy. The results are shown in Figure III-23.

4. Lower Hudson River Flood Development

A procedure similar to the Mohawk and Upper Hudson River procedures was employed to route the river hydrographs to the Indian Point site. The Lower Hudson River computer program, input or data file and

III-112 -51-

printouts are listed in Plates III-10 through III-12.
The input or data file includes the resultant flood hydrograph at Green Island, Ashokan Reservoir outflow, rainfall excess and unit hydrograph data for the Lower Hudson Subbasins.

The specific details of the Ashokan Reservoir flood routing is presented in the next chapter.

The coefficients used in the Lower Hudson River Channel routing are described below.

A conservative value of 0.3 representative of a wide rectangular channel was used for the dimensionless coefficient x. A lower x value corresponding to wide parabolic channel results in a small reduction in the probable maximum flood at Indian Point. The Lower Hudson River Channel is somewhere in between these two shapes. To be on the conservative side, however, the value of 0.3 was adopted. The results of this study, as will be shown later, indicate that the probable maximum flood at Indian Point is relatively insensitive to the value of x.

The celerity method suggested by the Corps of Engineers[7, 19] was

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employed to determine the coefficient k. This method is based upon

Seddon's principle, which may be expressed by the following equation:

$$V_w = \left[\frac{1}{B}\right] \left[\frac{dq}{dy}\right] \qquad \text{III-3}$$

in which

v_w = Rate of movement of flood wave

B = Breadth of channel at water surface

q = discharge

y = Height above bed

$\frac{dq}{dy}$ = Slope of discharge rating curve of a station whose cross-section is representative of the reach for steady flow

The value of k is then the ratio of reach length to wave celerity

V_w.

The relationship of Equation III-3 was discovered experimentally by James A. Seddon in studies of flood movements on the Mississippi and Missouri Rivers in 1889. The same relationship was developed mathematically by Kleitz in 1877.[19]

Several Lower Hudson River water surface profiles corresponding to flows ranging from 20,000 cfs to 1,100,000 cfs were developed using an accepted backwater program. The results are shown in

Figure III-24. The backwater program used is described in detail in Chapter V of this report.

The Lower Hudson River between Green Island and Indian Point was then divided into 11 reaches. A discharge rating curve was established for each reach using the surface profile curves of Figure III-24. The discharge rating curves are shown in Figure III-25.

As suggested by the Corps of Engineers,[7] a single value of k corresponding to the average of the initial flow and anticipated peak outflow for each subreach was used. The anticipated peak outflow was determined by trial and error.

The results of this step indicate that the total time of travel of the flood wave between Green island and Indian Point (a distance of more than 150 river miles) is 16 hours. The values for the individual subreaches are listed in Table III-16.

Several other x and k values were used to examine the influence of the routing coefficients on the probable maximum flood at Indian Point. The results are depicted in Figure III-26. It can be clearly seen that the PMF is relatively insensitive to the value of x.

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The dependence of the Indian Point PMF on k is due to the degree of synchronization of the Green Island peak and Esopus Creek peak. In other words, the shorter the time of travel between Green Island and Esopus Creek the smaller the lag time between the two peaks and vice versa.

The use of a weighted average k equivalent to the ratio of the area under k vs. time curve to the total time for each subreach resulted in a longer time of travel, i.e. lower PMF at Indian Point. However, as an added measure of conservatism, the averaging technique suggested in reference number 7 was adopted.

The routed hydrographs at several locations along the Hudson River are shown in Figure III-27.

5. Probable Maximum Flood at Indian Point

The peak discharge thus obtained at the Indian Point site amounts to 1,100,000 cfs and represents the probable maximum flood at Indian Point.

Onn the basis of drainage area size, this value agrees reasonably well with its Susquehana, Delaware and Potomac Rivers counterparts shown in Figure III-2.

III-140 -55-

It is also about 25 per cent and 60 per cent higher than the dimensional analysis value of 880,000 cfs of Figure III-5 and the statistically evaluated 10,000 year flood at Indian Point of Figure II-5.

As a final verification step, the unit hydrographs corresponding to the Green Island flood hydrograph and the Indian Point probable maximum flood were derived and compared to the Corps of Engineers generalized curves of Figure III-15. The 72-hour unit hydrographs were developed by inversing the flood hydrograph procedure, i.e. deducting the base flow from the flood hydrograph and dividing the resultant by the appropriate rainfall excess. The 6-hour unit hydrographs were then developed using the standard S-hydrograph method.[5]

The Green Island and Indian Point unit hydrographs thus obtained are shown in Figures III-28 and III-29 respectively. As shown in Figure III-15 these unit hydrographs produced peak discharges in excess of the Corps of Engineers mean line values. It was then concluded that the Indian Point PMF of 1,100,000 cfs represents a conservative estimate.

IV. FLOOD ROUTING & DAM FAILURE ANALYSIS OF HUDSON RIVER BASIN RESERVOIRS

A. Introduction

The purpose of this chapter is to present the results of the dam failure analysis of the Hudson river Basin Reservoirs and to determine the effect on the maximum possible stage at Indian Point 3 point.

A conservative estimate of the combined failure effect is presented first. The combined effect was obtained by locating the major Upper Hudson and Mohawk Basin reservoirs at Sacandaga and the Lower Hudson Basin reservoirs at Ashokan.

Following this is a detailed evaluation of the probable maximum effect of the largest two reservoirs in the basin (Sacandaga and Ashokan).

Reservoir flood routing procedures used in this study and referred to in the previous chapter are also presented in this chapter.

B. Combined Failure Effect

The purpose of this section is to show that the effect of failure of dams other than Sacandaga and Ashokan is insignificant.

The major reservoirs located upstream of the Indian Point site and their characteristics are summerized in Table II-5. The table values clearly indicate that the largest two reservoirs are the Sacandaga in the Upper Hudson and the Ashokan in the Lower Hudson Basin. The total capacity of both of these reservoirs is more than 250 per cent in excess of all other reservoirs combined. Moreover, most of these reservoirs are located in the Upper Hudson and Mohawk River basins and are all farther from Indian Point than is Sacandaga or Ashokan.

to obtain a conservative, but realistic, estimate of the failure effect of these dams, the Delta, Hinckley, Schoharie and Indian Lake were located at Sacandaga and Rondout at Ashokan.

A formula developed by Lishtwan[20] and a wave profile expression derived by Chow[21] were then used to compute the flow resulting from dam failure of the hypothetical Sacandaga and Ashokan Reservoirs, i.e. including the relocated capacity of the other reservoirs.

Chow's[21] equation of the wave profile, resulting from the failure of a dam is in the form of:

$$X = 2t\sqrt{gh} - 3t\sqrt{gy} \qquad \text{(IV-1)}$$

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in which
X = The distance measured from the dam site
IV-2
-58-
h = The depth of the impounding water

y = The depth of the wave profile

t = The time after dam failure

Equation IV-1 may also be written as follows:

$$y = \frac{1}{g}\left[\frac{2\sqrt{gh}}{3} - \frac{x}{t}\right]^2 \qquad \text{(IV-2)}$$

Integrating this equation from the origin at the dam site to the

wave front $2t\sqrt{gh}$ and solving for the total volume of water $\int yBdx$,

$$\text{Volume} = \frac{8\sqrt{g}}{27} Bh^{3/2} t \qquad \text{(IV-3)}$$

where

B = channel width

the initial discharge after dam failure Qo may be computed by dividing

the volume of equation IV-3 by a very conservative failure time of

one second. Therefore,

$$Qo = \frac{8\sqrt{g}}{27} Bh^{3/2t} \qquad \text{(IV-4)}$$

Lishtwan's [20] equation of the flow downstream of the dam site may

be expressed as follows:

$$Q' = \frac{Wo\ Qo}{Wo + QoLa} \qquad \text{(IV-5)}$$

where

Q' = The flow resulting from dam failure at a distance L downstream

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of the dam site

Qo = The initial dam failure discharge.

W_o = The volume of the stored water.

$$a = \frac{1}{2}\left(\frac{1}{v_2} - \frac{1}{v_1}\right)$$ = 0.224 for basins having a drainage area of up to about 20,000 square miles and river slopes ranging from .1 x 10^{-4} to 5 x 10^{-4}

v_1 = The wave front velocity.

v_2 = The wave tail velocity.

Equations IV-4 and IV-5 were used to determine the flow at Indian Point 3 Point resulting from failure of the hypothetical reservoirs at Sacandaga and Ashokan. The computations and results are summarized in plate IV-1.

Comparison of these results with those resulting from the failure of Sacandaga and Ashokan Dams alone indicated that the effect of the smaller dams represents a small percentage of the total effect. this, coupled with the fact that these dams are farther from Indian Point 3 Point then the two major reservoirs, concentrated the detailed analy- ysis on the failure of Sacandaga and Ashokan Dams alone.

Furthermore, no advantage was taken of the storage available in the the smaller reservoirs, (Delta, Hinckley, Gilboa, Indian Lake and

IV-4
IV-5

IV-6 -60-

Rondout) from the standpoint of PMP flood routing. In other words, the runoff resulting from the probable maximum precipitation was routed through Sacandaga and Ashokan Reservoirs only. The flood storage capacity of these two reservoirs is 12 and 2.3 billion cubic feet respectively.

C, Sacandaga Reservoir Studies

1. Description of Reservoir and Dam

As indicated earlier, the Sacandaga reservoir is the largest regulating feature in the entire Hudson River Basin. It controls all of the Sacandaga River drainage area of 1044 square miles – Subbasin No.8 in Figure II-1. The reservoir is of a multiple use – stream regulation and flood control – type and was formed by construction of the Conklingville Dam in 1930. The Conklingville Dam is lcoated on the Sacandaga River which discharges into the Upper Hudson River some 70 miles upsteam of Troy. The Sacandaga Reservoir and Conklingville Dam are located in Figure IV-1.

At elevation 771.0 feet above mean sea level (Crest of Conklingville Dam Spillway), the reservoir has a surface area of 27,000 acres and a total capacity of 37.8 billion cubic feet. The total capacity consists of the following three componenets:

Use	Elevation	Storage Capacity Billion Cubic Feet
Dead Storage	700 – 735	4.60
Stream Regulation Storage	736 – 768	29.67
Flood Control Storage	768 – 771	3.45

Figure IV-2 shows the capacity curve of Sacandaga Reservoir.

Other interesting characteristics are summarized below:

Length = 27 miles

Width = 2,000 to 28,000 feet

Length of Shore Line = 125 miles

Flow Line Elevation = 771.0' above sea level

Storage Above Elevation 768.0 = 12 billion cubic feet (total

flood storage)

A typical cross-section of the Conklingville Dam is depicted in Figure IV-3.

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The dam was built by the state of New York and is maintained and operated by the Hudson River-Black River Regulating District. It is an earth dam with a relatively impermeable core founded on rock and earth, with a concrete gravity spillway built on rock. The earth dam was built by the semi-hydraulic fill method. The
IV-9

IV-10

IV-11 -62-

earth fill was obtained from borrow pits located at the south abutment and consisted of glacial drift that proved to be ideal for the purpose.

Core samples, taken after completion of the dam construction, showed that the entire dam material is very well packed and the non-core material is an impermeable as the core material itself.[16] The soil characteristics of the core material are listed in table IV-1A and documented in Figure IV-3A. Subsequent to award of the construction contract, the rock excavation for the spillway channel was increased. The extra rock material was added to the downstream toe of the dam, providing an unusually large rock toe section.

The dam is very stable and has proven its ability to withstand severe earthquakes[4] and floods.[16]

A two-day inspection of the dam and reservoir, conducted after occurrence of the highest recorded earthquake on April 20, 1931 at Lake George, new York, showed no evidence of damage.[4]

The dam is relatively impervious and seepage is reduced by the use of a very broad base. The seepage water collects in the unusually large rock toe and moves to a point where it is safely discharged. the maximum measured seepage and leakage through the dam is about 20 cfs.[16]

IV-12
IV-14 -63-

The upstream slope of the dam is protected against wave action by a sizable cover of riprap. The riprap cover extends to above the crest of the core.
Major dam dimenstions and information are summarized below.

Crest Width = 43 feet

Crest Elevation = 795.0 feet

Base Width = 650 feet

Maximum Height = 96 feet

Length of Dam = 1100 feet

Length of Spillway = 405 feet

Crest of Spillway Elevation = 771.0 feet

Crest of Core Elevation = 780.0 feet

Record High Reservoir Stage Elevtion = 769.72 feet (on May 21, 1969)

The outlet works consist of three 8 foot diameter Dow Valves an two 18 foot by 8 foot siphons. The spillway and diversion canal are located in a rock section away from and to the left of the earth dam. The spillway is ungated free overflow section 405 feet long. In addition, a power house with a pumping capacity of some 5600 cfs at elevation 769.0 feet is also available.[16]

Figure IV-4 shows the combined discharge variation with the stage in the Sacandaga Reservoir. The curve values were based upon re-

IV-15

IV-16 -64-

sults of the January – February, 1931 test and information supplied by the Hudson River – Black River Regulating District. The curve values do not include the power house pumping capacity.

2. Flood Routing through Sacandage Reservoir

The outflow pattern that is produced by the Sacandaga River Basin (Subbasin #8) inflow into the reservoir was determined by an analytical procedure based upon a stepwise analysis of the various hydraulic occurrences: varying inflow, changing water level and varying outflow. The equation used for this purpose may be expressed follows:

$$\left[\left(\frac{I_n + I_{n+1}}{2}\right) - \left(\frac{O_n + O_{n+1}}{2}\right)\right]\left(t_{n+1} - t_n\right) = S_{n+1} - S_n \quad \text{(IV-6)}$$

in which,

I = Rate of inflow into the reservoir

O = Rate of outflow from the reservoir

S = Available storage above spillway level

n, n+1 = Subscripts denoting successive intervals of time of length $t_{n+1} - t_n$

Equation IV-6 was programmed as shown on Plate IV-2 and the input or data file included:

a. Subbasin #8 unit hydrograph – see Table III-11.

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IV-18

IV-19 -65-

b. Subbasin #8 inflow storm hydrograph computed using the procedure of Section E of Chapter III.

c. Reservoir stage – discharge curve – see Figure IV-4. This curve does not take advantage of the available power house pumping capacity of 5600 cfs. This added discharge capacity was not considered to be on the safe side and to account for the possibility of inoperative pumps during probable maximum conditions.

d. Reservoir storage curve – see Figure IV-2. A straight line extrapolation was used for elevations in excess of crest of spillway elevation. This was done in accordance with the data supplied by the HR-BRRD.[16]

e. A time interval of six hours.

It was assumed that the outlet works would start operation when the reservoir level reaches elevation 771.0, the crest of Spillway elevation. It was also assumed that the initial reservoir level, prior to the storm, would be at elevation 768.0 (bottom of flood storage – Figure IV-2). These assumptions are very conservative since the PMP considered in this report is a late summer or fall storm and the reservoir would usually be below the flood storage elevation and the outlet work would probably be started before the

IV-20 -66-

level reaches elevation 771.0. The long term late summer-fall average reservoir elevation ranges between 758.37 in August and 752.71 in November.

The program output includes subbasin #8 flood hydrograph (inflow to the reservoir), reservoir outflow hydrograph, and reservoir water surface elevation.

The program, data file and printout are shown in plates IV-2 through IV-4 respectively.

The results are shown graphically in Figures IV-5 and IV-6. Figure IV-5 depicts the reservoir inflow and outflow hydrographs. The variation in water surface elevation is documented in Figure IV-6. The computed reservoir inflow reached a peak of 262,362 cfs some 42 hours subsequent to the beginning of the probable maximum precipitation. The corresponding maximum reservoir outflow is 69,500 cfs occurring 78 hours after the beginning of

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PMP.

The maximum reservoir stage resulting frm routing subbasin #8 flood hydrograph is elevation 783.94. This elevation is about 11 feet below the crest of the Conklingville Dam and less than 4 feet above

IV-21
IV-22
IV-23
IV-24
IV-25
IV-26 -67-

the crest of the core. This value is almost the same as the maximum possible level, elevation 784.0, determined by Stone & Webster.[4]

These findings were discussed with Mr. Robert Forrest, Chief Engineer of the Board of Hudson River – Black River Regulating District, on April 27, 1970. Mr. Forrest stated that the Conklingville Dam is capable of "taking this load without damage." to support this conclusion, he gave the following reasons:

a. The dam was designed for large surcharge and has a sufficient freeboard (24 feet above the spillway crest).
b. The dam has a substantial cross-section and has proven its ability to withstand sever earthquakes.
c. The excellent material used and the settlement and consolidation during the past 40 years have resulted in an inherently stable dam. Core samples taken after completion of construction showed that the dam material is homogenous and relatively impermeable.
d. The unusually large rock toe section downstream of the dam provided a safe disposal system.
e. Perculation from the top will not occur because of the paved highway which has its crest at elevation 795.6

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These comments and the dam failure analysis of the next section of this chapter indicate that the Conklingville Dam will not fail.

3. Conklingville Dam Failure Analysis

The results of the previous section showed that the Conklingville Dam will not be overtopped. Another dam stability criterion involving the position of the free water surface within the dam was considered. This criterion was evaluated to determine the effect of the computed maximum reservoir elevation on the safety of the dam.

In general, earth dams will fail when seepage water escapes at the downstream face of the embankment in sufficient volume to erode the surface and carry soil grains with it. This phenomenon is usually called “piping.” To avoid this, seepage is usually deflected from the face of a dam by the proper use of an earth core on a rock toe.

As indicated earlier, the Conklingville Dam is provided with an unusually large rock toe as well as an earth core. The position of the free water surface within the dam was estimated for the following two downstream conditions:

Condition 1 No water downstream of the dam

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Condition 2 A more conservative case involving a water depth of 20 feet downstream of the dam.

the assumptions and equations used to determine the location of the seepage curve are summarized in Plate IV-5.

For the purpose of computing the seepage curve, the dam cross-section was simplified by the sketch shown in Plate IV-5 and divided into three parts A, B and C. No advantage was taken of the deflection in curve caused by the presence of the core. In other words, the material was assumed to be homogenous. The foundations were assumed to be practically impervious. These assumptions are justifiable,[16] conservative and realistic

The equations are based upon Darcy’s equation:

$q = kiy$ IV-7

or $q/k = iy$ IV-8

in which

q = seepage per unit width

y = depth of seepage curve

i = hydraulic quadient and is equivalent to the slope of the seepage curve $\frac{dy}{dx}$

k = permeability coefficient

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IV-30

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The product (iy) for parts A, B and C may be determined as follows:

Part A:

The flow lines in this portion may be approximated by circles having a common origin at the location of maximum water elevation (Point A in Plate IV-5) using a head loss of (a), the maximum depth at Point A may be taken as the difference between the total depth (H) and the head loss (a). The hydraulic quadient is equivalent to (a) divided by the arc length π (90 - θ) h/360 where tan θ is the slope of the up-stream face of the dam. Substitution in equation IV-8 yields:

$$q/k = \frac{ah}{\frac{\pi\ (90 - \theta)h}{360}} \cong \frac{115\ (H-h)}{90 - \theta} \qquad \text{IV-9}$$

Part B:

Integration of Darcy's equation and the use of the boundary conditions y(x-o) = h and y(x=s) = h_1 yields

$$q/k = \frac{h^2 - h_1^2}{2s} \qquad \text{IV-10}$$

and

$$x = \frac{k}{q}\left(\frac{h^2 - y^2}{2}\right) \qquad \text{IV-11}$$

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-71-

Part C:

When the distance s_1\ is small by comparison to s, the slope of the flow line at point B may be taken as 1, and Equation IV-8 becomes:

$$q/k = h_1 \qquad \text{IV-12}$$

When a downstream water depth of h_o is considered, equation IV-12 becomes:

$$q/k = h_1 - h_o \qquad \text{IV-13}$$

Since the dam is assumed to be homogeneous, the variable q/k for the three parts is the same under steady state conditions. Equations IV-9, IV-10 and IV-12 can be solved simultaneously to obtain the three unknowns h_1, h and q/k and equation IV-11 gives the profile within the middle part of the dam "B". The parameters used in solving these equations include:

H = 80 feet
θ = 18.43º
h_o = o and 20 feet for conditions 1 and 2 respectively
s = 169 feet and 189 feet for conditions 1 and 2 respectively

The results for both conditions are depicted in Figure IV-7. In both cases, the downstream ordinate of the seepage curve is well below the crest of the rock toe.

This indicates that the seepage water would collect in the rock toe

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and would be safely discharged without carrying soil grains with it. In other words piping would not occur.

As a conclusion, therefore, the safety of the Conklingville Dam under the probable maximum precipitation conditions considered in this study would not fail from earthquake, overtopping or piping.

C. Ashokan Dam Studies

1. Introduction

This section presents the results of the Ashokan Reservoir studies. As in the previous section, the presentation begins with a general description of the reservoir and dam followed by delineation of the reservoir flood routing procedure used in connection with the Indian Point PMF determination. Evaluation of the Ashokan Dam failure is given next.

The work required to achieve the dam failure analysis objective includes:

a. Determination of the effect of routing the Esopus Creek flood hydrograph resulting from the selected probable maximum storm (Figure III-8) through the reservoir on the dam.

b. Since the results of item (a) above showed that the dam will not fail, a more critical condition involving

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the effect of a local probable maximum storm (over Esopus Creek Basin only) on the dam was investigated. This condition resulted in a partial failure of the dam.

The last item in this section documents the results of the combined effect at Indian Point resulting from simultaneous occurrence of the Ashokan Dam failure and a Hudson River Standard Project flood.

2. Description of Ashokan Reservoir and Dam

The Ashokan Reservoir is the second largest regulating feature in the Hudson River Basin. It controls some 257 square miles of the Esopus Creek Basin which drains the southerly slope of the Catskill Mountains. This area is designated as Subbasin No. 19 in Figure II-1.

The reservoir is an artificial lake about 12 miles long, located about six miles west of the city of Kingston, and is part of the New York City Water Supply System. It was formed by construction of the Ashokan Dam during 1907-11. The dam is located on Esopus Creek which discharges into the Hudson River some 60 miles upstream of Indian Point. The layout of the reservoir is shown in Figure IV-8.

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The reservoir has a surface area of 8180 acres and consists of two basins connected by a dividing weir. The west basin controls about 90 per cent of the drainage area and has a storage capacity of 6.3 billion cubic feet at elevation 590.0. The east basin is 3 feet lower than the west basin and has a capacity of 10.8 billion cubic feet at elevation 587.0

The reservoir has a maximum surface width of three miles with an average of about 1 mile. The reservoir depth averages about 50 feet and reaches a maximum of 190 feet.

the reservoir is provided with a 900 foot long spillway having a crest elevation of 587.0. The spillway is located at the eastern end of the east dike. The stored water flows through the east basin and is carried by the Catskill Aqueduct to the New York City Water Supply System. The aqueduct has a capacity of some 600 million gallons per day. Except for the dividing weir, the west basin is not provided with any kind of outlet works.

The capacity curves of the east and west basins are shown in Figure IV-9.

As shown in Figure IV-8 the Ashokan Dam consists of several parts. The length and elevation of each part are summarized in Table IV-1.

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TABLE IV-1
LENGTH AND ELEVATION
OF ASHOKAN DAM PART

Part	Length (ft)	Elevation, MSL	
		Crest of Dam	Crest of Core
Main Dam			
Masonry Part	1000	610	
Earth Part	3650	610	596

West Dike	1700	610	596
Middle Dike	6700	606	593
East Dike	2600	602	593
Spillway	900	587	

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Typical crosss-sections of the main dam (both the masonry and earth parts), west, middle and east dikes are shown in Figures IV-10 and IV-10a.

3. Flood Routing through Ashokan Reservoir

The outflow pattern that is produced by the Esopus Creek Basin (Subbasin No. 19) inflow into the Ashokan Reservoir was determined by an analytical procedure similar to the one used for Sacandaga Reservoir routing. Because the Ashokan Reservoir consists of two basins, Equation IV-6 was adjusted to account for the various occurrences as follows:

$$\left[\left(\frac{I_{wn} + I_{wn+1}}{2} - \frac{O_{wn} + O_{wn+1}}{2}\right) + \left(\frac{I_{en} + I_{en+1}}{2} + \frac{O_{wn} + O_{wn+1}}{2} - \frac{O_{en} + O_{en+1}}{2}\right)\right]$$

$$(t_{n+1} - t_n) = S_{wn+1} - S_{wn} + S_{en+1} - S_{en} \qquad \text{IV-14}$$

in which

e and w = subscripts denoting east basin and west basin respect-
tively.

Equation IV-14 was programmed as shown on Plate IV-6 and the input or data file included:

a. Subbasin No.19 unit hydrograph – see Table III-11

b. Subbasin No. 19 inflow storm hydrograph comupted using

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the procedure of Section E of Chapter III. This flood hydrograph was divided between the east and west basin on the basis of their drainage areas.

c. Reservoir storage curves of Figure IV-9. A straight line extrapolation was used for elevations in excess of elevations 587.0 and 590.0 for the east and west basins

respectively.
d. A time interval of 2 hours.
e. Reservoir state – discharge formulas. No advantage was taken of the 600 million gallons per day capacity of the Catskill Aqueduct.

The outflow from the east basin, O_e, (spillway discharge only) was determined using the following equation:

$$O_e = CLH'^{3/2} \qquad \text{(IV-15)}$$

in which

C = Constant equal to 4 feet $^{1/2}$ /second determined from data supplied by new York City Department of Water Resources.
L = Length of Spillway (900 feet).
H = Water head above the spillway in feet.
The west basin outflow, $O_{w'}$ (over the dividing weir) was computed using the following equation:

$$O_w = C\acute{L}H_w^{3/2}\left[1 - \left(\frac{H_e}{H_w}\right)^{3/2}\right] \qquad \text{(IV-16)}$$

in which
C = Constant equal to 4 feet $^{1/2}$ /second
L = Length of weir in feet
H_w = Water head above the crest in the west basin in feet
H_e = Water head above the crest in the east basin in feet.

The Ashokan Reservoir outflow was then combined with subbasin No. 20 flood hydrograph (directly downstream of the dam) to obtain the total Esopus Creek flood hydrographs.

The program, data file and printout are shown in Plates IV-6 through IV-8 respectively.

Figure VI-11 depicts the reservoir inflow and outflow hydrographs. The computed inflow reached a peak of about 58,000 cfs some 44 hours after the beginning of the storm. The corresponding maximum reservoir outflow is about 26,000 cfs occurring 8 hours later.

The variation in the water surface elevation in the east, as well as in the west basin, is shown in Figure IV-11a. The maximum elevation in the east basin is less than 591. This is more than two feet below the crest of the east dike core and more than 11 feet below

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the crest of the east dike. The maximum elevation in the west basin is about 595 feet above mean sea level. This value is about one foot below the crest of the concrete wall core and 15 feet below the crest of the dike.

4. Ashokan Dam Failure Analysis

a. Under Hudson River Probable Maximum Flood Conditions

The results of the previous section showed that under the Hudson River PMF conditions, the water level in the east and west basin would reach a maximum elevation of 590.74 and 595.14 respectively. These values are below the elevations of the crest of the concrete wall core and well below the dike crest elevations. Therefore, seepage or overtopping of the dam would not occur. this conclusion is supported by the events of the March, 1951 flood. The elevation of the water in the east basin reached a record high of 592.23 feet at 4:45 A.M. on March 31, 1951.[22] This value is about 1.5 feet higher than the above presented value of 590.74. The March, 1951 flood was the highest of records extending back to 1904.

Moreover, structural analysis of the Ashokan Dam indicated that the dam is stable under these conditions. Details of the structural analysis under a more critical condi-

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tion are presented in the subsequent section.

Therefore, there is no reason to believe that the Ashokan Dam would fail, under the Hudson River PMF conditions from overtopping or seepage.

b. Under Esopus Creek Probable Maximum Flood Conditions

As indicated earlier, a more critical condition suggested by the AEC personnel was investigated. It was considered because the lower center of the selected PMS (Figure II-8) covers asn area to the east of the dam site. Another storm pattern would have resulted in a storm center over the Esopus Creek Basin. It was then decided to evaluate a probable maximum storm over Esopus

Creek Basin, study its effect on Ashokan Dam and determine the water surface elevation at Indian Point as a result of this flood and a standard project flood over the rest of the Hudson River Basin.

The Esopus Creek PMF and its effect on the dam are presented in this section The effect on the Indian Point site is considered in the next section of this chapter.

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Much of the discussion on the determination of the probable maximum flood at Indian Point presented in Chapter III applies here, mutatis mutandis.

The depth-duration values of PMP for a basin having a drainage area of 257 square miles were obtained from Figure III-6. The results are shown in Figure IV-12. The total rainfall for such basin is about 25 inches in 72 hours.

The 6-hours incremental values of PMP over this subbasin were then rearranged in accordance with the Hydrometeorological Report No. 40 criteria.

The rainfall excesses were obtained by substracting the rainfall losses of Chapter III from the PMP values. the incremental PMP values and related runoff for subbasin No.19 are depicted in Figure IV-13. The results are also summarized in Table IV-2.

Subbasin No.19 unit hydrograph – see Table III-12 – was then applied to the rainfall excess to obtain the flood hydrograph.

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this hydrograph was routed through the Ashokan Reservoir following the procedure described earlier.

Computer outputs of the reservoir inflow and outflow hydrographs as well as the related water surface elevations in both basins are shown in Plate IV-9.

The reservoir inflow hydrograph is shown in Figure IV-14. The computed inflow reached a peak of about 200,00 cfs some 44 hours after the beginning of the Esopus Creek probable maximum flood. This peak is about four times the peak resulting from the Hudson River probable maximum flood and resulted in a water surface elevation higher than the crest of the masonry wall in the west dike and earth part of the main dam.

The variation in water elevation caused by runoff generated by the Esopus Creek PMP in both basins is shown in Figure IV-15.

It was assumed that failure of the dikes and earth part of the main dam would occur with the reservoir water level at a foot higher than the crest of the

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masonry core wall. In other words, failure of the west dike and earth part of the main dam would occur when the west basin water level reaches elevation 597.0. The corresponding east basin failure elevation is 594.0.

The results of Figure IV-15 indicate that the failure elevation would be reached in the west basin 40 hours after the beginning of the storm. The corresponding maximum water elevation in the east basin is about 590.5 or 2.5 feet below the crest elevation of the east basin dikes.

Structural analysis of the main dam, presented in the next section of this chapter, showed that the masonry part would not fail under the Esopus Creek PMP conditions.

It was concluded, therefore, that only the west dike and the earth part of the main dam would fail 40 hours after the beginning of the Esopus Creek PMP.

The influence of this failure on the reservoir outflow

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and elevation is depicted in Figures IV-14 and IV-15 respectively. The specific details and effect of dam failure on the Indian Point site are presented in the last part of this chapter.

c. Structural analysis of the masonry Part of the Main Dam (Olive Bidge Dam)

The purpose of this section is to show that the masonry part of the main dam shown in Figure IV-10 would not fail under the Esopus Creek PMP conditions. the following two conditions were evaluated:

Condition i – An upstream water level at elevation 610.0 and no water downstream of the dam.

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Condition ii – An upstream water level equivalent to the east basin failure elevation of 597.0 and a water depth of 20 feet downstream of the dam.

Condition i:

The major forces acting on the dam are indicated in Figure IV-16 and defined below:

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F_H = Horizontal component of hydrostatic pressure, acting along a line H/3 feet above the base

= $1/2H^2$, where y = specific weight of water.

F_V = vertical component of hydrostatic pressure

= weight of fluid mass vertically above the upstream face, acting through the centroid of that mass.

W = Weight of dam = (area of cross-section of dam) (Sy) where S = specific gravity of masonry, approximately 2.4 or 2.5, acting through centroid of cross-section

F_u = Uplift force on base of dam, as determined by foundation seepage analysis, and integration of point pressure intensities over base area;

if foundation is homogeneous and permeable, pressure varies approximately linearly from full hydrostatic head at the heel to full tailwater head, and F_u is approxi-

mately 1/2yHB, acting at B/3 from the heel. This value is often multiplied by some fraction less than 1 if the foundation is relatively impermeable, but it is on the safe side to assume uplift over the entire base area.

F_E = Vertical component of earth pressure acting on downstream face, i. e. weight of earth mass vertically above the downstream face, acting through the centroid of that mass.

R = Resultant of foundation shear and bearing pressures: horizontal component, $R_H = F_H + F_E$ acting along the base; vertical component, R_v = $W + F_V – F_u$ acting at a distance x from the toe that can be determined by the requirement for rotational equilbrium of the dam, by equating to zero the sum of the moments of all the foregoing forces about the toe of the dam.

G = Resultant of all forces acting on the dam, equal to R but in the opposite direction.

For further explanation of these formulas, the reader is refered to reference 23. The equations used to determine these forces and the results are listed in Table IV-3. The horizontal and vertical components of the resultant of all forces are 1,510 x 10^3 and 2542 x 10^3 pounds respectively. The location of the resultant is in the middle third of the base about 72.5 feet from the toe.

The factor of safety against sliding and overtuning for this condition are 1.09 and 1.87 respectively. The maximum normal stress and shear stress are less than the allowable limits.

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These results indicate that the masonry Part of the main dam would not fail.

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Condition ii

The equations used to compute the various forces and the results for this condition are listed in Table IV-4.

The horizontal and vertical components of the resultant are 1088 x 10^3 and 2044 x 10^3 pounds respectively. The location of the resultant is in the middle third of the base about 91 feet from the toe.

The factors of safety against sliding and overturning for this condition are 1.22 and 1.86 respectively. These results indicate that condition i is more critical.

Therefore failure of the masonry part of the Olive Bridge Dam from overturning or sliding would not occur.

5. Combined Effect at Indian Point resulting from Simultaneous Occurrence of Ashokan Dam Failure and Hudson River Standard Project Flood

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The following procedure was employed to determine the flood hydro-

Graph at Indian Point resulting from Ashokan Dam failure and Hudson

River standard project flood:

a. Computation of Reservior Outflow Hydrograph after Dam Failure

The work required to achieve this objective includes:

i. The use of Equation IV-4 to determine the initial dam failure discharge. The affected area of the west basin dike and main dam is 238,670 feet with an average depth of 47 feet. This area was conservatively approximated by a rectangular cross-section. The computed initial dam failure flow is about 2.6 million cubic feet per second.

ii. Determination of west basin emptying time. The dam break hydrograph was conservatively assumed to have a triangular shape with the initial flow as the height and emptying time as the base. The emptying time is, therefore, equivalent to $\frac{2v}{Q_o}$ where v is the total volume.

of water stored in the west basin (6.765

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billion cubic feet) and Q_o is the initial flow (2.6 million cfs). These values give an emptying time of 5200 seconds or 1.45 hours.

iii. Combination of dam break hydrograph with the east basin outflow over the main spillway to obtain the reservoir outflow hydrograph. the east basin was computed in a manner similar to that outlined in Section 3 of this chapter.

The reservoir outflow hydrograph including dam break wave is shown in Figure IV-14. Subsequent to the emptying time the resultant hydrograph coincides with the reservoir inflow hydrograph. This is a conservative assumption since no advantage is taken of the "cushioning effect" of the east basin.

b. Routing of Esopus Creek Flood Hydrograph to the Hudson River.

This step requires the following:

i. Combination of Ashokan reservoir outflow Hydrograph (including dam break wave) of

previous step with the flood hydrograph of the remaining part of the Esopus Creek Basin, i.e. Subbasin No. 20. Since the contribution of

Subbasin No. 20 represents only a fraction of the reservoir outflow hydrograph, the resultant is essentailly the same as the reservior outflow hydrograph shown in Figure IV-14.

ii. Computation of Valley storage in Sub-basin No. 20 (between the dam and Hudson River). It was assumed, as suggested by reference 21, that the wave front moves in a steeply inclined wall of water whose profile is unchanging as long as the channel conditions remain fixed and the source of supply is constant. Therefore, this can be considered a special case of uniformly progressive flow, known specifically as the roll wave. The wave front discharge, $Q_{w'}$ may be obtained using Manning's Equation:

$$Q = \frac{1.49}{n} \, BY'^{5/3} \quad S_o^{1/2} \qquad \text{IV-17}$$

In which

S_o = Channel slope

Y' = Depth at the crest of the wave

B = Channel width

n = Manning's coefficient (.035 was used for this purpose)

As a conservative estimate of the wave profile, a constant flow equivalent to the maximum initial failure flow of 2.6 million cfs and the valley characteristics (B and S_o) obtained from the **U.S.G.S.** maps were used.

A careful study of the topography of Esopus Creek Valley indicated that the flood generated by the Esopus Creek PMP and Ashokan Dam failure would create a lake extending from the dam to Glenerie Falls some 5 miles west of the Hudson River. The lake would have a total length of some 14 miles and a surface width ranging from 4000 to 1800 feet. The lake would have a control section at Glenerie Falls where width decreases

suddenly from 6000 feet to 500 feet. A sketch showing the size, direction of flow through the lake and the location of the control section is presented in Figure IV-17.

The maximum water surface level at Glenerie Falls would be at elevation 180.0. This value was computed using Equation IV-17 and the standard step method of computing gradually varied steady flow profiles. Elevation 180, therefore, delineate the shore lines of the lake at this elevation is about 7 billion cubic feet. This value is almost the same as the storage capacity of the Ashokan west basin. This provides an added confirmation of the computed elevation of 180.0.

The storage capacity curve of this lake is shown in Figure IV-18.

The outflow from the lake at Glenerie Falls was computed using the standard critical depth formula:

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$$Q = \sqrt{\frac{g A^3}{B}}$$

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A stage-discharge curve was established for this lake and is shown in Figure IV-18.

ii. Routing of Esopus Creek Flood Hydrograph through the lake was accomplished by using a computer program similar to the one described in Section 3. The program, data file and printouts are given in Plates IV-10 through IV-12. The input includes the initial lake outflow of 620,000 cfs at elevation 180.0, Subbasin No. 19 flood hydrographs generated by Esopus Creek PMP and Subbasin No. 20 flow caused by runoff generated by the Hudson River Standard Project Flood.

The 620,000 cfs represents the effect of the dam failure at Glenerie Falls and is equivalent to the lake outflow at elevation 180.0. This is a conservative, but realistic, estimate of the failure effect since the volume of the stored water in the lake is

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essentially the same as the Ashokan west basin capacity.

c. Routing of Esopus Creek Flood Hydrograph and Hudson River Standard Project Flood to Indian Point

The Esopus Creek flood hydrograph resulting from Ashokan dam failure, Subbasin No. 19 PMP and Subbasin No. 20 SPF as computed above is represented by Curve I in Figure IV-19.

The Hudson River Standard Project Flood upstream of Esopus Creek was obtained by dividing the probable maximum flood values of Figure III-27 by two. The factor two is within the range of 40 per cent to 60 per cent developed by the Corps of Engineers for the Susquehanna River Basin[24] and was suggested by Mr. Nunn of the AEC.[17] Curve II of Figure IV-19 shows the SPF above Esopus Creek.

Curve III in Figure IV-19 represents the combined effect (Curve I + Curve II) just downstream of the mouth of Esopus Creek.

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The combined hydrograph was then routed through the Hudson River Channel to Indian Point using the procedures developed in Chapter III. The resultant at Indian Point is shown as Curve IV in Figure IV-19.

The program, data file and printouts for this condition are listed in Plates IV-13 through IV-15 respectively. The maximum discharge at Indian Point resulting from the conditions considered in this chapter is 705,000 cfs. The probable maximum flood at Indian Point is about 50 per cent higher than this value.

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V. MAXIMUM RIVER ELEVATION FOR FLOODING CONDITIONS AT INDIAN POINT

A. Introduction and Summary of Procedures Employed

The purpose of this chapter is to determine the maximum water surface elevation at the Indian Point site resulting from several flooding conditions including:

1. Runoff generated by the Hudson River probable maximum precipitation presented in Chapter III.
2. Flooding caused by the occurrence of the Ashokan Dam failure concurrent with the standard project flood for the Hudson River Basin considered in Chapter IV.
3. Flooding resulting from the probable maximum hurricane for the New York Harbor area concurrent with spring high tide discussed in Appendix A.

To achieve this objective, backwater curves were calculated using the well known standard step method[21] as well as a newly developed numerical method of computing gradually varied flow profiles,[25] starting with several assumed water-surface elevations at a control point on the river. The hydrodynamic and physical characteristics of the Hudson River dictated the use of the ocean entrance at the battery as the control section. Therefore, the backwater computa-

tions were initiated at this location and the following boundary conditions, at the Battery, were evaluated:

1. Mean sea level
2. High tide water elevation
3. Low tide water elevation
4. Standard project hurricane
5. Probable maximum hurricane

The tidal variation in the Hudson River Estuary discussed in Chapter II influenced the choice of the proper boundary conditions as well as the value of the discharge for which the flow profile is desired. As in the case of flooding from dam failure and unlike the previous attempts, the tidal variation was treated as an integral part of the system and its influence was simultaneously coupled with the other relevent hydraulic occurrences.

In the evaluation of the maximum water surface elevation at Indian Point resulting from the above delineated flooding and boundary conditions, the wind produced local oscillatory short period waves were also considered. The computed Indian Point stages corresponding to the various flooding conditions were conservatively increased by one foot to account for this effect. A detailed discussion of this effect appears in Appendix A.

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A detailed description of the standard step method employed in this study as well as the newly developed flow profile numerical method is presented first. Following this is a discussion of the influence of the Hudson River tidal variation. The results of the various flooding and boundary conditions are discussed and presented last.

B. Methods of Backwater Computation

The so-called "Standard Step Method"[21] was employed in this study to determine the water elevation at Indian Point as well as the time of travel of the flood hydrograph between Troy and Indian Point Point.

However, a new numerical method of computing gradually varied steady flow profiles developed by Prasad[25] was used to verify the Standard Step Method results.

A discussion of both methods follows.

1. Standard Step Method

This method is characterized by dividing the channel into several short reaches and carrying the computation step by step from one end of the reach to the other. The basic equation that defines the procedure may be expressed as

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follows:

$$H_{i+1} = H_i + h_f + h_e \tag{V-1}$$

in which

H = Total head (elevation of the energy line) above a horizontal datum. The mean sea level was selected in this study as the datum.

h_f = Friction loss between two end sections i and i + 1

h_e = Eddy loss. For convenience of computation, this loss was considered part of the friction loss h_f

The friction loss h_f may be computed as follows:

$$h_f = \bar{S}_f \Delta x = \tfrac{1}{2}(S_i + S_{i+1})\Delta x \tag{V-2}$$

where

$\bar{S}_f$ = Friction slope taken as the average of the slopes at the two end stations, i.e. $\bar{S}_f = \tfrac{1}{2}(S_i + S_{i+1})$

Δx = Longitudinal distance between the two end sections.

Manning's formula was used to compute the friction slope

$$S_f = \left(\frac{nv}{1.49}\right)^2 \frac{1}{R_h^{4/3}} \tag{V-3}$$

where

n = Manning's roughness coefficient

v = River velocity and is equal to $\frac{Q}{A}$

R_h = Hydraulic radius

Q = River discharge

A = Cross-sectional area

The total head or elevation of the energy line above mean sea level may be expressed as follows:

$$H_i = Y_i + \alpha \frac{v_i^2}{2g} \qquad \text{(V-4)}$$

in which

Y = Water surface elevation above mean sea level

α = Energy coefficient. This coefficient was assumed to be unity because of the fairly straight alignment and regular cross-section of the river between the Battery and Indian Point.

The river channel was conservatively approximated by a rectangular cross-section defined by the mean water surface width B and the mean depth D. Thus

$$A_i = B_i D_i = B_i (Y_i + Z_i) \qquad \text{(V-5)}$$

in which

Z_i = Water depth below mean sea level

No advantage was taken of the increase in surface width

above mean sea level. Since the channel is of a wide rectangular shape, the hydraulic radius R_h was approximated by the mean depth $Y_i + Z_i$.

The foregoing equations were programmed for solution on RAPIDATA time-sharing facilities. The program is listed in Plate V-1.

The Lower Hudson River Channel was divided into 136 reaches ranging in length from 0.3 to 2 miles. The wide variation in channel geometry influenced the selection of these subreaches. The surface width at mean water B and mean depth below mean sea level Z corresponding to the 136 sections are listed in Plate V-2. The locations of these stations expressed in miles above the ocean entrance at the Battery are also shown in Plate V-2.

An estimated average value 0.025 was used for Manning's roughness coefficient n in the Lower Hudson. Limited information is available for an accurate determination of this coefficient. The 0.025 value was selected on the basis of several U.S.G.S. estimates of the Lower

Hudson River flow resistance coefficient. These estimates were derived from velocity measurements made by the U. S. Geological Survey at the Poughkeepsie gaging site. Values ranging from .019 to .024[26] were computed using avail-able, but inconclusive, data and several computer programs developed by the U.S.G.S.[27-32]

Moreover, the selected n value falls within the range recommended by Dronkers[33] for tidal rivers. (Chezy coefficients of 50 to 70 meter ½/sec for shallow waters and deep inlets respectively. The corresponding Manning n values, for a channel having an average hydraulic radius of 30 feet are 0.029 and .021 respectively.)

The procedure used to compute the backwater profiles may be summarized below:

a. Select a boundary condition, i.e. water surface elevation above MSL at the control section (the Battery).

b. Select a flooding condition in the Lower Hudson River such as flooding resulting from PMP, dam failure etc. Use a constant flow

value Q equivalent to the peak of the flood hydrograph. This is a conservative assumption since the flow, prior and subsequent to the peak time, is substantially less than this value.

c. Compute the control section velocity corresponding to the selected water surface elevation – Step a – and flooding condition

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- Step b – using the continuity equation

$V_i = Q \diagup B_i (Y_i + Z_i)$

d. Compute the friction slop S_f^i using

Equation V-3.

e. Determine the energy line elevation

H_i above MSL using Equation V-4.

f. Estimate the water surface elevation at

the upstream section of the first reach i.e.

section i + 1. As a first approximation use

$Y_{i+1} = Y_i = S_f^{(i)} \Delta x$

g. Using the estimated Y_{i+1}- Step f – repeat

steps c, d and e to determine V_{i+1}, S_f (i + 1)

and H_{i+1}.

h. Compute the difference between the elevation

of the energy line at sections i and i + 1, i.e.

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$\Delta H = H_i +1 = H_i$

i. Compute the friction loss h_f between the

two sections using Equation V-2.

j. If the difference between ΔH and h_f is

greater than .05 feet, assume a new value

of Y_{i+1} and repeat steps g through i until agreement with the control $\Delta H - h_f \leq .05$ feet is obtained.

k. Proceed to the next reach using the final Y_{i+1} value of the first reach as the downstream control for the second reach.

l. Repeat until the water surface elevation at the last section is determined.

2. Numerical Method of Computing Gradually Varied Flow Profiles

This method was developed by Prasad in 1968.[25] Prasad describes his method as "simple, fast, efficient, and does away with most of the approximations and limitations of other methods of computing gradually varied flow profiles."

The method is based on the following two equations:

$$D' = \left[S_o - \frac{n^2 Q^2}{2.22 A^2 R_h^{4/3}} \right] \Big/ \left[1 - \frac{\alpha Q^2 B}{g A^3} \right] \qquad \text{(V-6)}$$

and

$$D_{i+1} = D_i + \left(\frac{D'_{i+1} + D_i}{2} \right) \Delta x \qquad \text{(V-7)}$$

in which

D = River Depth

D' = $\underline{dD}$

dx

S_o = Slope of river bed

Equation V-6 is the well-known differential equation of gradually varied flow expressed in terms of channel geometry and hydraulic characteristics. Equation V-7 utilizes the classical trapezoidal method of integration to express the depth in terms of its derivative with respect to the longitudinal distance x. This equation requires a very small reach length Δx.

These two equations can be solved numerically to determine the two unknowns D and D'. The procedure suggested by Prasad is reproduced below.

a. Compute D'_i from equation V-6 in which D_i is given as an initial condition or from

a previous calculation.

b. Assume $D_{i+1} = D'_i$ as a first approximation

c. Calculate an approximate value of D_{i+1} from equation V-7 using the value of D'_{i+1} obtained from step b or d.

d. Compute a new value of D_{i+1} obtained in step c.

e. If this new value of D'_{i+1} is not very close to the previously assumed value in step b, repeat steps c through e. Otherwise repeat the whole procedure for another i, or, advance the solution by one integration step.

The foregoing procedure was programmed for solution on RAPIDATA time-sharing facilities. A listing of the program appears on Plate V-3. The input data are identical to those used in the Standard step method. However, because of the use of the trapezoidal integration method in this program, the reach size Δx was reduced by 400 per cent.

C. Influence of Hudson River Tidal Variation on Water Surface Level at Indian Point

As indicated earlier, the influence of tidal variation in the Hudson River during probable maximum flood conditions was treated as an integral part of the problem and simultaneously coupled with the other hydraulic occurrences. In the past studies, a somewhat piecemeal method consisting of adding a constant component flow equivalent to the maximum observed ebb flow to the the flood hydrograph at Indian Point was employed. A careful evaluation of this effect during high runoff conditions revealed the following findings:

1. The tides and currents in the river are subject to large seasonal variations due primarily to fluctuations in the fresh-water discharge.

2. Near the mouth of the river this variation is relatively small,

but in advancing up the river the variations become more important and reach their maximum at the head.

3. At the time of high runoff from the Upper Hudson and Mohawk Basins, the tide may be completely masked in the upper portion of the river, the water continuing to rise or fall for a period of several days without any tidal oscillation.

4. The ocean derived salinity intrusion length is influenced, to a great extent, by the upland runoff. The relationship between

the intrusion length and the fresh water flow in the Lower Hudson River is shown in Figure II-12. Extrapolation based upon the equations of Figure II-12 shows that under the PMP conditions the salinity concentration in the Lower Hudson would be less than 250 ppm. This indicates that only a small reach of negligible length would be influenced by the ocean derived salt.

On the basis of these observations, a more realistic approach was adopted. The new approach may be illustrated by the following steps:

1. Assume that the tidal variation at the ocean entrance is not influenced by the probable maximum precipitation. This assumption is supported by the USC and GS tidal measurements over a very long period (1899-1932).[34] A comparison between the tidal characteristics at the mouth (the Battery) and head (Albany) for the year 1922 is shown in Table V-1. This

year was selected for this purpose because of the high runoff that occurred in April (154,000 cfs at Troy).

The table values clearly indicate the influence of land runoff at the Battery is relatively small. The April values are within 1 to 2 per cent of their annual average counterparts.

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In the vicinity of Albany, on the other hand, the April values are 60 to 150 per cent higher than the annual average.

2. For a given flooding condition, such as the probable maximum flood at Indian Point (1,100,000 cfs) and a known boundary condition, such as mean water level at elevation 0.0, compute the Lower Hudson River flow profile. The standard step method may be used for this purpose. Designate time (t) to this profile.

3. At the beginning of the flood cycle, the state at the Battery starts to increase until it reaches a maximum elevation of +2.2 some three hours later. This increase in elevation causes a decrease in the slope of the Lower Hudson River water surface profile. This change results in a stage increase along the river ranging from a maximum of 2.2 feet at the Battery to zero at some upstream location. The amount of water stored within the reach of influence (ΔV) is equivalent to the product of the time interval (Δt) and decrease in river flow caused by channel storage (ΔQ).

The process is then reversed during the second half of the flood cycle, i.e. the water stored during the first half of

of the flood cycle is released causing an increase in the flow and a decrease in water surface elevation within the same influenced reach. About three hours later the flow profile approaches its initial condition (before the beginning of the flood cycle)

4. During the first half of the ebb cycle (from mean water to low water), the stage at the Battery starts to decrease until it reaches a minimum elevation of –2.2 during maximum ebb. This control section stage variation causes an increase in the water surface gradient resulting in a stage decrease along the influenced reach. The water volume released during this interval is again equivalent to the product of the time interval and the difference between the initial and final flows. The process then continues until the flow profile returns to its original shape.

The tidal cycle at the Battery was divided into 16 intervals having a duration of about 45 minutes. The boundary conditions corresponding to these intervals were obtained from the USC and GS tidal measurements at the Battery. A trial and error procedure was then employed to determine the length of influenced reach, amount of

water stored or released during each interval and related channel flow values. The standard step method was then employed using the computed flows and given boundary conditions to obtain the flow profiles in the Lower Hudson. The final results are presented in the next part of this chapter.

D. Water Surface Level at Indian Point

Several severe Hudson River flooding conditions and water-surface elevations at the Battery were presented in the previous chapters. These conditions were conservatively grouped in seven different ways on the basis of a simultaneous occurrence of an appropriate set of flooding, hurricane and tidal conditions.

The seven groups are outlined in the following tabulation:

Case No.	Flooding Condition at Indian Point	Water-surface Elevation at the Ocean Entrance
1	Probable Maximum Flood	Mean Tide, i.e. M.S.L.
2	Probable Maximum Flood concurrent with tidal flow	High Tide
3	Probable Maximum Flood concurrent with tidal flow	Low Tide
4.	Dam failure concurrent with standard project flood	Mean Tide
5.	Standard Project Flood	Standard Project Hurricane

Case No.	Flooding Condition at Indian Point	Water-surface Elevation at the Ocean Entrance
6.	Dam Failure concurrent with standard project flood	Standard Project Hurricane
7.	Probable Maximum Hurricane concurrent with Spring high tide	Probable Maximum Hurricane

A detailed description of these cases and their significance follows:

Case 1

This case considers the Indian Point stage at peak discharge due to runoff generated by the probable maximum precipitation over the Hudson River drainage area upstream of the site. The mean tide at the ocean entrance which corresponds to the mean sea level was selected as a boundary condition for this case.

The probable maximum flood of 1.1 million cubic feet per second, discussed in Chapter III, was considered to prevail in the Lower Hudson River for a long period. In other words, a steady state flow condition of 1.1 million cfs in the reach between the ocean entrance and the site was assumed. This assumption is conservative since the flood hydrograph under PMP conditons (Figure III-27) indicates that the duration of the peak discharge (PMF) is only several hours.

The mean sea level at the ocean entrance (the Battery) was selected as the boundary condition for this case since the tidal variations under PMP conditions would be relatively small. Examination of the USC & GS measurements[35] indicate that under such conditions the tide would be completely masked in a significant portion of the river. The results of cases No. 2 and 3 suggest that only the lower 27 miles of the river would be subjected to significant tidal oscillation.

The standard step method for water surface profiles presented earlier was employed to determine the stage at the Indian Point site for the above-delineated conditions, i.e. flow of 1.1. million cfs and mean sea level at the Battery. The results are listed in Plate V-4 and shown in Figure V-1. The results of Cases No. 2 and 3 are also shown in Figure V-1.

The flow profile corresponding to Case No. 1 conditions was verified using the newly developed Prasad's method discussed in Section B of this chapter. These results are shown in Plate V-5 and compared to their Standard Step Method counterparts in Table V-2. The agreement between the two methods is remarkable. In general the new method gives slightly lower stage values with a maximum difference of –0.5 feet.

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The probable maximum flood would result in a river stage of elevation 12.7 feet above the mean sea level at Indian Point.

The account for the influence of the local oscillatory wave at Indian Point referred to in Section A of this Chapter, the computed stage must be increased by one foot. This estimate is conservative and discussed in detail in Appendix A.

Therefore, the Indian Point river stage corresponding to Case No. 1 conditions and including the influence of the local oscillatory wave would be 13.7 feet above mean sea level.

Cases No. 2 and 3

These cases were considered to evaluate the influence of the tidal variation on the Indian Point stage during the probable maximum conditions. The procedure outlined in the previous section of this chapter was used for this purpose. The tidal variation at the ocean entrance, its influence on the probable maximum flood and river stage at Indian Point are depicted in Figure V-2. The specific details of the procedure are listed in Table V-3.

Figure V-1 documents the flow profiles corresponding to cases No. 2 and 3 conditions, probable maximum flood concurrent with

tidal flow at Indian Point and high water and low water elevations at the Battery respectively.

As indicated earlier, these results suggest that the river above the Tappan Zee Bridge, some 27 river miles above the Battery, would experience relatively small tidal variations under the PMP conditions. The upstream extent of the reach influenced by tidal variations was located using the material balance procedure described earlier.

The conclusion seems to be in good agreement with the lack of tidal oscillation observed in the upper portion of the river during the floods of March 28, 1913 and March 19, 1936. The maximum discharge measured at the head of the estuary during the 1913 and 1936 floods were 223,000 and 215,000 cfs respectively. These floods pushed the tide back downstream resulting in a flattening of the slope of the measured flood profiles some 30 miles downstream of Troy.[35] Thus, a peak discharge of more than one million cfs would be expected to mask the tide in a significantly longer reach. Figure V-1 estimates the length of this reach to be some 120 miles (between Troy and the Tappan Zee Bridge).

Theoretically speaking, therefore, the ocean-derived tidal influence would not be experienced at Indian Point under the probable maximum precipitation conditions.

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However, to be on the conservative side, the reach of tidal influence was extended upstream to the site at Indian Point. The tidal effect resulted in the following changes:

Case No.	Stage at the Battery (M.S.L.)	Peak Flow at Indian Point (cfs)	Water-surface elevation at Indian Point
1	Mean Water	1,100,000	12.70
2	High Water	1,013,500	12.37
3	Low Water	1,164,800	13.05

As a conclusion, the water-surface elevation at the site would range between 12.37 and 13.05 depending upon the tidal phase at the Battery. As before, these values must be increased by one foot to account for the local oscillatory wave at Indian Point.

Case No. 4

This case considers the maximum river stage resulting from a failure of the Ashokan Dam concurrent with the Hudson River Standard Project Flood at Indian Point. This flooding condition was discussed in detail in Chapter IV.

It should be also recalled that one of the centers of the probable maximum storm was placed over the largest reservoir in the basin. Due to the cross-shaped watershed of the basin, the reshaping of

the isohyets recommended by the Weather Bureau placed the lower center several miles upstream of the second largest reservoir.

As indicated in Chapter IV, the runoff generated by the probable maximum precipitation over the entire basin as described above would not result in failure of the basin dams.

However, because the Ashokan Dam contains the second largest volume of stored water in the basin and is closer to Indian Point than any other dam in the basin, a flooding condition resulting in a failure of this dam was considered. As suggested by the AEC, the flooding condition consisted of the simultaneous occurrence of a probable maximum flood over the Esopus Creek Basin upstream of Ashokan and a standard project flood over the rest of the Hudson River Basin.

As far as the Ashokan Dam failure is concerned, the suggested condition proved to be more critical than a hypothetical condition placing the lower center of the selected PMS directly over the Ashokan Reservoir.

The suggested condition resulted in a failure of the Ashokan Dam and a peak flow of 705,000 cfs at Indian Point. This flow would

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result in a river stage of Elevation 7.2 above mean sea level at the Indian Point site. This value corresponds to a mean sea level elevation at the ocean entrance. Because this condition is less critical than the probable maximum flood conditions (Cases No. 1 through 3), additional evaluation concerning the influence of the tidal variation was not undertaken.

Case 4 results, as well as Cases 5 and 6, are depicted in Figure V-3.

Cases No. 5 and 6

To carry the degree of severity a step further, a more critical boundary condition was imposed on two flooding conditions. The peak storm surge at the Battery resulting from the Standard Project Hurricane for the New York Harbor area was used as the boundary condition for these two cases. The flooding condition of Case No. 4, as well as a similar flooding condition, in which the effect of Ashokan Dam failure was excluded, were used for these two cases.

The New York Harbor hurricane studies[36, 37] undertaken by the New York District Corps of Engineers showed that the peak storm surge height at the Battery resulting from the Standard Project Hurricane is 11.0 feet. This value was based on the transposition of the September, 1944 hurricane to a path critical to the New York Harbor

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area, using parameters given in U.S. Weather Bureau Memoranda

HUR 7-60 and HUR 7-60a dated 27 March 1959 and 7 April 1959 respectively. The parameters finally selected for this storm were a maximum wind speed of 116 miles per hour, a central pressure range from 27.55 to 27.95 inches of mercury with a normal pressure of 30.12 inches, a radius to maximum winds of 30 nautical miles, and a forward speed of 40 knots. The corresponding peak storm surge height at Sandy Hook, some 17 miles downstream of the Battery, is approximately 12 feet.

Similar computations were made by the Corps of Engineers for a 1938 hurricane transposed to a path critical to New York Harbor. The computations, which were based on parameters given in the U.S. Weather Bureau Memorandum HUR 7-25, dated 25 February 1957, and a forward storm speed of 35 knots, yielded surges of 8.9 feet at Sandy Hook and 8.8 feet at Fort Hamilton. The maximum surge of record at Fort Hamilton, which is 8.2 feet, was experienced during the extratropical storm of 25 November 1950.

To maintain the selection of severe conditions and in accordance with the high degree of conservatism adopted in this study, the value based upon the transposed September 1944 hurricane was selected. This value is 75 per cent higher than the maximum storm surge during

hurricane “Donna” that struck on September 12, 1960. Hurricane “Donna” was the latest storm to have the greatest effect in the New York Harbor area since the September 1821 hurricane.

Simultaneous occurrence of the standard project hurricane with a flooding condition resulting from runoff generated by a standard project precipitation over the entire basin would result in a river stage of elevation 13.0 at the site.

Case No. 6 considers an even more critical set of occurrences consisting of the simultaneous occurrence of the following three severe conditions:

1. Probable Maximum Precipitation over the Ashokan Reservoir drainage basin resulting in a failure of the Ashokan Dam.

2. Runoff generated by the standard project precipitation over the rest of the basin.

3. Peak storm surge at the Battery resulting from the Standard Project Hurricane for the New York Harbor area.

This case would result in a river stage of elevation 14.0 at Indian Point. This value is higher than any of the previous five estimates. The simultaneous occurrence of the above-delineated three severe conditions is extremely remote. Moreover, each one

of these three individual conditions was based upon several conservative assumptions discussed in previous chapters. Combination of these conditions, therefore, increases the degree of conservatism substantially.

The resulting flow profiles corresponding to this case as well as Case No. 5 are shown in Figure V-3.

For convenience, the results of the six cases are summarized and compared in Table V-4. The computer printouts for the individual cases are presented in Plates V-2 and V-6 through V-10 respectively.

Case No. 7 considers the water surface elevation at the site resulting from the occurrence of a PMH concurrent with spring high tide. This case was the main subject of Q.L. & M's report of February, 1969 which is appended to the main body of this report. The results are included in Table S-1 for convenience and comparison purposes. A summary of the determination of the results follow.

Using the general equations for surge in an estuary, presented in a paper by J. Proudman, an internationally respected oceanographer, a procedure to route the hurricane surge through the Hudson Estuary was developed. This procedure considered channel frictional resis-

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tance, geometric changes in cross-section, and estuary storage of the surge flow. The large storage capacity of Haverstraw Bay acts as an on-line reservoir to attenuate the hurricane surge as it flows toward Indian Point.

A conservative estimate of the routed hurricane surge height at Indian Point is 8.8 feet above mean tidal level, with an additional surge due to wind setup of 1.0 feet, and combined with a spring high tide of 3.0 feet. The total calculated elevation is 13.5 feet above mean sea level.

In addition, estimated local oscillatory waves will reach a height of 1.0 feet above the calculated level in the rive. Provision should be made for protecting any structures and appurtenances from this 1.0 foot wave action above the 13.5 foot mean sea level calculated elevation. This value was added to all of the computed stages corresponding to the seven cases and the results are listed in the last column of Table S-1.

APPENDIX A

MAXIMUM RIVER ELEVATION AT INDIAN POINT RESULTING FROM PROBABLE MAXIMUM HURRICANE AND SPRING HIGH TIDE *

A. Hurricane Parameters

The probable maximum hurricane (pmh) for zone 4, latitude 41ºN, as defined by the United States Weather Bureau, has the following meteorological characteristics.

Central Pressure	27.26 inches of Hg
Radius of Maximum Winds	24 nautical miles (weighted mean)
	8 nautical miles (min. limit)
	48 nautical miles (max. limit)
Forward Speed of Hurricane	34 knots (weighted mean)
	15 knots (min. limit)
	51 knots (max. limit)
Maximum Wind Speed	124 mph to 136 mph
	127 mph (at mean forward speed)

Inspection of published data showed that the vast majority of the hurricane parameters cluster around the mean values reported above.

The minimum and maximum ranges represent the limits of the frequency distribution of the parameters as defined by isolated storms.

Parameters used in the analysis included a forward speed of 34 knots, a radius of maximum wind of 24 nautical miles, and a maximum wind speed of 127 mph. These values represent a probable maximum

* The material included in Appendix A is essentially that of Chapter III of the February, 1969 report.

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hurricane with a recurrence interval estimated within the range of 1,000 to 10,000 years.

I should be noted that the very low probability of occurrence of the pmh must be multiplied by the probability of occurrence of spring tides, which occur twice in a lunar cycle of approximately one month, in order to determine the probability of occurrence of the hurricane surge concurrent with maximum high tides. The resulting flooding condition is, therefore, very conservative.

For this hurricane, procedures published by the U.S. Army Coastal Engineering Research Center were used to determine the surge height at the Battery.

B. Summary of Procedures Employed

The initial step involved the construction of the isovel field for the meteorological characteristics of the pmh. The field was constructed using standard procedures developed by the U.S. Weather Bureau in the aforementioned report.

Following the construction of this field, the wind stresses for several lines passing through the hurricane and parallel to the direction of its movement were computed. The stress coefficient for a hurricane moving over a sloping bottom was used. The maximum

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wind stresses occur on a line which passes through the radius of maximum winds.

the maximum wind stresses were routed along several tracks along the Atlantic Coast, to determine which tract would produce the highest surge at Sandy Hook. After selecting this critical hurricane track, the maximum hurricane stress was then routed through New York Harbor to the Battery.

The greatest surge height resulting from the probable maximum hurricane was computed to be approximately 15.6 feet in New York Bay. This calculation agrees with a published estimate of a maximum surge of 15.3 feet by B. Wilson.

Using an additional stage increase of 1.9 feet due to a condition of atmospheric pressure reduction, the estimated surge in New York Bay is 17.5 feet above mean sea level. A stage-time hydrograph was estimated according to the variation predicted by Wilson.

J. Proudman solved the general equations of motion and continuity for a long progressive wave in an estuary. He developed an equation to compute stage at any point in the estuary as a function of an input stage variable at the estuary's mouth.

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Using Proudman's equation for the case of an estuary where the product of base width multiplied by celerity at various cross-sections are similar, a routing of a critical hurricane stage hydrograph was made from the Battery to Dobbs Ferry. This reach is essentially uniform.

However, the reach from Dobbs Ferry through Haverstraw Bay, to Indian Point is approximately double the width of the former reach. This allows considerable attenuation of the hurricane surge as a result of the increased channel storage. In this latter reach, a more complex procedure was used to route the Dobbs Ferry stage-time hydrograph upstream to Indian Point.

The surge routing procedure employed considers the effect of friction in the channel, and the large storage capacity of Haverstraw Bay. Haverstraw Bay acts as an on-line reservoir to attenuate any surge condition.

The attenuated surge stage at Indian Point is approximately 8.8 ft. above mean tidal level. The calculated subsidence is affected somewhat by variations in the channel cross-section and the assumed resistance formula. Values of depth and width were computed at channel sections where the river's geometry changed. Linear vari-ations were assumed at intermediate sections. The channel was divided into eleven reaches between the Battery and Indian Point.

Prior calculations show that the additional increase in surge height due to the forward advance of the hurricane from the Battery to Indian Point was about 1.0 feet. These calculations assumed that the hurricane would essentially follow a direction parallel to the Hudson, but that winds would be reduced somewhat by the sheltering topography surrounding the river.

The basic theory used was the Corps of Engineers procedure for surge routing in open seas.

In the evaluation of the maximum surge height, it should be remembered that the wind which produces the storm surge, also produces oscillatory short period waves. Calculations for the area downstream from Indian Point show that the maximum fetch length for wave development is 6.85 miles. In order to transfer energy fully from wind to water, a high speed wind must exist for a minimum duration of many hours or days. The short duration of high speed winds as the hurricane passes the critical Indian Point fetch reduces the height of waves. Increased stage is conservatively estimated at 1 foot.

It is possible for the hurricane surge to occur at high tide conditions. The spring high tide at Indian Point is estimated to increase the stage by 3.0 feet above M.T.L. (very conservative),

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The routed surge and water elevations for the probable maximum hurricane moving along a critical track to Indian Point are summarized below:

	Elevation
Routed Hurricane Stage	8.8 feet
Additional surge due to wind set up from the Battery to Indian Point	1.0 feet
Spring high tide	3.0 feet
	12.8 feet above M.T.L.
	or
	13.5 feet above M.S.L.

Estimated maximum local oscillatory
wave height above M.S.L. 1.0 feet

The routed hurricane surge elevation of 13.5 feet exceeds the 11.7 feet elevation computed for the flood runoff condition and is therefore considered to be the controlling elevation for flood protection. Historical records of flooding at Indian Point show that the extreme high water mark of 7.4 feet above M.S.L. occurred during the November 25, 1950 hurricane Unfortunately, data concerning water elevations at both the Battery and Indian Point are meager. A general idea of the magnitude of surge reduction in the Hudson Estuary can be obtained from the depth increases between mean tidal level to mean high water. At the Battery, the depth increases by

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2.3 feet, and at Indian Point, by 1.5 feet. This is a ratio of about 1.5. The corresponding ratio of the 17.5 feet hurricane surge at the Battery to the attenuated 9.8 surge at Indian Point is about 1.8. The 1.8 ratio is reasonable, since a higher surge travels at a faster velocity, with greater friction losses that a lower surge. The necessary calculations, figures and tables for the hurricane surge routing are presented below.

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APPENDIX A'

Selected Calculations and Figures from
the Preliminary Report

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APPENDIX B

FLOOD STUDY

OF

UPPER HUDSON RIVER BASIN

MARCH 21, 1969

STONE & WEBSTER ENGINEERING CORPORATION

BOSTON, MASS.

B-1

QUIRK, LAWLER & MATUSKY ENGINEERS

REPORT ON
FLOOD STUDY
UPPER HUDSON RIVER BASIN

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INTRODUCTION

PURPOSE AND SCOPE

In order to develope hydrological data for the Upper Hudson River Basin during flood conditions, an investigation was made to determine the maximum predictable stage of the Hudson River at Stillwater less than five miles upstream of Mechanicville and immediately above the mouth of the Hoosic River.
A determination of stage was made for the following two conditions:

1. Probable Maximum Flood – Stage at peak discharge due to runoff from the probable maximum precipitation over the drainage area upstream of Stillwater.

2. Dam Failure – Stage due to an assumed failure of the Conklingville Dam at the time the Sacandaga Reservoir is at its highest stage from the probable maximum flood.

SUMMARY OF RESULTS

The results of the investigation fro the two conditions is as follows:

1. The probable maximum flood would have a peak flow of

300,000 cfs and result in a river stage of El. 110.0 (USGS Datum) at the Stillwater.

2. The maximum river stage resulting from a failure of the Conklingville Dam would be El. 124.0 (USGS Datum) at the Stillwater.

DESCRIPTION OF THE BASIN

The basin covered by this study is that portion of the Upper Hudson River Basin north of Stillwater, New York and is located in the central part of the great trough that extends northward from New York harbor to the St. Lawrence River. Its drainage area, lying principally in east central New York, is about 3,760 square miles. (See Figure 1.)

The major topographical feature of the Watershed is the rugged eastern portion of the Adirondack Mountains which has peaks rising up to abut 5,000 ft. The northern part of the basin is wilderness country which has dense forests and contains many lakes and small streams. The valleys of the basin are covered by extensive deposits of glacial material produced by the ice sheets which formerly covered the area.

The Hudson River has its source in the Adirondack Mountains at Lake Henderson at an elevation of 1,808 ft above sea level and flows southerly for a distance of about 315 river miles to enter New York Bay. From its source to Stillwater Dam the river drops about 1,725 ft in a distance of about 140 river miles. Abrupt drops of 80, 45 and 60 ft occur at Palmer Falls, Glens Falls and Bakers Falls, respectively.

2.

The tributaries of the Upper Hudson are numerous, the more important in order of size being the Sacandaga River uniting with the main stream at Hadley, the Schroon River at Thurman, Batten Kill and Fish Creeks at Schuylerville, and the Indian River near Gooley. The flow into the Hudson from these tributaries is controlled by dams which form reservoirs that are maintained at relatively fixed levels.

The climate of the Upper Hudson River Basin is characterized by long, cold and snowy winters and short, mild summers. The average annual rainfall varies from about 40 in. in the central portion to over 50 in. in the headwaters of the Sacandaga River. The precipitation is normally fairly evenly distributed throughout the year, with only a slight rise during the summer.

The average annual runoff from the basin is about 55 percent of the average rainfall, with the runoff from the easterly half being a lower percentage than the westerly portion.

STREAMFLOW DATA

Stream gaging stations have been maintained by the U.S. Geological Survey and others for various periods of time at over 30 locations within the upper Hudson River watershed above Stillwater.

The main river gaging station with the longest period of published record is located at Mechanicville, New York, four and one-half miles downstream of Stillwater The drainage area above the gage is 4,500 square miles, of which approximately 650 square miles is in the Hoosic River watershed which enters the Hudson River one and one-half miles above the gage. Continuous stream flow data at Mechanicville have been published for the period October 1887 to September 1956. The USGS have maintained a gaging station on the Hoosic River at Eagle Bridge with a drainage area of 510 square miles from 1910 to date. Using the records of both these gages, an estimate of past stream flow at Stillwater can be made withr easonable confidence.

A summary of runoff data at 10 stations representative of the basin in general is shown in Table 1.

STREAMFLOW REGULATION

The Upper Hudson River is well regulated from the standpoint of flood control. The most important regulating feature is the Sacandaga Reservoir, operated by the Hudson River – Black River Regulating District. This reservoir, controlling the Sacandaga River drainage area of 1,044 square miles or nearly 30 percent of the total upper basin area, has a storage capacity of 866,000 acre-ft at spillway crest level. The high crest and relatively

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3.

small spillway at Conklingville Dam also provide the reservoir with a large amount of flood detention capacity above the rated storage. Additional tributary regulation is provided by Indian Lake Reservoir, operated by the Indian River Company. This reservoir has a capacity of 114,000 acre-ft and controls 131 square miles of Indian River drainage.

Natural storage for reducing flood flows is also provided by widespread lake areas such as Schroon Lake, Chain Lakes and

Saratoga Lake. Small dams at the outlet of Schroon Lake and on Fish Creek downstream of Saratoga Lake effectively increase the natural lake storage and reduce the flood potential of drainages of about 560 and 250 square miles on the Schroon River and Fish Creek, respectively.

On the main stem of the Upper Hudson there are a number of man-made controls, principally in the form of power dams. The river profile is shown on Figure 2. From Corinth to Fort Edward, a distance of about 23 river miles, there is a series of eight privately owned power dams. In this distance, the river drops about 420 ft. Although the storage capacity of none of these dams is very large, their combined effect is to retard flood flows and reduce flood stage downstream.

Proceeding downstream, the Champlain Canal, part of the New York State Barge Canal System, makes use of the Hudson between Port Edward and Troy. The river is channeled and controlled for navigation by a series of dams and locks. The river drops about 60 ft between Fort Edward and the Upper Mechanicville Dam. This section of the river channel, generally larger in cross sectional area than upstream, provides additional flood control regulation through natural valley storage.

RECORD FLOODS

The highest flood on record on the Hudson River at Mechanicville occurred in March 1913, prior to construction of the Sacandaga Reservoir. The flow reached a peak of 120,000 cfs on March 28. The rainfall causing this flood was general over the entire north-central and northeastern United States. The storm was preceded by a thaw with temperatures rising to 70 F, followed by over 5 in. of rain in five days. The estimated maximum mean daily flow at Stillwater was 107,000 cfs during this storm.

The second greatest flood at Mechanicville was the New Year flood of 1949, which had a peak flow of 118,000 cfs on January 1, 1949. The runoff was the result of over 6 inc. of precipitation which fell as a combination of rain, snow and sleet. The estimated maximum mean daily flow at Stillwater was 61,5000 cfs. The significantly lower flood flow at the Stillwater in comparison

with the 1913 storm is due to large measure to the flood storage provided by the Sacandaga Reservoir. The flood flows in the Hudson River above the Sacandaga River and in the Sacandaga above Conklingville for both storms were approximately equal. However, the discharge of the Sacandaga during march 1913 reached a peak of 35,5000 cfs, while during the 1949 flood the two-day average discharge was less than 500 cfs.

Following is a list of the major floods and maximum mean daily flows recorded at Mechanicville and estimated at Stillwater:

Date	Recorded Flow at Mechanicville, cfs	Estimated Flow at Stillwater cfs
March 1913	113,500	107,000
April 1914	64,800	59,000
April 1922	72,900	67,000
March 1936	72,700	54,500
September 1938	65,600	38,000
January 1949	94,400	61,500

Maximum stages and discharge for key gaging stations within th watershed are shown on Table 2.

5.

DEFINITION

The probable maximum flood has been defined as an estimate of the hypothetical flood characteristics that are considered to be the most severe “reasonably possible” at a particular location, based on comprehensive hydrometeorological analyses of critical runoff producing precipitation and hydrologic factors favorable for maximum flood runoff. [(1)] It has been further described as the estimate of the boundary between possible floods and impossible floos. The objective, therefore, is to obtain a flood that has a chance of occurance approaching zero or a return period of infinity.

Using the above definition as a guide the probable maximum flood for the Hudson River at Stillwater was developed as follows:

1. The basin was divided into subbasins or hydrologice units on the basis of tributary drainage area and location of hydraulic controls and unit hydrographs developed for each subbasin.

2. The probable maximum precipitation was applied to the unit hydrographs with the appropriate infiltration losses to develop the flood hydrograph for each subbasin.

3. Starting at the uppermost reach in the Hudson River, the subbasin flood hydrographs were combined in their proper time sequence and routed downstream. The process of combining inflows and routing continued in the downstream direction until the flow at Stillwater was determined.

A detailed description of the development of the hydrometeorological characteristics of the basin and synthesis of the probable maximum flood are given in the following sections.

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6.

PROBABLE MAXIMUM PRECIPITATION

The records show that a number of large floods on the Hudson river are the results of an early spring storm combined with melting snow. However, the data contained in Hydrometeorology Report No. 33(2) indicate that the probable maximum precipitation for the spring months would be about 50 percent of the all season probable maximum precipitation. In addition a study made by the U. S. Army Corps of Engineers for the adjacent Connecticut River Basin concluded that the maximum storm rainfall would be at least equal to the maximum combination of rain and melting snow(3). It was therefore concluded that use of a summer or fall storm would produce a runoff of at least as great as a spring storm with snow melt.

The probable maximum precipitation used in this study was developed from the 72-hour precipitation and depth-duration-drainage area curves prepared by the Hydrometeorologoical Section, U.S. Weather Bureau, for the Delaware River at the U.S. Army Corps of Engineers dam site at Tocks Island. (4)

Tocks Island rainfall data include two subtropical storm patterns A and B, which have been given transposition limits. However, since there is a general similarity in orientation, size, shape and topography of the Delaware River Basin above Tocks Island and the Upper Hudson River Basin, both storms were transposed over the Upper Hudson River Basin within the orientation limits prescribed by the Weather Bureau so as to produce the maximum rainfall on critical areas. It was found that transposed storm pattern B produced the greater rainfall and it was used in this study. This storm pattern is of the extra-tropical type where rainfall centers are associated with convergence in the vicinity of waves on frontal boundaries. In its early stages, it was of tropical character.

The 72-hour isohyets for storm pattern B in the transposed location used in this study are shown in red on Fig. 1A. The pattern produces a 72-hour total precipitation equivalent to an average of 14,3 in. over the entire drainage area above Stillwater. Using applicable rainfall data contained in Hydrometeorological Report No. 28, [5] and Hydrometeorological Report No. 40, [6] it was determined that the probable maximum 72-hour precipitation for the area under study would be 13.0 and 14.4 in., respectively.

Incremental average depths for subdurations were obtained by applying the time ratio from the Tocks Island DDA curves to the 72-hour basin probable maximum precipitation. The precipitation for each subbasin was determined by planimetering the area between isohyets within the subbasin boundaries. The incremental

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average depth for the subduration for each subbasin was obtained by applying the same ratio as found for the entire basin. The six-hour incremental probable maximum values were rearranged in accordance with the following criteria recommended in Hydrometeorological Report No. 40:

a. Group the four highest six-hour increments of the 72-hour PMP in a 24-hour period, the middle four increments in a 24-hour period, and the lowest four increments in a 24-hour period.

b. Within each of these 24-hour periods, arrange the four increments in accordance with sequential requirements; that is, the second highest next to the highest, the third highest adjacent to htese, and the fourth highest at the either end.

c. Arrange the three 24-hour periods in accordance with the sequential requirements: that is, the second highest 24-hour period next to the highest, with the third at either end. Any possible sequence of three 24-hour periods is acceptable with the exception of placing the lowest 24-hour period in the middle.

The 6-hour incremental precipitation depths for the entire area under study arranged in order according to the above criteria are as follows:

Hour	Incremental Depth-in	Accumulative Depth-in
6	0.2	0.2
12	0.4	0.6

18	0.6	1.2
24	0.2	1.4
30	1.3	2.7
36	7.8	10.5
42	2.3	12.8
48	1.1	13.9
54	0.1	14.0
60	0.1	14.1
66	0.1	14.2
72	0.1	14.3

UNIT HYDROGRAPHS

Unit hydrographs were derived for 19 subbasins in the basin by the methods described in "Unit Hydrographs – Part – Principles and Determination,"[7] and "Hydrology Guide for Use in Watershed Planning".[8]

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8.

Unit hydrographs were determined from records of nine gaging stations located on the various streams and reservoirs within the basin having drainage areas varying in size from 90 to 1,044 square miles. In general, the unit hydrographs were developed from the hurricane storm of September 1938 and compared with the next largest storm for which data were available. The second storm used varied from area to area. The storm of September 1938 produced the largest floods without snow melt for which adequate records are available. It was not possible to develop a unit hydrograph for the Indian Lake Reservoir and Batten Kill from the September 1938 storm because of inadequate rainfall data, and other storms were used. A computer program described by D. W. Newton and J. W. Vinyard,[9] was used in developing the unit hydrographs for the gaged areas. Data for the observed unit hydrographs are shown on Table 3.

Unit hydrographs for approximately 70 percent of the total drainage area were developed from rainfall and runoff records. The unit hydrographs for the Sacandaga Reservoir and Indian Lake inflows and the Hudson River at the USGS gage at Gooley were used without modification. The unit hydrographs developed at the gaging stations on the Schroon River, Batten Kill and Kayaderosseras Creek were transposed to their respective mouths using Snyder's coefficients and the method outlined in "Unit Hydrographs – Part I Principles and Determination."

Synthetic unit hydrographs were developed for the large ungaged subbasins using Snyder's relations and coefficients developed from the gaged area unit hydrographs and data developed by Taylor and Schawarz, [10] for basins in the north and middle Atlantic states.

The remaining unit hydrographs were developed by the method contained in "HYdrology Guide for Use in Watershed Planning." These unit hydrographs are for the short, very steep streams immediately adjacent to the Hudson River. In general, the drainage areas for the individual streams are small, and they have been combined into large subbasins extending between routing points on the Hudson River.

No unit hydrographs have been developed for a few small, mostly water surface subbasins adjacent to the Hudson River. It was assumed that runoff from the precipitation is instantaneous for these areas.

The data for the 19 subbasin unit hydrographs are shown on Table No. 4 and the 6-hour unit hydrographs are listed on Table No. 5.

STORM HYDROGRAPHS

In order to define the time discharge relationship for inflows to the major reservoirs and to the Hudson from its tributaries and the drainage areas along its banks, storm hydrographs were developed using the unit hydrographs previously described. The time distribution of the rainfall in each subbasin was the same as was used for the entire basin. For a subbasin, the amount of direct runoff resulting from the rainfall during each time increment was then determined using the method described in Chapter II, Section 53, "Design of Small Dams."[11] the curve numbers used in the above method were selected after analysis of precipitation and runoff data in the basin. Where these data were available for more than one storm, the curve number was used which corresponded to the largest runoff-to-precipitation ratio.

The direct runoff increments were applied to the unit hydrograph to produce the flood hydrograph for beach subbasin. With base flows added where applicable, these storm hydrographs defined the inflows into the Hudson River from its tributaries and bank drainage areas.

RESERVOIR FLOOD ROUTING

Inflows to Indian Lake, Sacandaga Reservoir, Stewarts Bridge Reservoir and Saratoga Lake resulting from the probable maximum storm were routed through these reservoirs using a graphical solution of the step method of flood routing. The data used for routing were: (1) inflow storm hydrographs, except for Stewarts Bridge; (2) reservoir storage curves, and (3) spillway rating curves. At Stewarts Bridge, the inflow used for routing was the discharge from Conklingville Dam with a small adjustment for inflow from the intervening drainage area.

Reservoir conditions assumed prior to the storm and predictions of reservoir operation during routing of the probable maximum flood are shown in the following tabulation:

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10.

Reservoir	Water Surface Elevation Before Storm	Conditions During Routing Max W.S. El.	Max Discharge Cfs	Time Hours
Indian Lake	1651.35 (spillway crest)	1660.5	10,950	72
Sacandaga	768.0 (bottom of flood storage)	784.0	44,000	90
Stewarts Bridge	705.0 (normal H.W.)	710.5	43,000	90
Saratoga Lake	202.4 (top of flash-boards at Winnies Reef)	215.4	14,200	84

*Time from beginning of P.M.P.

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11.

The Sacandaga Reservoir flood control storage pool between Elev. 768 and 771 is rarely used except during the spring months when an effort is made to store the maximum volume of snow melt and spring precipitation. In over 45 years of operation the maximum reservoir state was Elev. 769.43 on April 19, 1954, and at that elevation less than half of the flood control storage pool was used. The probable maximum precipitation for this study is a late summer or fall storm and the Sacandaga Reservoir would normally be below the flood pool. However it is possible that a prior storm could result in an abnormally high level and it is judged that an initial reservoir level of Elev. 768.0 for this study is conservative and reasonable. The other reservoirs have no specified volume allocated for flood storage and the initial water surfaces were chosen as being the maximum that could exist at the time.

Discharges were calculated at Indian River Dam with the sluice gates closed, at Conklingville Dam with the Dow Valves open and no flow through the E. J. West hydroelectric Station, and at Stewarts Bridge with the five Tainter gates open and no flow through the turbines.

At Indian Lake, the maximum water surface elevation predicted is approximately at the crest of the earth dike section of the dam. it was assumed that for a flood of this magnitude and duration, precautionary measures would be taken at the dam to prevent overtopping of the earth sections and overflow discharge would be confined to the masonry sections. Minimum freeboards at Conklingville and Stewarts Bridge are 10.5 ft and 3.5 ft, respectively. Winnies Reef, the control for Saratoga Lake, would be submerged at the flow predicted, but overtopping this low concrete structure presents no serious structural hazard.

Outflows derived from the preceding reservoir routings were then routed down their respective channels to the Hudson River using the stream flood routing techniques described in the following sections.

CHANNEL ROUTING METHODS

Two general methods were used in routing the probable maximum flood down the main stem of the Hudson River. The first method, used for the upper portion of the river between North Creek and Hadley and for the lower portion between Fort Edward and Stillwater, was based on what is commonly called the coefficient method of routing. The procedures used generally followed the methods outlined in Corps of Engineers Engineering and Design Manual on Flood Routing. (12) The second method used was the step method of reservoir routing. This method was used for the

portion of the river between Hadley and Fort Edward because of the many dams within the reach which control the river flow.

RIVER CHANNEL CHARACTERISTICS

Most of the effort required to perform the flood routing was connected with the collection and evaluation of physical data necessary to define the river flow and valley storage characteristics. The task of determining these characteristics at the stages anticipated was greatly complicated by the fact that there are no floods of record approaching the maximum discharges being studied.

In order to route the flood using the two methods mentioned above, several river parameters had to be defined. Reduced to simplest form, these were:

a. Hydraulic characteristics of the river, which include area, top width and hydraulic radius at enough river cross sections to adequately define the channel.

b. Manning's "n"

c. State-discharge-volume relationships for the various reaches of the river

c. X and K, which are constants used to determine routing coefficients

The methods used to determine the above parameters will be described briefly.

Hydraulic Characteristics

The hydraulic characteristics of the river were determined mainly from river cross section data. Approximately 120 river cross sections were used between Mechanicville and Thurman Station, a distance just under 75 river miles. The majority of the above water portions of these sections were taken from large scale topographic maps furnished by Niagara Mohawk Power Corporation. Large scale topographic maps were not available for a section of the river below Fort Edward and another section between Palmer and Hadley, and field cross sections were made in the fall of 1968. Underwater soundings, by fathometer, for most cross sections were obtained from the Albany office of the U.S. Geological Survey, which were combined with the above data to obtain a complete cross section of the river channel and banks. Additional valuable information on the section of the river below Fort Edward was obtained from the New York State Barge Canal charts of the Chamblain Canal. Use was made of all available

U.S. Geological Survey maps of the area for general river channel topography and to fill in any gaps in the above-mentioned information.

In the fall of 1968, prior to making field surveys, a field reconnaissance of the river and its major tributaries was made by two engineers from Stone & Webster Engineering Corporation.

Accompanied by engineers from Niagara Mohawk Power Corporation, they visited all the major hydraulic structures in the drainage basin and collected hydraulic data and pertinent drawings of the structures wherever available from state and federal agencies and private industry. In addition, a determination was made of where field survey data were required and a photographic record made of these areas and at the locations of available cross sections.

Mannings's "n"

Limited data were available for a determination of Manning's "n" from velocity measurements made by the U.S. Geological Survey. These data were used to determine an "n" value at three locations using the velocity distribution method. (reference 13, p.207). An additional value for "n" was calculated from flood stage discharge data. Based on these calculations and the visual observations recommended by Barnes[(14)] made during the field reconnaissance, an estimated average value of 0.025 was used in the final calculations. However, because of uncertainty in the determination of average "n" values at flood stages, trial calculations were made assuming n=0.035. The higher value, while tending to increase river stage for a particular discharge, also increased the valley storage which resulted in a somewhat lower maximum discharge at the site. Therefore, from these calculations it was judged that the stage variation which would result from varying "n" within reasonable limits would be less than the tolerances of the other physical data upon which the calculations were based.

Stage-Discharge-Volume

Both of the methods used for flood routing required determination of the state-discharge-volume relationships for certain reaches of the river. To facilitate this, the river from Mechanicville to Thurman Station was divided into 13 reaches, each reach starting at a control point on the river. Backwater curves were calculated using the standard step method (reference 13, p.265) for each reach, starting with assumed discharges at the downstream control. A digital computer program was used to make these calculations. Input required included the hydraulic characteristics, Manning's "n," a discharge rating curve for the downstream control and an estimate of eddy and form losses from bends and contractions in the river channel. The output

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consisted of a water surface profile and volume of water stored in the reach at a particular steady state discharge.

At high flows, the river stage would in many cases overflow the banks. A precise determination of the amount of flow in the overbank under these conditions in the absence of records is impossible. A solution to this problen was effected by the use of two different cross sections at each river station where hydraulic characteristics were determined. One section was made for the complete river valley including overbank. The second section, which was in most cases less than the first, included only that portion of the river valley in which appreciable flow was judged to take place. The determination of this second or flow section was based largely on judgement after careful consideration of topography, vegetation and all physical features, including those which were man-made, in the river valley. All storage values were based on the overbank sections, while backwater profiles for steady state flow were determined using the flow sections.

Rating curves for most of the dams in this section of the river were furnished by the owners. Theoretical rating curves were calculated at several points and all rating curves were extended theoretically when the discharge exceeded the design capacity of the spillways. The rating curves were developed using the fixed crest of the structures, assuming no flash boards in place, all spillway gates open, and no flow through turbines or supplementary outlets.

In computing the valley storage for the reach between Stillwater and Hadley, it was assumed that all dams could pass the probable maximum flood without failure. It was further assumed that the volume below the fixed dam crest under consideration could be ignored and that only the volume above the crest was usable.

X and K

The interrelated constants S and K, used to determine the coefficients for routing by the coefficient method, were evaluated on the section of the river above Thurman Station by the inverse flood routing process. Below Thurman Station, because of lack of adequate flood data, evaluation of these constants was made from the stage-discharge-volume relationship for each reach. This procedure required determination first of K, which is the travel time for an elemental discharge wave to traverse the reach. Then an X, which is a measure of the wedge storage in the reach, was assumed and the flood was routed.

Finally, to check the correctness of the assumed X, the actual wedge storage was calculated using a step backwater calculation with varying flow within the reach corresponding to the inflow and outflow hydrographs. Trials were continued until the assumed and calculated X's were substantially in agreement.

Values used for X and K were as follows:

River Reach	X	K(Hr)
Gooley to North Creek	0.4	3
North Creek to Thurman Station	0.3	3
Thurman Station to Hadley	0.1	6
Fort Edward to Thomson	0.3	3
Thomson to Stillwater	0.3	6

The routing coefficients used were those tabulated in reference 12 for the above values of X and K when t=3 hrs.

ROUTING PROCEDURE

Routing the maximum probable flood from the headwaters to Hadley was accomplished using the coefficient method and the procedure described in Reference 12. The routing started with the inflow flood hydrograph at Gooley and proceeded to Hadley in two steps. Tributary inflow and riverbank runoff were combined with main stream flow at the appropriate times. Discharge from Indian Lake was routed to the Hudson River using the coefficient method also.

From Hadley to Fort Edward the river was divided into reaches varying from about two to seven miles long, each reach terminating at one of the eight dams shown on Figure 2 for that stretch of the river. Inflow from the Sacandaga River was combined with flow in the Hudson at Hadley and routed from dam to dam using the step method of reservoir routing. Using the storage curves developed from backwater computations, routing was accomplished in three-hour time steps with addition at appropriate locations of the flood hydrographs of the inflows along the riverbanks from the subareas of subbasin 9 shown on Figure 1. Checks of wave travel times for the various reaches were made by calculating the change in storage with discharge for the various discharges considered. These travel times were found to vary from a few minutes to about one and a half hours, considerably less than the time step used.

Flow characteristics and water surface profiles for the reaches below Fort Edward were determined starting at the Upper

Mechanicville Dam, because it was not known initially where river

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control might be for establishing stage at Stillwater. It was found that critical depth occurred at Stillwater Dam for flow up to at least 300,000 cfs. It was also determined that critical depth did not occur at Crockers Reef except at flows below 50,000 cfs. Therefore, this section of the river was divided into three reaches terminating at the dams at Fort Miller, Thomson and Stillwater. Calculations of change in storage with discharge were made for these reaches and the travel times were determined as being about three hours from Fort Edward to Thomson and about six hours from Thomson to Stillwater. The discharge at Fort Edward was accordingly routed to Thomson and then to Stillwater using the coefficient method, again adding tributary and riverbank inflow at appropriate times and locations.

STAGE-DISCHARGE RELATION

the stage-discharge relation shown on Figure 4 is for the river section at the northern end of the Easton Site, approximately 4 ½ miles upstream of Stillwater Dam.

The rating curve was developed from backwater curves calculated from Stillwater or Upper Mechanicville Dams. Because Stillwater Dam is a relatively low structure, the backwater computations were initiated at Upper Mechanicville and flow conditions checked for control at Stillwater Dam. It was found that at flows greater than 500,000 cfs Stillwater Dam became submerged and control shifted downstream to Upper Mechanicville Dam. The backwater curves were computed as described in the section under flood routing and verified, where possible, with Champlain Barge Canal gage records.

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17.

DAM BREAK WAVE

UPSTREAM RESERVOIRS

The major existing dams with significant storage located above Stillwater are listed in Table No. 6. In addition, there are a number of smaller dams with minor storage capacities located on the various tributaries above Stillwater. It is obvious that the largest single block of storage is contained in the

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Sacandaga Reservoir behind the Conklingville Dam, with a volume greatly in excess of all other reservoirs combine. Failure of the Conklingville Dam would release the largest confined volume of water in the basin and would result in the highest conceivable stage at Stillwater if it should occur coincident with flood conditions.

CONKLINGVILLE DAM

The Conklingville Dam, completed in 1930, is located on the Sacandaga River, as shown on Figures 1 and 2. It is an earth dam founded on rock and earth, with a concrete gravity spillway built on rock. The earth dam was built by the semi-hydraulic fill method. It has a crest width of 43 ft. at El. 794.5 and a base width of 650 ft. at its maximum height of 96 ft., the width-to height ratio being 6.75 to 1. At spillway crest El. 771.0, the reservoir has a total capacity of 37.8 billion cubic feet. The reservoir volume above El. 768.0 is reserved for flood control purposes. The diversion canal and spillway are located in a rock section away from and to the left of the earth dam. The outlet works consists of three 8 ft. diam. Dow Valves and two 18 ft. by 8 ft. siphons. The spillway is an ungated free overflow section 405 ft. long.

The dam was designed for large surcharge, having a freeboard height of 23.5 above the spillway crest. The maximum level attained by the reservoir during the 38 years of operation was El. 769.43, 1.57 ft below the spillway crest. When routing the probable maximum flood through the reservoir, the maximum stage reached was El. 784.0, 10.5 ft below the crest of the earth dam.

The dam was built by the State of New York and is maintained and operated by the Hudson River – Black River Regulating District. Subsequent to award of the construction contract, the rock excavation for the spillway channel was increased. This rock material was spoiled at the downstream toe of the dam, providing an unusually large rock toe section. The dam has an ample cross section, is inherently stable, and has proven its ability to withstand severe earthquakes, as mentioned below.

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A series of significant earthquakes have occurred in the area since completion of the dam. The highest recorded earthquake occurred on April 20, 1931, at Lake George, New York. This earthquake was recorded at Intensity VII (Modified Mercalli Scale) and was perceptible over an area of 60,000 square miles. The Sacandaga Reservoir level was at E. 752.2 at the time.

Following the earthquake, a two-day inspection of the dam and reservoir was conducted and no damage was found.

there is no reason to believe that the Conklingville Dam would fail from earthquake, overtopping, or any other natural cause. However, because it does contain the largest volume of stored water in the entire basin, it was used in determining the stage at Stillwater from an assumed dam failure.

It was assumed that failure would occur with the reservoir at its maximum possible level, El. 784.0, which is caused by runoff generated by the probable maximum precipitation. At this surcharged elevation, the reservoir contains approximately 54 billion cubic feet of water.

It was further assumed that Stewarts Bridge Dam would fail prior to Conklingville Dam. This assumption is based on the fact that, in routing the maximum flood through the tow reservoirs, it is found that the freeboard at the Conklingville Dam would be 10.5 ft as compared to 3.5 ft at the Stewart Bridge Dam. While this shows that neither of the dams would be overtoppped during the maximum probable flood, in the event of a catastrophe, the smaller freeboard at the Stewart Bridge Dam would make it more likely to fail first.

MODE OF FAILURE

In a hypothetical study of this type, the first assumption to be made concerns the mode in which the structure fails, for this will greatly effect the resulting hydrograph. Earth fill structures such as Conklingviell Dam generally fail progressively, failure starting from an initial breach which increases in size by erosion of material under influence of the current. This mode of failure produces a hydrograph with an initially low discharge, increasing with time to a maximum, then decreasing as the reservoir elevation drops. The quantitative determination of this type hydrograph depends on the assumption of size and location of the initial breach, and rate of erosion, both of which are subject to question.

The maximum discharge rate would be obtained if failure were assumed to be instantaneous and complete and, for a discrete

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19.

volume of water, produces a hydrograph of shortest duration, maximizing the flow concentration. This is the most conservative mode of failure that can be assumed, and was used for this study.

DAM BREAK HYDROGRAPH

The physical laws governing unsteady flow in natural channels caused by a dam failure are among the most complex in the field of hydraulics. The first attempt to solve the problem was carried out by Saint-Venant who gave two differential equations bearing his name. These are based on a series of hypotheses which render them applicable only to gradually unsteady flow. While no integration of the equations is possible in the general case, certain simplifications and additional hypotheses have been used which have allowed integration and furnished solutions of limited applicability. Contributions based on theory and experiments have been made by many researchers including Ritter[15] Schocklitsch [16], Re[17], Pohle[18], Levin[19], Dressler[20-21], Stoker[22], Snyder[24], and U.S. Army Engineers[25].

Essentially, the sudden destruction of a dam results in a positive wave, advancing in downstream direction in the river channel, and a negative wave propagating in upstream direction into the storage reservoir. The wave velocity and profile depend on many factors including height of dam, channel and reservoir cross-section, channel resistance initial stream flow conditions, and length of storage reservoir. The simplifications commonly adopted by most researchers with a view to reducing the mathematical complexity of the problem included consideration of a prismatic, rectangular channel, horizontal channel bottom, and negligible flow resistance. The analytical methods which have been developed based on these simplifications have been confirmed by model studies, and as such can now be used as an engineering tool for determining flow conditions generated a sudden dam failure.

The objective of the Conklingville Dam failure study, as related to determination of the maximum possible stage at Stillwater, was to calculate a dam-break hydrograph to be used in flood routing. The hydropraph was determined by two independent approaches.

The first approach is essentially based on Stoker's method[22] which is the outgrowth of most of the theoretical and analytical work carried out to date. According to this method, the water depth in a rectangular channel at the dam site is 4/9 of the head in the reservoir until the negative wave reaches the end of the reservoir. To apply this method to they hypothetical failure of Conklingville Dam, the channel cross section at the Dam was approximated by three rectangles with a total area equaling that of the actual section. The downstream depth of flow was taken equal to that determined for the total flow at any time. The total releases were determined by the summation of the flows from each rectangle under a given head in the reservoir. The water surface in the reservoir was considered horizontal at any time and the drops in water level ranged between 0.5 ft and 5 ft, depending on stage. By plotting the calculated releases versus time, the Dam-break hydrograph was obtained. The results have been closely checked by the more recent work of the U.S. Army Corps of Engineers[24].

The second approach was developed as an independent check of the previously discussed method and was aimed at determining a conservative upper limit for the releases after hypothetical dam failure. It is assumed that flows are controlled only by the reservoir stage and channel characteristics and no energy is expended for negative wave motion. Critical depth is assumed to prevail at the dam section throughout the period under consideration and for all the flows after dam break. Essentially, this would mean that the channel bottom at the control section forms a broad-crested weir over which the water from the reservoir must flow. This would not be inconsistent with the relative steepness and geometry of the Sacandaga River channel below the dam. This assumption is the most conservative, since for a given head, the maximum flow is always released at critical depth. In determining the releases, a minor adjustment was made for head losses due to a change in channel cross section upstream of the dam. The results were again based on the assumption that the water surface in the reservoir is horizontal at any time. By plotting the calculated releases versus time, the dam-break hydrograph was obtained.

The hydrographs obtained with the two independent approaches described above are as follows:

	Stoker's Method		Critical Depth Method	
Time, Hr	Discharge, Cfs x 10^3	Total Outflow, Cu ft x 10^9	Discharge, Cfs x 10^3	Total Outflow, Cu ft x 10^9
0	990	0	1,410	0
5	780	14.5	980	20.0
10	616	25.9	690	33.3
15	482	34.3	479	42.1
20	380	40.8	336	48.2
25	300	46.0	215	52.0

As anticipated, the critical depth method, which was used as an upper limit verification for Stoker's method, leads to higher flows, with the peak discharge 42 percent greater, and the volume released in the first 25 hours 11 percent greater.

ROUTING OF DAM BREAK WAVE

The downstream movement of the dam break wave is described as unsteady flow governed by the principles of conservation of energy and conservation of matter. Continuity and dynamic equations which mathematically describe the phenomenon of unsteady flow were first presented in the 19th century by Saint-Venant and are found in most texts dealing with unsteady flow. The equations have been verified by observations and experiments. However, owing to their mathematical complexity, an exact integration of the equations is almost impossible. Solutions therefore must be made by methods based on simplifying assumptions and approximate step methods.

The stream channels of the Sacandaga River and Hudson River between Conklingville Dam and Stillwater are of widely varying characteristics. The river has many sharp bends, man-made and natural constrictions and abrupt drops, all of which made the use of wave theory impracticable. The method used was the same as described in the section on routing the maximum probable flood. This method neglects the energy relationship and is based on the conservation of matter and, in effect, is a storage routing procedure. Because the energy relationship is neglected , the results obtained by using this method are very approximate for locations a short distance downstream of a breached dam. However as the reach length increases, the storage relationship becomes the more predominate factor in wave attenuation and results are of greater accuracy. Stillwater is approximately 60 miles downstream of the Conklingville Dam and it is believed that the storage routing procedure produces results within the accuracy of available physical data.

In routing, no advantage was taken of the storage available in the Sacandaga River. This reach is about six miles long, with a relatively narrow channel, without flood plains, containing a very small amount of valley storage, and it was conservatively assumed that the dam outflow hydrograph would be transposed to the confluence of the Sacandaga and Hudson Rivers undiminished.

The stage-storage-discharge relationships for the reaches downstream of the confluence were determined as described under the probable maximum flood routing section with the exception of the reach above Palmer Falls, which includes the mouth of the Sacandaga. It was recognized that a large flow from the

Sacandaga would divide when it reached the Hudson River and flow

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would be in the upstream as well as the downstream direction. This storage upstream of the confluence was calculated from the stage at the confluence determined by steady state backwater calculations from Palmer Falls, assuming a horizontal water surface upstream of the confluence. An adjustment in this volume was made by subtracting the volume occupied by the flow in the Hudson River at the time the dam break wave arrives.

It was assumed that all the dams downstream of the mouth of the Sacandaga River would pass the Conklingville Dam break wave without failure. It is realized that for some of the dams this assumption is not valid. However the combined volume impounded by all the dams on the Hudson River above Stillwater is about 1 billion cubic feet. This is about 2 percent of the total volume in the Sacandaga Reservoir at the time of the hypothetical failure and considerably less than the difference in the two dam break hydrographs discussed in the previous section. To include this volume in the computations would greatly increase the compexity of the solution without increasing the validity of the results.

The dam break hydrographs determined by both methods were routed to Stillwater using a time increment of 20 minutes. Tributary inflow and river bank runoff from the probable maximum precipitation were combined with the dam break wave in the proper time sequence as the wave was routed.

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23.

DISCUSSION OF RESULTS

PROBABLE MAXIMUM FLOOD

The probable maximum flood at Stillwater reaches a peak discharge of 300,000 cfs and a river stage elevation of 110 ft approximately 64 hours after the start of the precipitation. The maximum 24 hour mean discharge is 260,000 cfs. Figure 3 shows the hydrograph of the flood at the site.

At the existing Bakers Falls Dam the probable maximum flood has a peak discharge of 230,000 cfs and a maximum headwater elevation of 221 ft.

It has been said that no method has been devised which can accurately indicate the frequency of large floods in the absence of stream flow records over long periods [25]. If the large flood being considered is several times larger than any observed evetn, as is the case for the predicted probable maximum flood of this study, the above statement can hardly be questioned. In fact, the probable maximum flood by definition has a frequency which is extremely large. However, in an attempt to bring some degree of perspective to the question of flood probability on the Hudson River at Stillwater, discharge-frequency curves based on a statistical evaluation of the available date were prepared, as shown on Figure 5, and extended to 10,000 years.

The data used for plotting the curves were based on USGS flow records at the Mechanicville gage from 1911 to 1956. Maximum daily discharge at Stillwater was obtained by correcting the Mechanicville flow for discharge from the Hoosic River and, prior tp 1930, for discharge from the Sacandaga River which would have been impounded by the Conklingville Dam.

the two curves, plotted on extreme value paper, are based on two of the numerous methods which have been suggested for estimating discharge-frequency relationships. These curves are thought to define the extremes of these relationships for the Hudson River at Stillwater. Curve A, based on a Type I extreme value distribution as suggested by Gumbel, represents an unsymmetrical data distrubution with a fixed skew. When the data are plotted, the flood of record falls considerably above the curve. Curve B is based on a graphical fit of the data distributed according to a method used by the U.S. Geological Survey[26]. The points plotted are for the latter distribution, but they are located approximately in the same position for both methods. The graphical fit shown by Curve B assumes a difference in the distribution parameters for the four floods with return periods exceeding 10 years. It has been suggested that outstanding

floods may in fact follow some law of their own which comes into operation at very long interval.,[25].

From Figure 5, the estimated discharge for a flood with a frequency of 10,000 years is as follows:

	Discharge – Cfs	
	Mean Daily	Peak=Mean Daily x 1.15

Curve A	121,000	139,000
Curve B	232,000	267,000

The factor of 1.15 used to determine the peak discharge from the mean daily, is the ratio of these discharges found for the probable maximum flood.

Based on the above analysis, the peak discharge for a flood with a return period of 10,000 years would fall between 139,000 and 267,000 cu. ft. per second. These results indicate that the maximum flood predicted has a return period well in excess of 10,000 years and meets the requirements of obtaining a flood that has a change of occurance approaching zero.

It should be noted that the predicted maximum flood produces stages along the river which would inundate large areas, including many existing communities. However, it is recognized by most experts that all work can not be economically protected against such remote occurance and lesser floods are normally used for most design purposes.

Often the U.S. Army Corps of Engineers use a lesser flood designated as the Standard Project Flood for design. This flood represents critical concentration of runoff from the most severe combination of precipitation that is considered “reasonably characteristic” of the drainage basin involved. The Standard Project Flood Peak discharge and volume is usually equal to 40 percent to 60 percent of the probable maximum flood for the same drainage basin[1]. There are some other design floods criteria used, which consider the degree of risk, hazzards involved and consequences of failure. The use of probable maximum flood as a design flood is not always justified or warranted for all projects and the design flood should be selected only after a complete study of all the factors involved.

DAM BREAK WAVE

The decision to assume that Conklingville Dam would fail in determining the effect of a dam break wave at Stillwater was based only on determining a hypothetical stage Stillwater. We believe that the probability of a

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failure of Conklingville Dam is extremely remote. The probable maximum flood study clearly indicates the reservoir has sufficient storage and the dam ample freeboard to pass this flood safely without danger of overtopping. The dam has successfully withstood a significant earthquake without damage. However, because a hypothetical failure of Conklingville Dam would cause the highest possible stage at Stillwater this possibility was included in the study.

The determination of the maximum river stage at a point almost 60 miles downstream from a hypothetical dam failure is, at best, a rough estimate, greatly dependent on a large number of assumptions for solution. A conscientious effort was made to make all assumptions as reasonably conservative as possible. Two different approaches were used in determining the initial dam break hydrograph. Stoker's method is the most reasonable based on the present state of knowledge. The critical depth approach was used only as an upper limit verification, since it is admittedly ultraconservative. By routing the dam-break hydrographs obtained with the two approaches to Stillwater the following results were obtained:

	Stoker's Method	Critical-Depth Method
Max Discharge at Conklingville	990,000 cfs	1,410,000 cfs
Max Discharge at Stillwater	670,000 cfs	810,000 cfs
Max Stage at Stillwater	El. 124	El. 128

From the above tabulation it is apparent that in terms of maximum state at Stillwater the results obtained with the two independent approaches are reasonably close, In our opinion, this confirms the validity of Stoker's method which is itself based on many conservative assumptions. As pointed out before, the results obtained with this method were closely checked with a more recent work of the U.S. Army Corps of Engineers. Therefore, the maximum possible river stage at Stillwater resulting from the failure of Conklingville Dam coincident with the maximum flood is El. 124.

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REFERENCE LIST

1. Definition of Terms, "Survey Conducted by the Committee on Failure and Accidents to Large Dams Other Than in Connection with the Foundations," United States Committee on Large Dams, Feb. 1968

2. Hydrometerological Report No. 33 "Seasonal Variation of the Probable Maximum Precipitation East of the 105th Meridian for Areas from 10 to 1,000 Square Miles and Durations of 6, 12, 24 and 48 Hours" U.S. Department of Commence, Weather

Bureau, U.S. Government Printing Office, 1956

3. "Report on Project Flood for Review of Reports on Flood Controls for the Connectict River Basin" Unpublished Report, U.S. Army Corps of Engineers, Providence District, August 1946

4. Gilman, C. S., Chief Hydrometeorological Section, U.S. Weather Bureau, Memorandum to Mr. A. L. Cochran, Civil Works, Office of Chief of Engineers, "Preliminary Estimate of Probable Maximum Precipitation for the Delaware River at Tocks Island Dam Site," July 3, 1956

5. Hydrometeorological Report No. 28, "Generalized Estimate of Maximum Possible Precipitation Over New England and New York," U.S. Department of Commerce, Weather Bureau, U.S. Government Printing Office, 1952

6. Hydrometeorological Report No. 40, "Probable Maximum Precipitation – Susquehanna River Drainage Above Harrisburg, Pa.", U.S. Department of Commerce, Weather Bureau, U.S. Government Printing Office, 1965

7. "Unit Hydrographs, Part I – Principles and Determinations," Civil Works Investigation, Project 152, U.S. Army Engineer District, Baltimore: Corps of Engineers, Baltimore, Md., 1963

8. "Hydrology Guide for Use in Watershed Planning," National Engineering Handbook, Sec. 4, Supplement A, U.S. Department of Agriculture, Soil Conservation Service Central Technical Unit, Beltsville, Md., 1957

9. Newton, D. W., and Vinyard, J. W., "Computer-Determined Unit Hydrograph from Floods," Proceeding ASCE, Journal of the Hydraulic Division, Vol. 93, Hy-5, September 1967

10. Taylor, A. B., and Schwartz, H. E. , "Unit-Hydrograph Lag and Peak Flow Related to Basin Characteristics," Trans. Amer. Geophysical Union, Vol. 33, No. 2, April 1952

11. "Design of Small Dams," U.S. Department of Interior, Bureau of Reclamation, U.S. Government Printing Office, Washington, D.C., 1965

B-29

2

12. "Routing of Floods Through River Channels" U.S. Army Corps of Engineers, Engineering and Design Manuals EM 1110-2-1408, U.S. Government Printing Office, Washington, D.C., March

QUIRK, LAWLER & MATUSKY ENGINEERS

1960

13. Chow, V. T., "Open Channel Hydraulics," McGraw-Hill Book Company, New York 1959

14. Barnes, H. H., Jr., "Roughness Characteristics of Natural Channels," Geological Survey Water Supply Paper 1849, U.S. Government Printing Office, Washington, D.C., 1967

15. Ritter, A., "Die Fortpflanzung der Wasserwellen," (Propogation of Waves) , Zeitschrift des Verelines deutscher Ingenieure, Vol. 36, No. 33, pp. 947-954, Aug. 13, 1892

16. Shocklitsch, A., "Uber Dambruchwellen," (On Waves Produced by Broken Dams), Stilzungsberichte, Mathematisch-naturwess-enschaftliche Klasse, Akademie der Wissenschaften in Wien, Vol. 126, Part IIa, pp. 1489-1514, Vienna, 1917

17. Re., R., "Etude de Lacher Instantane d'une Retenue d'eau Dans un Canal par La methods Graphique," (Study of the Instantaneous Release of Water in a Reservoir to a Canal by the Graphic method), La Houille Blanche, 1st year, No. 3, pp. 181-187, Grenoble, May 1946

18. Pohle, F. V., "Motion of Wave Due to Breaking of a Dam and Related Problems," Paper No. 8, in Symposium on Gravity Waves, U.S. National Bureau of Standards, Circular 521, pp. 47-53, 1952

19. Levin, L., "Mouvement Non Permanent Sur Le Cors d'eau a la Suite de Ruptrue de Barrage' (Unsteady Flow in Channels Following the Rupture of Dam), Revue Generale de l'Hydraulique, Vol. 18, No. 72 pp. 293-315, Paris, Dec. 1952

20. Dressler, R. F., "Hydraulic Resistance Effect Upon the Dam-Break Functions," Paper 2356, Journal of Research, U.s. National Bureau of Standards, Vol. 49, No. 3, pp. 217-225, Sept. 1952

21. Dressler, R. F., "Comparison of Theories and Experiements for the Hydraulic Dam-break Wave," Assemblee Generale de Rome, 1954, International Association of Scientific Hydrology, Publication No. 38, Vol. 3, pp. 319-328, 1954

22. Stoker, J. J., "Water Waves, " Vol. IV of "Pure and Applied Mathematics," Interscience Publishers, Inc. New York, 1957

23. Snyder, F. F. "Hydrology of Spillway Design; Large Structures Adequate Data" Proceedings of ASCE, Journal of the Hydraulics Division, Vol. 90, HY-3, May 1964

QUIRK, LAWLER & MATUSKY ENGINEERS

24. "Floods Resulting from Suddenly Breached Dams" Miscellaneous Paper 2-374, Conditions of Minimum Resistance Report No. 1, February 1960, Condition of High Resistance Report No. 2, November 1961, U.S. Army Engineers Waterways Experiment Station, Viksburg, Mississippi

25. "Review of Flood Frequency Methods," Final Report of the Subcommittee of the Joint Division Committee on Floods, Transactons, ASCE vol. 118, pp. 1220-1230, 1953

26. Riggs, H. C., Frequency Curves, Chap. A2, "Techniques of Water Resources Investigations of the United States Geological Survey," U.S. Department of the Interior, U.S. Government Printing Office, Washington, D.C. 1968

B-31

OTHER SOURCES

Chow, V. T., "Handbook of Applied Hydrology," McGraw-Hill Book Company, New York, 1964

Lensley, R. K., Jr., Kohler, M.A., and Paulkus, J.C. H. "Applied Hydrology," McGraw-Hill Book Company, New York, 1949

Davis, C.V., "Handbook of Applied Hydraulics,"McGraw-Hill Book Company, New York, 1952

"Backwater Curves in River Channels," U.S. Army Corps of Engineers, Engineering and Design Manual EM 1110-2-1409, U.S. Government Printing Office, Washington, D.C., December 1952, Change 1, September 1960

Topographic Maps – Entire Basin, 7.5 Min and 15 Min Quadrangles and 1:250,000 Scale, U.S. Department of Interior Geological Survey, Topographic Division, Washington, D.C.

Topographic Maps – Hudson River, Fort Edward Dam to Palmer Falls Dam, Scale 1 In. = 200 ft, Niagara Mohawk Power Corporation

Topographic Maps – Hudson River, Palmer Falls Dam to Curtis Falls Dam, Scale 1 In. – 100 Ft, International Paper Company, 1966

Topographic Maps – Hudson River, Hadley to Mouth of the Schroon River, Scale 1 In. = 400 Ft, Niagara Mohawk Power Corporation, 1922

U. S. Lake Survey Chart No. 180, New York State Barge Canal System, U. S. Army Corps of Engineers, Detroit, Michigan,

QUIRK, LAWLER & MATUSKY ENGINEERS

1964

River Soundings, Fathometer Readings, Hudson River, Troy Locks to Corinth Bridge, U.S. Geological Survey, Albany, New York, 1967

Field Cross Sections and River Soundings, Clarkeson and Clough Associates, Albany, New York, 1968

Precipitation Records of Stations Within and Adjacent to Basin, National Weather Records Center, Asheville, North Carolina

Water Supply Papers, U.S. Department of Interior Geological Survey, Water Resources Division, U. S. Government Printing Office, Washington, D.C.

Unpublished Stage Recordings and Rating Tables, U.S. Department of Interior, Geological Survey, Water Resources Division, Albany, New York

B-32

2

Unpublished Gage Readings, Champlain Barge Canal, Lock 3c to Lock 7C, New York State Barge Canal, Canal Engineer, Waterford, New York

"Report on the Water Power and Storage Possibilities of the Hudson River" New York Water Power Commission, State of New York, 1922

U. S. Army Corps of Engineer "308 Report," Hudson River Basin, House Document No. 149, 72nd Congress, 1st Session, December 1931

Plans and data on the following dams were obtained from the listed owners:

Dam	Owner
Indian Lake Dam	Indian River Company
Conklingville Dam	Board of Hudson RIver-Black River Regulating District
Stewarts Bridge Dam	Niagara Mohawk Power Corporation
Curtis Falls Dam	International Paper Company

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Palmer Falls Dam	International Paper Company
Spiers Falls Dam	Niagara Mohawk Power Corporation
Sherman Island Dam	Niagara Mohawk Power Corporation
Feeder Dam	State of New York, Moreau Manufacturing Corporation
Glens Falls Dam	Finch, Pruyn & Company, Inc., Niagara Mohawk Power Corporation
Bakers Falls Dam	Niagara Mohawk Power Corporation
Crockers Reef	State of New York

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Dam	Owner
Fort Miller Dam	Fort Miller Pulp & Paper Company
Thomson Dam	United paperboard Corporation, Thomson Paper Company
Stillwater Dam	Niagara Mohawk Power Corporaiton
Upper Mechanicville Dam	State of New York, West Virginia Pulp and Paper Company
Winnies Reef	Niagara Mohawk Power Corporation

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APPENDIX C

NOTATIONS & SYMBOLS USED IN THE REPORT

A	=	Cross-sectional area
a	=	$0.5(1/v_2 - 1/v_1)$
A_n, A_n+1	=	Area bounded by isohyets number n and n+1
B	=	Channel width
C	=	Constant equal to 4 feet $^{1/2}$/second determined from data supplied
D	=	River depth
D'	=	$\frac{dy}{dx}$
dg/dy	=	Slope of discharge rating curve at a station whose cross-section is representative of the reach for steady flow
F_E	=	Vertical component of earth pressure acting on downstream face, i.e. weight of earth mass vertically above the downstream face, acting through the centroid of that mass.
F_H	=	Horizontal component of hydrostatic pressure, acting along a line H/3 feet above the base. $1/2yH^2$, where y = specific weight of water
F_u	=	Uplift force on base of dam, as determined by foundation seepage analysis, and integration of point pressure intensities over base area; if foundation is homogeneously permeable, pressure varies approximately linearly from full hydrostatic head at the heel to full tailwater head, and F_u is approximately 1/2yHB, acting at B/3 from the heel. This value is often multiplied by some fraction less than 1 if the foundation is relatively impermeable, but it is on the safe side to assume uplift over the entire base area.
F_v	=	Vertical component of hydrostatic pressure. Weight of fluid mass vertically above the upstream face, acting through the centroid of that mass.

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NOTATIONS & SYMBOLS USED IN THE REPORT (Continued)

H	=	Resultant of all forces acting on the dam, equal to R but in the opposite direction.
H	=	Total head (elevation of the energy line) above a horizontal datum. The mean sea level was selected in this study as the datum.
H'	=	Water head above the spillway crest in feet
h	=	Depth of impounding water
h_e	=	Eddy loss. For convenience of computation, this loss was considered part of the friction loss h_f.
h_f	=	Friction loss between two end sections i and i+1
h_o	=	Depth of water below the dam
I	=	Rate of inflow into reservoir
i	=	Hydraulic quadient. It is equivalent to the slope of the seepage curve dy/dx.
I_1, I_2	=	Inflow at upstream end of reach 1 and 2
K	=	Permeability coefficient (ft/sec)
k	=	Time of travel of flood wave and also the change of storage per unit change of discharge
L	=	Distance from dam
L′	=	Length of Spillway or weir
n	=	Manning's roughness coefficient
n, n+1	=	subscripts denoting successive intervals of time of length $t_{n+}1 - {}^{t}n$
O	=	Rate of outflow from the reservoir

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NOTATIONS & SYMBOLS USED IN THE REPORT (Continued)

$0_1{'}0_2$	=	Outflow from reach 1 and 2
$P_1{'}P_{n+1}$	=	Average depth of rainfall over the areas bounded by Isohytes 1 and 2
Q	=	River discharge
Q′	=	Flow resulting from dam failure at a distance L Downstream of the dam site
Q_o	=	Initial discharge after dam failure
$\mathcal{Q}$	=	Seepage (cfs/ft)
R	=	Resultant of foundation shear and bearing pressures: Horizontal component, $R_H = F_H + F_E$ acting along the base; Vertical component, $R_V = W + F_V – F_U$ acting at a distance X from the toe that can be determined by the requirement For rotational equilibrium of the dam, by equating to zero the sum of the moments of all the foregoing forces about the toe of the dam.
R_h	=	Hydraulic radius
S	=	Available storage above spillway level
S_f	=	Friction slope taken as the average of the slopes at the two end stations, i.e. $\int_f = ½ (S_i + S_{i+1})$
So	=	Slope of river bed
T	=	Time after dam failure
V	=	River velocity and is equal to Q/A
V_1	=	Velocity of wave front
V_2	=	Velocity of wave tail
Vn’Vn+1	=	Values of isohyets number n and n+1. These numbers refer to the lettered isohyets shown in Figure III-8
V_W	=	Rate of movement of flood wave.
W	=	Weight of dam – (area of cross-section of dam) (Sy) where S = specific gravity of masonry, approximately 2.4 or 2.5, acting through centroid of cross-section

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NOTATIONS & SYMBOLS USED IN THE REPORT (Continued)

W_O	=	Amount of water stored in reservoir
X	=	A dimensionless constant representing an index of the wedge storage in a routing reach
x	=	Distance measured from dam site
Y	=	Water surface elevation above mean sea level
Y'	=	Depth at the crest of the wave
y	=	Depth of seepage curve
Z_1	=	Water surface elevation above mean sea level
$\theta°$	=	Angle between horizontal line and upstream face of the dam.
Δt	=	Length of the routing period having a maximum value of $2K_x$ and may be taken as equal to k. The routed hydrograph is relatively insensitive to the value bf Δt
α	=	Energy coefficient. This coefficient was assumed to be unity because of the fairly straight alignment and regular cross-section of the river between the Battery and Indian Point

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APPENDIX D

REFERENCES

1. U. S. Department of Commerce, Weather Bureau, Hydrometeorological Report No. 40, PROBABLE MAXIMUM PRECIPITATION, SUSQUEHANNA RIVER DRAINAGE BASIN ABOVE HARRISBURGH, PA., Washington, May 1965

2. U. S. Department of Commerce, Weather Bureau, Hydrometeorological Report No. 33, SEASONAL VARIATION OF THE PROBABLE MAXIMUM PRECIPITATION EAST OF THE 105TH MERIDIAN FOR AREAS FROM 10 TO 1,000 SQUARE MILES AND DURATION OF 6, 12, 24, AND 48 HOURS, U. S. Government Printing Office, 1956.

3. U. S. Army Corps of Engineers, Unpublished Report, REPORT ON PROJECT FLOOD FOR REVIEW OF REPORTS ON FLOOD CONTROLS FOR THE CONNECTICUT RIVER BASIN, Providence District, August, 1946.

QUIRK, LAWLER & MATUSKY ENGINEERS

4. Stone & Webster Engineering Corporation, FLOOD STUDY OF UPPER HUDSON RIVER BASIN, Boston, Mass., March 21, 1969.

5. U. S. Army Engineer District, Baltimore Corps of Engineers, Civil Works Investigations, Project 152, UNIT HYDROGRAPHS, PART I, PRINCIPLES AND DETERMINATIONS, Baltimore, Maryland, 1963.

6. U. S. Army Corps of Engineers, Engineering and Design Manuals EM 1110-2-1405, FLOOD HYDROGRAPH ANALYSIS AND COMPUTATIONS, U. S. Government Printing Office, Washington, D. C., August 31, 1959.

7. U. S. Army Corps of Engineers, Engineering and Design Manuals EM 1110-2-1408, ROUTING OF FLOODS THROUGH RIVER CHANNELS, U. S. Government Printing Office, Washington, D. C., March 1, 1960.

8. State of New York Hudson River Valley Commission, THE HUDSON WATER RESOURCES, New York, 1966

9. New England, New York Inter-Agency Committee, THE RESOURCES OF THE NEW ENGLAND-NEW YORK REGION, Part Two, Chapter XXXVII Subregion E (Hudson River Basin), New York, Vermont, Mass., 1955

10. Water Pollution Control Board, New York State Department of Health, LOWER HUDSON RIVER DRAINAGE BASIN SURVEY SERIES REPORT NO. 5, New York, 1953

D-1

11. Water Pollution Control Board, New York State Department of Health, LOWER HUDSON RIVER DRAINAGE BASIN SURVEY SERIES REPORT NO. 8, New York, 1960.

12. Summary Report of the Hudson River Valley Commission, THE HUDSON, New York, February 1, 1966.

13. Hudson River Valley Commission of New York, HUDSON RIVER ECOLOGY, New York, October, 1966.

14. Water Resources Commission, State of New York Conservation Department, Bulletin 61, THE HUDSON RIVER ESTUARY, A Preliminary Investigation of Flow and Water-Quality Characteristics, New York, 1967.

15. Quirk, Lawler & Matusky, HUDSON RIVER DISPERSION CHARACTERISTICS; Memo Report to Con Ed, October, 1965.

16. Several telephone conversations with Mr. Robert Forrest, Chief Engineer, Board of Hudson River-Black River Regulating District, March-April, 1970.

17. Several meetings and telephone conversations with Mr. Dwight E. Nunn, the Atomic Energy Commission, and Mr. P. Carpenter, FWPCA, March-April, 1970.

18. United States Geological Survey, FLOODS OF AUGUST-OCTOBER 1955, NEW ENGLAND TO NORTH CAROLINA, Water Supply Paper No. 1420.

19. Gilcrest, B. R., FLOOD ROUTING, Chapter X in Engineering Hydraulics, John Wiley & Sons Inc., New York, 1950.

20. Listvan, L.L., RASCOT MAKSIMALNOGO RASCHODA VODY OT PRORYVA NEKAPITALNYCH PLOTIN PRI PROJEKTIROVANII MOSTOVYCH PERECHODOR in Russian, or Computation of Flow after Dam Break for Design of Bridges, STPMS, Moscow, 1948.

21. Chow, Ven Te, OPEN-CHANNEL HYDRAULICS, McGraw-Hill Book Company, Inc., New York, 1959.

22. Data and Correspondence supplied by New York City Department of Water Resources concerning Ashokan Dam and Reservoir.

23. Morris, Henry M., APPLIED HYDRAULICS IN ENGINEERING, the Ronald Press Company, New York, 1963.

D-2

24. A meeting with Mr. Andrew Matusky of the U. S. Army Engineers District, Baltimore, February, 1970.

25. Prasad, Ramanand, A NUMERICAL METHOD OF COMPUTING GRADUALLY VARIED FLOW PROFILES, Presented to the Specialty Conference of the Hydraulics Division ASCE, at the Massachusetts Institute of Technology, Cambridge, Massachusetts, August 21, 1968.

26. Several telephone conversations between the author, Karim A. Abood of Q. L. & M. and Mr. Kenneth I. Darmer, Supervisory Hydrologist, Water Resources, Division, U.S.G.S., Albany, N.Y., March-April, 1970.

27. Baltzer, R. A. and Shen, J., FLOWS OF HOMOGENEOUS DENSITY IN TIDAL REACHES, U.S.G.S., Washington, September, 1961, Reprinted July, 1966.

28. Lai, Chintu, FLOWS OF HOMOGENEOUS DENSITY IN THE REACHES, SOLUTION BY THE METHOD OF CHARACTERISTICS, U.S.G.S., open file report, Washington, D.C., 1965.

29. Lai, Chintu, , FLOWS OF HOMOGENEOUS DENSITY IN TIDAL THE REACHES, SOLUTION BY THE IMPLICIT METHOD, U.S.G.S., open file report, Washington, D.C., 1965.

30. Lai, Chintu, , NUMERICAL SIMULATION OF WAVE-CREST MOVEMENT IN RIVERS AND ESTUARIES, Extract of "The Use of Analog and Digital Computers in Hydrologie," Symposium of Tucson, December, 1968.

QUIRK, LAWLER & MATUSKY ENGINEERS

31. Lai, Chintu, COMPUTATION OF TRANSIENT FLOWS IN RIVERS AND ESTUARIES BY THE MULTIPLE-REACH IMPLICIT METHOD, U.S.G.S. Prof. paper 575-B, 1967.

32. Lai, Chintu, COMPUTATION OF TRANSIENT FLOWS IN RIVERS AND ESTUARIES BY THE MULTIPLE-REACH METHOD OF CHARACTERISTICS, U.S.G.S. Prof. paper 575-D, 1967.

33. J. J. Dronkers, TIDAL COMPUTATIONS IN RIVERS AND COASTAL WATERS, Interscience publishers, Division of John Wiley and Sons, N. Y., 1964.

34. U.S.C. & G.S., TIDAL AND CURRENTS IN HUDSON RIVER, Special publication No. 180 by Paul Schureman, Washington, 1934.

D-3

35. Darmer, Kenneth I., HYDROLOGIC CHARACTERISTICS OF THE HUDSON RIVER ESTUARY, presented at the 2nd HRVC Hudson River Symposium, 1969.

36. Gofseyeff, S. and Panuzio, Frank L., HURRICANE STUDIES OF NEW YORK HARBOR, Journal of the Waterways and Harbors Division, proceedings of the ASCE, February 1962.

37. Wilson, B., THE PREDICTION OF HURRICANE STORM-TIDES IN NEW YORK BAY, Technical Memorandum No. 120, Beach Erosion Board Corps of Engineers, 1960.

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2.6 Meteorology

The description of site meteorology is given in Section 2.6.1 while a brief discussion of specific site meteorological and atmospheric diffusion studies are given in section 2.6.2. The technical reports pertaining to these site specific studies are also include. The safety analysis presented in section 2.6.3 and 2.6.4 is based on the site specific studies discussed in section 2.6.2. Section 2.6.5 provides a brief description of the onsite meteorological monitoring program.

The discussion of site meteorology given in section 2.6.1 is based on selected individual years in which analysis of meteorological data was performed. It should be pointed out that although the years selected are representative of the site meteorology, at least some year to year variability in the meteorological parameters will occur.

2.6.1 Site Meteorology

Winds

An important meteorological characteristic of the Indian Point Environment is that both northerly and southerly winds occur at maximum frequency. This is evident in all meteorological data collected at Indian Point from 1955 to the present.

Figures 2.6-1 a, b, and c present some constructed wind roses prepared using meteorollogical data collected during 1984 from the onsite 122 meter meteorological tower [1,2,3,4]. These wind roses provide an example of typical wind direction and frequency distributions that occur at Indian Point, on a quarterly basis for the 10 meter, 60 meter, and 122 meter levels of the tower. These wind roses show clearly the bidirectional frequencies in the wind directions, with frequency maximas in the north and south direction.

A comparison of the 10 meter level wind roses between each of the four quarterly periods during 1984 (Figure 2.6-1a) shows that north winds had the highest frequency during the period January-March, while northeast winds dominated during the remaining three quarterly periods. The period July-September had the highest frequency of northeast and south winds. South winds occurred with the lowest frequency during the period January-March.

At both the 60 meter and 122 meter levels (Figures 2.6-1b, 2.6-1c), a distinct peak in frequency of north winds occurred for all four quarterly periods. The 60 meter level, like the 10 meter level also displayed a peak in frequency of northeast winds particularly during the July-September period. This peak in northeast winds was not nearly as pronounced at the 122 meter level. The frequency of south and southeast winds was lowest during the period January-March and more pronounced during the remaining three quarterly periods. These figures also indicate a smaller third peak in the frequency of northwest winds which was most pronounced during the January-March period at all three tower levels. The relatively low frequency of south winds and the third peak in the frequency of northwest winds is likely to be the result of the stronger large scale (gradient) winds during the January-March period.

These wind characteristics for 1984 are generally consistent with wind observations collected during other years, with the most significant feature being the tendency for air flow along the axis of the valley. Differences in wind distributions that do occur between years can be attributed to year to year variability in the strength and movement of synoptic scale weather systems (cyclones and anticyclones).

The 1984 data, as well as analysis of meteorological data from other years (see section 2.6.2), suggests that winds in the region are controlled primarily by topography. It appears that both terrain channeling and a thermally driven valley wind is contributing to the observed wind direction frequency distribution.

Terrain channeling occurs when surface air flow at some angle to the valley, is deflected by the elevated valley walls and forced to flow along the valley axis. Terrain channeling is dependent only on the orientation of the valley, and the strength and direction of large scale winds. The thermally driven valley winds are induced by differential heating between one region of the valley and an adjacent region with different topography. The differential heating induces an along valley pressure gradient which drives the up or down-valley wind. Up-valley winds are confined during the daytime when surface heating is occurring while down-valley winds are primarily nocturnal, when there is significant surface radiative cooling. Consequently, up-valley winds will occur during hours with unstable stability classes while down-valley winds are characteristic of hours when low level inversions are occurring and stability classes are stable. These up and down-valley winds are most prevalent during the summer and fall season when the large scale (gradient) winds are weakest. Under these conditions it is common to observe north or northeast winds during the night and early morning at Indian Point, with a shift to southerly winds occurring within a few hours of sunrise, when surface heating commences. Thus, diurnal variations in winds at Indian Point will have the highest frequency of occurrence during the summer and fall season.

The diurnal variation of the vector mean wind as measured 70-feet above river during September-October 1955 is shown in Figure 2.6-2 for conditions in which the large scale flow was virtually zero (12 days) and in Figure 2.6-3 for conditions in which the large scale flow (gradient 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 on the order of 6 MPH.

A measure of the reliability of the diurnal shift in wind direction is shown in figure 2.6-4 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 (an extraordinarily high value for meteorological wind systems in this latitude), the reliability of a given wind direction is very high, It may be seen that the down-valley nocturnal flow is extremely reliable from 20-08 hours while the up-valley flow is as reliable from about 14-16 hours under zero pressure gradient conditions. For weak pressure gradient conditions the nocturnal flow direction is very probable from 24 to 08 hours and thereafter becomes quite unreliable. In short, these data indicate that a consecutive 24 hours down-valley flow with light wind speeds and inversion conditions is extremely improbable.

<u>Atmosphere Stability</u>

Tables 2.6-1, 2.6-2 and 2.6-3 provide the wind direction percent frequency distribution as a function of stability at the 10 meter level of the 122 meter meteorological tower.(5) The Pasquill stability classes are based on vertical temperature gradients (0C/100 meters) and are the same as the NRC classification of atmospheric stability (6). Table 2.6-1 shows the joint frequency distribution for a one year period while Table 2.6-2 and 2.6-3 give distributions for the summer season (May – October) and winter season (November – April), respectively.

Inspection of tables 2.6-1, 2.6-2 and 2.6-3 show that stability Class E occurred with the highest frequency for all wind directions (total) both for the one year period and for the summer and winter seasons. For the one year period it occurred 37.17% of the time. Similar percent frequencies are shown for the summer and winter seasons. This stability class occurred most frequently during south southwest winds with a second peak in frequency occurring for northeast winds. The total percent frequencies show stability Class D occurs with the second highest frequency while Class G had the lowest frequency occurring only 1.69% of the time for the one year period. Again similar percent frequencies are indicated for the summer and winter seasons.

Joint Frequency Distributions

Tables 2.6-4 (sheets 1 through 28) provides recent joint frequencies of wind direction, wind speed and atmospheric stability for the quarterly periods in 1986. Sixteen wind directions, seven wind speed categories including calm winds and seven Pasquill stability classes (A-G) are used. The stability classes are determined from 61-10 meter vertical temperature difference (delta-T). Data recovery during 1986 was 99 percent (13 missing hours) during the April-June period and 100 percent for the remaining quarterly periods.

Thunderstorms

Thunderstorms, although not unique to the Indian Point Site, are important since they can produce wind and precipitation patterns in the Indian Point environment that have considerable spatial and temporal variability. An important characteristic of thunderstorms is a downdraft of relatively cold air which spreads radially outward at the earth's surface. This cold air outflow, commonly called a gust front, can at times travel significant distances from the immediate storm environment. A typical gust front will appear as a sharp change in wind speed and direction and a drop in ambient air temperature.

Figure 2.6-5 shows the mean annual distribution of days with thunderstorms for the northeast United States. (7) This map is based on data from the period 1952-1962. Figure 2.6-5 shows that in the vicinity of Indian Point an average of between 20 and 30 days per year will have thunderstorms. Most of these thunderstorm days will occur during the summer season.

2.6.2 Meteorological and Atmospheric Diffusion Studies at Indian Point

New York University under a contract with Consolidated Edison Company made extensive tests on the meteorological conditions at the Indian Pont site. The testing program started in 1955 and was completed in 1957. Site meteorology (wind direction, wind speed and vertical temperature gradient) and atmospheric diffusion characteristics as determined from this testing program are described in three technical reports prepared by the New York University staff under the immediate direction of Professor Benjamin Davidson. The original New York University reports, or applicable excerpts there from, which were reviewed by Professor Davidson and the Consolidated Edison staff, are provided on pages Q1-Q44 and R1-11. In addition, information on precipitation, the prevalent wind directions associated with precipitation, a table of wind directions during thunderstorms and associated wind roses are given on pages R12-R20.

Due to questions concerning the relevancy of certain meteorological data obtained in the 1956-1957 period a new meteorological monitoring program in the Hudson River Valley was initiated to try to verify the results of the old study. The locations of the meteorological towers erected for the new program did not correspond to the locations of the towers used in the earlier program, and the data were not reliable, due to instrumentation difficulties. The two sets of data were, therefore, difficult to compare although it was evident that no substantial change occurred in the valley meteorology from 1956-1969.

The experimental program was reorganized in the fall of 1969, and a new meteorological tower site was selected as close to the original 1956 one as was possible under current conditions. Wind observations were made at this 100-foot tower at Indian Point and at a ship anchored in the Hudson River northwest of Indian Point. The results of the program for the period 26 November 1969 to 1 October 1970 are presented in Dr. Halitsky's report NYU-TR-71-3 (see pages Y-1 to Y-32).

The conclusions, as stated in the report, are:

1) Annual average statistics of wind speed direction and vertical temperature differences were substantially the same for 1956 and 1970.

2) Average wind hodographs, as the ships exhibited the same diurnal reversal pattern and the same 2.5 m/sec nighttime down-valley speed in both years. The average wind hodograph at the tower showed a similar pattern of reversal, but the nighttime down-valley speed was about 2 m/sec.

3) All sixteen daily wind hodographs used for calculating the average hodograph at the tower showed the diurnal reversal and exhibited considerable variability in speed and direction from day to day through a complete cycle.

4) Maximum persistencies of low-speed inversion winds in the critical 005-020 sector were 2 hours, 4 hours and 3 hours for 1, 1.5 and 2 m/sec speeds, respectively, during the entire ten-month data record.

Additional data acquired from 1 January 1970 to 31 December 1971 is presented in NYU-TR-73-1 (see pages Z-1 to Z-82).

In addition to these meteorological studies, several diffusion studies pertaining to atmospheric diffusion modeling applied to the Indian Point Site were conducted. The final reports pertaining to these diffusion studies are given on pages 2.6.K-1 to 2.6.K-15, 2.6.L-1 to 2.6.L-67 and 2.6.M1 to 2.6.M-11.

2.6.3 Application of the Site Meteorology to the Safety Analysis of the 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 which 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 directions was five days. The winds in this case were from the northwest and speeds ranged from 15 to 30 MPH. In view of the large wind speeds and slightly unstable to adiabatic range of thermal stability associated with this period of maximum wind direction duration, 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 010-030o, is also the direction of maximum wind frequency at the 100 ft. tower level. Because of the relatively high frequency of inversion conditions associated with this wind direction, the safety analysis assumed 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 005-020o wind direction. The valley wind is diurnal in nature, i.e., up-valley during unstable hours and down-valley during stable hours.

The safety analysis of the Loss-of-Coolant Accident assumed that the accident occurs during down-valley inversion flow conditions and that these conditions persisted for 24 hours with average wind speeds slightly less than 2 m/sec. Figures 2.6-2 and 2.6-3 indicate that the duration of the down-valley flow is about 12 hours rather than 24 hours, and that the vector mean wind speeds are on the order of 2.5 m/sec.

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

The remainder of the safety analysis assumed that for the next 30 days, 35% of the winds are in the 20o sector corresponding to the nocturnal down-valley flow and that wind speed and thermal stability were as observed over the period of one year as measured at the 100 ft. tower location. If the observations were distributed uniformly throughout the year, slightly over 100 hours per month of 005-020o winds could be expected to occur. The analysis assumes 276 hours of 005-020o 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 hours per month. From Figure 2.6-4, the hours during which the down-valley flow is quite reliable under weak pressure gradient conditions are from 00-08 hours. Assuming that the reliability is 1.0 during these hours (it is 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 hours per month. This indicates that the meteorology assumed in the safety analysis beyond the first 24 hours is about right for the worst months (September and October) and is undoubtedly conservative, with varying degrees of conservatism, for about ten months of the year.

The inversion frequency assumed for the 30-day accident case is conservative because the evaluation was made from joint assumptions concerning the postulated meteorological conditions viz.,

1) Inversion conditions prevail for 42.4% of the time
2) The wind direction is within a narrow 20o sector, for 35% of the time

This is equivalent to assuming that in the model 20o sector, the inversion frequency is 14.8 percent for the 30-day period. The observed annual maximum inversion frequency for a 20o sector is 6.2% (p.29, Table 3-3, NYU Tech. Report 372.3, Section 1.6). If we assume that the inversion frequency is spread uniformly throughout the year, almost three months worth of inversion in the model 20o sector are considered to occur in the first 31-day month after the accident. The assumptions of uniform spread of inversion frequency over the year are examined above, where an attempt was made to isolate those local meteorological conditions at Indian Point which might yield the highest 30-day dose. It is concluded that the “worst” meteorological conditions are associated with the nocturnal down-valley flow which is most frequent during September and October.

2.6.4 Conservatism of Indian Point Site Meteorology with Respect to Calculation of Off-Site Doses

The conservatism of the site meteorology was evaluated with respect to wake dilution factors, Pasquill categories for stability classification and site shaping characteristics.

Building wake dilution factors, documented in reports by Dr. Halitsky titled “An Analysis of the Con Edison and AEC-DRL Wake Diffusion Models as Applied to the Indian Point Site”, (see pages 2.6.K-1 to 2.6-K-15) and “An Analysis of the Con Edison and AEC-DRL Accident meteorology models as Applied to the Indian Point Site”, (see pages 2.6.L-1 to 2.6.L-67) demonstrate that limiting the building wake dispersion correction factor to a value of 3, as required by AEC Safety Guide No. 4, is overly conservative. Both the Con Edison wake model and the Safety Guide model, without limiting the building wake dispersion correction factor, are realistically conservative when compared to actual field and wind tunnel measurements. The reports also evaluate the overall conservatism of the Con Edison accident diffusion model. Specific investigations of the turbulence characteristics and wind persistence for the site are presented.

In addition, these two reports show that the classification of atmospheric stability using the criteria documented in Safety Guide No. 23 is not appropriate for the Indian Point site. The significance of the valley influence in generating lateral dispersion, and meandering of the wind, create horizontal standard deviations of greater magnitude than those determined by using vertical temperature gradients. The data indicate Pasquill categories measured under inversion conditions with horizontal wind fluctuations similar to a Pasquill D category while, vertically, Pasquill categories are E, F, or G.

Pickard, Lowe and Garrick of Washington D.C., in the report, “A Study of Atmospheric Diffusion Condition Probabilities using the Composite Year of Indian Point Site Weather Data” (see pages 2.6.M-1 to 2.6.M-11), illustrate the effects of the site shaping technique for estimating the 95% confidence level of the annual average dispersion coefficient at the exclusion area envelope. In addition, the report shows the effect of using the “split sigma” model to account for the lateral wind meander observed in the valley.

The composite year of measured meteorological data was compiled in a form compatible with AEC Safety Guide No 23 in sheets 8 to 14 of Table 2.6-5. In order to conform with the sensor heights specified in Safety Guide No. 23 the measured ΔT was multiplied by a ΔT correction factor. The method used to determine this factor assumes an exponential relationship between temperature and height, such that measured temperature difference between any two heights can be represented as a temperature difference between two other heights, according to the following relationship:

$$\Delta T \text{ correction factor} = 1n\,(h_{ue}/h_{le})/\,1n\,({}^{h}um/\,{}^{h}lm)$$

where:

h_{ue} =	height of upper extrapolated temperature (ft)
h_{le} =	height of lower extrapolated temperature (ft)
h_{um} =	height of upper measured temperature (ft)

h_{lm} = height of lower measured temperature (ft)

Sheets 1 to 7 of Table 2.6-5 show data normalized to the sensor heights specified in Safety Guide No. 23.

The composite year reflects those periods in which data recovery was the greatest. The composite year consists of January through July of 1970, August 1971, September and October 1972 and November and December of 1970.

Incorporation of the aforementioned characteristics unique to the Indian Point valley site into diffusion calculations, insure that off-site doses following a loss-of-coolant accident are within the limitations outlined in 10 CFR 100.

2.6.5 Onsite Meteorological Measurements Program

The meteorological measurement program consists of three instrumented towers, redundant power and ventilation systems, redundant communication systems, and a mini-computer processor/recorder. The meteorological measurement program complies with the acceptance criteria stated in Section 2.3.3. and in Section 17.2 of NUREG-75/087 Revision 1 (superseded by NUREG-0800, Rev. 2, July 1981) with the former section dealing with meteorological sensors and recorders, and the latter dealing with the Quality Assurance Program. The meteorological measurements program consists of primary and backup systems. The accuracy of the meteorological sensor and recording systems meet the system specifications given in the Section C.4 of proposed Revision 1 to Regulatory Guide 1.23.

Primary System

A 122-meter, instrumented tower is located on the site and provides:

1. Wind direction and speed measurement at a minimum of two levels, one of which is representative of the 10-meter level:

2. Standard deviations of wind direction fluctuations as calculated at all measured levels;

3. Vertical temperature difference for two layers (122-10 meters and 60-10 meters);

4. Ambient temperature measurements at the 10-meter level;

5. Precipitation measurements near ground level;

6. Pasquill stability classes as calculated from temperature difference.

To assure acceptable date recovery, the meteorological measurements system and associated controlled environmental housing is connected to a power supply system which has a redundant power source. A diesel generator has been installed to provide immediate power to the meteorological tower system in the event of a power outage. The generator becomes fully powered within 15 seconds after an automatic transfer switch is tripped. Various support systems include an uninterruptible power supply, dedicated ventilation systems, halon fire protection, and dedicated communications.

The meteorological data is transmitted simultaneously to two data loggers located at the Primary Tower site. One data logger transmits 15-minute average meteorological data to a computer to determine joint frequency distributions, and the second data logger transmits 15-

minute average meteorological data to a computer located in the Buchanan Service Center, which provides the capability for accessing the meteorological data remotely.

Meteorological data can be transmitted simultaneously to the IP3 / IP2 emergency response organization and the NRC in a format designated by NUREG-0654/FEMA-REP-1.

Fifteen minute averages of meteorological parameters covering the 12-hour period previous to a recall command is available upon interrogation of the system.

Backup Systems

In the event of a failure of the primary meteorological measurement system, a backup meteorological system is used at the Indian Point site. This system is independent of the primary system and consists of an instrumented meteorological tower (a backup tower located approximately 2700 feet north of the primary tower). The associated data acquisition system for the backup tower is located in the Emergency Operations Facility. The backup system provides measurements of the 10-meter level of wind direction and speed, and an estimate of atmospheric stability (Pasquill category using sigma theta which is a standard deviation of wind fluctuation). The backup system provides information in the real-time mode. Changeover from the primary system to the backup system occurs automatically. In the event of a failure of the backup meteorological measurement system, a standby backup system exists at the 10-meter level of the Buchanan Service Center building roof. It also provides measurements of the 10-meter level of wind direction and speed, and an estimate of atmospheric stability (Pasquill category using sigma theta which is a standard deviation of wind fluctuations). The changeover from the backup system to the standby system also occurs automatically.

As in the case of the primary system, the backup meteorological measurements system and associated controlled environmental housing system is connected to a power system which is supplied form redundant power sources.

In addition to the backup meteorological measurements system, a backup communications line to the meteorological system is operational. During an interim period, the backup communications is provided via telephone lines routed through a telephone company central office separate from the primary circuits.

Atmospheric Dispersion Factors for Routine Releases

Extensive analyses and calculations were carried out in 1991[8] to reevaluate the atmospheric dispersion factors for routine releases at Indian Point. The primary objective was to ensure the applicability of the dispersion factors in the Offisite Dose Calculation Manual (ODCM) in view of potential changes in the meteorological conditions at the site.

In the analyses, consideration was given to an extended meteorological database and up-to-date analytical models. Briefly, use was made of the following:

1. 10 years' worth of hourly meteorological data collected on site for the period 1981 through 1990.

2. Mixed-mode releases, whereby released plumes can be totally elevated, totally at ground level, or in between,

3. Valley-flow considerations for the assessment of channeling and recirculation effects (based on a site specific study[9]), and

4. Analytical models which permit the computation of radiation exposures due to inhalation and due to immersion in finite clouds of radioactive material.

The locations of interest were the site boundary, the nearest residences in each sector, and milking animals at 5 miles. See Ref. 8 for complete details and results.

References

1. Kaplan, Edward J. and B. Wuebber 1984(a), Quarterly Summary of Meteorological Data from Indian Point Meteorological Systems, First Quarter, January 1 – March 31, 1984, Prepared for the New York Power Authority, May 1984, Project No. 01-4251-02-1.

2. Kaplin, Edward J. and B. Wuebber, 1984(b), Quarterly Summary of Meteorological Data from Indian Point Meteorological Systems, Second Quarter, April 1 – June 30, 1984. Prepared for the New York Power Authority, September 1984, Project No. 01-4251-02-1.

3. Kaplin, Edward J. and B. Wuebber, 1984(c), Quarterly Summary of Meteorological Data from Indian Point Meteorological Systems, Third Quarter, July 1 – September 30, 1984. Prepared for the New York Power Authority, December 1984, Project No. 01-4251-02-1.

4. Kaplin, Edward J. and B. Wuebber, 1985, Quarterly Summary of Meteorological Data from Indian Point Meteorological Systems, Fourth Quarter, October 1 – December 31, 1984, prepared for the New York Power Authority, June 1985, Project No. 01-4251-05-1.

5. Kaplin, Edwar J., B. Wuebber (1981) Facility Safety Analysis Report (FSAR), consolidated Edison Company of New York, Inc., Indian Point Nuclear Generating Unit No. 2, Meteorological update, September 1981, YSC Project No. 01-4122.

6. Nuclear Regulatory Commission, 1980, Proposed Revision 1 to Regulatory Guide 1.23, Meteorological Programs in Support of Nuclear Power Plants. U.S. Nuclear Regulatory Commission, Washington D.C.

7. Bryson Reid A. and F. K. Hare, 1974, World Survey of Climatology, volume 11, Climatology, volume 11, Climate of North America, Helmut Landsbert, Editor in Chief.

8. NYPA Corporate Radiological Engineering Calculation IP3-CALC-RAD-00001, “IP3 – Revised ODCM Atmospheric Dispersion Parameters (Multi-Year Hourly Data, Mixed-Mode Releases and Valley Effects” (10/1191)

9. Kaplin, Edwar J., “Wind Field Analysis at Indian Point,” York Services Corporation, Stamford, CT, Technical Report No. 4873-02 (3/19/91)

Table 2.6-1

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

Wind	Stablility Class						
Direction	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 2.6-2
[Historical Information]

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 2.6-3

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

Table 2.6-4
(Sheet 1 of 28)

JOINT FREQUENCY DISTRIBUTION
INDIAN POINT JAN-MAR 1986
10 METER WIND SPEED & DIR. WITH 61-10 METER DELTA T
PASQUILL CLASS A

WIND DIRECTION	WIND SPEED (MPH)						
	01-03	04-07	08-12	13-18	19-24	>24	TOTAL
N	7.	49.	20.	0.	0.	0.	76.
NNE	3.	7.	0.	0.	0.	0.	10.
NE	7.	2.	0.	0.	0.	0.	9.
ENE	1.	0.	0.	0.	0.	0.	1.
E	2.	0.	0.	0.	0.	0.	2.
ESE	2.	1.	0.	0.	0.	0.	3.
SE	6.	4.	0.	0.	0.	0.	10.
SSE	12.	21.	0.	0.	0.	0.	33.
S	8.	18.	6.	0.	0.	0.	32.
SSW	7.	11.	7.	0.	0.	0.	25.
SW	2.	2.	1.	0.	0.	0.	5.
WSW	0.	0.	0.	0.	0.	0.	0.
W	4.	6.	5.	1.	0	0.	16.
WNW	0.	29.	13.	0.	0.	0.	42.
NW	4.	35.	16.	0.	0.	0.	55.
NNW	12.	33.	7.	0.	0.	0.	52.
TOTAL	77.	218.	75.	1.	0.	0.	371.
CALM	0.						

Table 2.6-4
(Sheet 2 of 28)

JOINT FREQUENCY DISTRIBUTION
INDIAN POINT JAN-MAR 1986
10 METER WIND SPEED & DIR. WITH 61-10 METER DELTA T
PASQUILL CLASS B

WIND DIRECTION	WIND SPEED (MPH)

	01-03	04-07	08-12	13-18	19-24	>24	TOTAL
N	5.	14.	8.	0.	0.	0.	27.
NNE	2.	1.	0.	0.	0.	0.	3.
NE	3.	0.	0.	0.	0.	0.	3.
ENE	3.	0.	0.	0.	0.	0.	3.
E	3.	0.	0.	0.	0.	0.	3.
ESE	1.	0.	0.	0.	0.	0.	1.
SE	0.	0.	0.	0.	0.	0.	0.
SSE	2.	1.	0.	0.	0.	0.	3.
S	4.	5.	2.	0.	0.	0.	11.
SSW	3.	5.	0.	0.	0.	0.	8.
SW	1.	0.	0.	0.	0.	0.	1.
WSW	1.	0.	0.	0.	0.	0.	1.
W	0.	0.	1.	0.	0	0.	1.
WNW	3.	6.	6.	0.	0.	0.	15.
NW	2.	9.	8.	0.	0.	0.	19.
NNW	5.	8.	1.	0.	0.	0.	14.
TOTAL	38.	52.	26.	0.	0.	0.	113.

CALM 0.

Table 2.6-4
(Sheet 3 of 28)

JOINT FREQUENCY DISTRIBUTION
INDIAN POINT JAN-MAR 1986
10 METER WIND SPEED & DIR. WITH 61-10 METER DELTA T
PASQUILL CLASS C

WIND
DIRECTION WIND SPEED (MPH)

	01-03	04-07	08-12	13-18	19-24	>24	TOTAL
N	4.	7.	3.	0.	0.	0.	14.
NNE	7.	6.	0.	0.	0.	0.	13.
NE	3.	0.	0.	0.	0.	0.	3.
ENE	3.	0.	0.	0.	0.	0.	3.
E	1.	0.	0.	0.	0.	0.	1.
ESE	3.	0.	0.	0.	0.	0.	3.
SE	3.	2.	0.	0.	0.	0.	5.
SSE	1.	4.	0.	0.	0.	0.	5.
S	2.	2.	0.	0.	0.	0.	4.
SSW	0.	3.	0.	0.	0.	0.	3.
SW	1.	0.	0.	0.	0.	0.	1.
WSW	2.	0.	0.	0.	0.	0.	2.
W	1.	0.	1.	0.	0	0.	2.

WNW	3.	8.	3.	0.	0.	0.	14.
NW	2.	10.	13.	0.	0.	0.	25.
NNW	4.	10.	3.	0.	0.	0.	17.
TOTAL	40.	52.	23.	0.	0.	0.	115.
CALM	0.						

Table 2.6-4
(Sheet 4 of 28)

JOINT FREQUENCY DISTRIBUTION
INDIAN POINT JAN-MAR 1986
10 METER WIND SPEED & DIR. WITH 61-10 METER DELTA T
PASQUILL CLASS D

WIND DIRECTION	WIND SPEED (MPH)						
	01-03	04-07	08-12	13-18	19-24	>24	TOTAL
N	128.	109.	40.	0.	0.	0.	277.
NNE	62.	23.	2.	0.	0.	0.	87.
NE	29.	1.	0.	0.	0.	0.	30.
ENE	17.	0.	0.	0.	0.	0.	17.
E	11.	0.	0.	0.	0.	0.	11.
ESE	7.	0.	0.	0.	0.	0.	7.
SE	11.	0.	0.	0.	0.	0.	11.
SSE	13.	12.	0.	0.	0.	0.	25.
S	17.	23.	2.	0.	0.	0.	42.
SSW	5.	3.	5.	0.	0.	0.	13.
SW	0.	0.	0.	0.	0.	0.	0.
WSW	9.	0.	0.	0.	0.	0.	9.
W	6.	9.	2.	0.	0	0.	17.
WNW	11.	31.	18.	1.	0.	0.	61.
NW	22.	81.	45.	1.	0.	0.	149.
NNW	54.	64.	30.	0.	0.	0.	148.
TOTAL	402.	356.	144.	2.	0.	0.	904.
CALM	6.						

Table 2.6-4
(Sheet 5 of 28)

JOINT FREQUENCY DISTRIBUTION
INDIAN POINT JAN-MAR 1986
10 METER WIND SPEED & DIR. WITH 61-10 METER DELTA T
PASQUILL CLASS E

WIND DIRECTION	WIND SPEED (MPH)						
	01-03	04-07	08-12	13-18	19-24	>24	TOTAL
N	71.	13.	0.	0.	0.	0.	84.
NNE	67.	8.	0.	0.	0.	0.	75.
NE	49.	3.	0.	0.	0.	0.	52.
ENE	14.	0.	0.	0.	0.	0.	14.
E	20.	0.	0.	0.	0.	0.	20.
ESE	7.	0.	0.	0.	0.	0.	7.
SE	14.	2.	0.	0.	0.	0.	16.
SSE	12.	12.	0.	0.	0.	0.	24.
S	19.	14.	1.	0.	0.	0.	34.
SSW	7.	6.	1.	0.	0.	0.	14.
SW	5.	2.	0.	0.	0.	0.	7.
WSW	0.	0.	0.	0.	0.	0.	0.
W	9.	1.	0.	0.	0	0.	10.
WNW	16.	4.	0.	0.	0.	0.	20.
NW	13.	6.	1.	0.	0.	0.	20.
NNW	37.	5.	0.	0.	0.	0.	42.
TOTAL	360.	76.	3.	0.	0.	0.	439.
CALM	8.						

Table 2.6-4
(Sheet 6 of 28)

JOINT FREQUENCY DISTRIBUTION
INDIAN POINT JAN-MAR 1986
10 METER WIND SPEED & DIR. WITH 61-10 METER DELTA T
PASQUILL CLASS F

WIND DIRECTION	WIND SPEED (MPH)						
	01-03	04-07	08-12	13-18	19-24	>24	TOTAL
N	28.	1.	0.	0.	0.	0.	29.
NNE	61.	3.	0.	0.	0.	0.	64.
NE	19.	2.	0.	0.	0.	0.	21.
ENE	3.	0.	0.	0.	0.	0.	3.
E	10.	0.	0.	0.	0.	0.	10.
ESE	4.	0.	0.	0.	0.	0.	4.
SE	3.	0.	0.	0.	0.	0.	3.
SSE	9.	0.	0.	0.	0.	0.	9.
S	4.	3.	6.	0.	0.	0.	7.
SSW	1.	0.	0.	0.	0.	0.	1.

SW	1.	0.	0.	0.	0.	0.	1.
WSW	4.	0.	0.	0.	0.	0.	4.
W	1.	0.	0.	0.	0	0.	1.
WNW	2.	1.	0.	0.	0.	0.	3.
NW	0.	0.	0.	0.	0.	0.	0.
NNW	8.	0.	0.	0.	0.	0.	8.
TOTAL	158.	10.	0.	0.	0.	0.	168.
CALM	1.						

Table 2.6-4
(Sheet 7 of 28)

JOINT FREQUENCY DISTRIBUTION
INDIAN POINT JAN-MAR 1986
10 METER WIND SPEED & DIR. WITH 61-10 METER DELTA T
PASQUILL CLASS G

WIND DIRECTION	WIND SPEED (MPH)						
	01-03	04-07	08-12	13-18	19-24	>24	TOTAL
N	5.	0.	0.	0.	0.	0.	5.
NNE	10.	1.	0.	0.	0.	0.	11.
NE	3.	2.	0.	0.	0.	0.	5.
ENE	1.	0.	0.	0.	0.	0.	1.
E	2.	0.	0.	0.	0.	0.	2.
ESE	2.	0.	0.	0.	0.	0.	2.
SE	0.	0.	0.	0.	0.	0.	0.
SSE	1.	0.	0.	0.	0.	0.	1.
S	0.	0.	0.	0.	0.	0.	0.
SSW	1.	0.	0.	0.	0.	0.	1.
SW	2.	0.	0.	0.	0.	0.	2.
WSW	1.	0.	0.	0.	0.	0.	1.
W	3	0.	0.	1.	0	0.	3.
WNW	0.	0.	0.	0.	0.	0.	0.
NW	1.	0.	0.	0.	0.	0.	1.
NNW	0.	0.	0.	0.	0.	0.	0.
TOTAL	32.	3.	0.	0.	0.	0.	35.
CALM	0.						

Table 2.6-4
(Sheet 8 of 28)

JOINT FREQUENCY DISTRIBUTION

INDIAN POINT APR-JUNE 1986
10 METER WIND SPEED & DIR. WITH 61-10 METER DELTA T
PASQUILL CLASS A

WIND DIRECTION	WIND SPEED (MPH)						
	01-03	04-07	08-12	13-18	19-24	>24	TOTAL
N	7.	69.	31.	2.	0.	0.	109.
NNE	2.	4.	10.	0.	0.	0.	16.
NE	0.	2.	1.	0.	0.	0.	3.
ENE	0.	2.	1.	0.	0.	0.	3.
E	0.	0.	0.	0.	0.	0.	0.
ESE	1.	0.	0.	0.	0.	0.	1.
SE	2.	4.	0.	0.	0.	0.	6.
SSE	7.	30.	2.	0.	0.	0.	39.
S	7.	43.	12.	0.	0.	0.	62.
SSW	0.	10.	6.	0.	0.	0.	16.
SW	1.	15.	1.	0.	0.	0.	17.
WSW	1.	5.	0.	0.	0.	0.	6.
W	3.	13.	0.	0.	0	0.	16.
WNW	1.	9.	2.	0.	0.	0.	12.
NW	2.	20.	16.	0.	0.	0.	38.
NNW	4.	39.	11.	0.	0.	0.	54.
TOTAL	38.	265.	93.	2.	0.	0.	398.
CALM	0.						

Table 2.6-4
(Sheet 9 of 28)

JOINT FREQUENCY DISTRIBUTION
INDIAN POINT APR-JUNE 1986
10 METER WIND SPEED & DIR. WITH 61-10 METER DELTA T
PASQUILL CLASS B

WIND DIRECTION	WIND SPEED (MPH)						
	01-03	04-07	08-12	13-18	19-24	>24	TOTAL
N	1.	24.	4.	2.	0.	0.	31.
NNE	1.	6.	5.	0.	0.	0.	12.
NE	1.	1.	2.	0.	0.	0.	4.
ENE	0.	2.	0.	0.	0.	0.	2.
E	1.	0.	0.	0.	0.	0.	1.
ESE	0.	0.	0.	0.	0.	0.	0.
SE	2.	1.	0.	0.	0.	0.	3.

SSE	2.	6.	0.	0.	0.	0.	8.
S	2.	11.	1.	0.	0.	0.	14.
SSW	1.	2.	1.	0.	0.	0.	4.
SW	3.	1.	0.	0.	0.	0.	4.
WSW	0.	1.	0.	0.	0.	0.	1.
W	3.	1.	0.	0.	0	0.	4.
WNW	1.	2.	2.	0.	0.	0.	5.
NW	1.	6.	2.	0.	0.	0.	9.
NNW	1.	6.	0.	0.	0.	0.	7.
TOTAL	20.	70.	17.	2.	0.	0.	109.
CALM	0.						

Table 2.6-4
(Sheet 10 of 28)

JOINT FREQUENCY DISTRIBUTION
INDIAN POINT APR-JUNE 1986
10 METER WIND SPEED & DIR. WITH 61-10 METER DELTA T
PASQUILL CLASS C

WIND DIRECTION	WIND SPEED (MPH)						
	01-03	04-07	08-12	13-18	19-24	>24	TOTAL
N	6.	10.	4.	0.	0.	0.	20.
NNE	4.	7.	4.	0.	0.	0.	15.
NE	0.	2.	0.	0.	0.	0.	2.
ENE	1.	0.	0.	0.	0.	0.	1.
E	1.	0.	0.	0.	0.	0.	1.
ESE	0.	0.	0.	0.	0.	0.	0.
SE	2.	0.	0.	0.	0.	0.	2.
SSE	6.	3.	0.	0.	0.	0.	9.
S	7.	11.	0.	0.	0.	0.	18.
SSW	1.	4.	3.	0.	0.	0.	8.
SW	2.	1.	1.	0.	0.	0.	4.
WSW	0.	1.	0.	0.	0.	0.	1.
W	3.	4.	0.	0.	0	0.	7.
WNW	0.	3.	0.	0.	0.	0.	3.
NW	0.	1.	1.	0.	0.	0.	2.
NNW	3.	5.	1.	0.	0.	0.	9.
TOTAL	36.	52.	14.	0.	0.	0.	102.
CALM	0.						

Table 2.6-4
(Sheet 11 of 28)

JOINT FREQUENCY DISTRIBUTION
INDIAN POINT APR-JUNE 1986
10 METER WIND SPEED & DIR. WITH 61-10 METER DELTA T
PASQUILL CLASS D

WIND DIRECTION	WIND SPEED (MPH) 01-03	04-07	08-12	13-18	19-24	>24	TOTAL
N	28.	67.	20.	13.	5.	0.	133.
NNE	34.	44.	22.	0.	0.	0.	100.
NE	45.	24.	2.	0.	0.	0.	71.
ENE	32.	8.	0.	0.	0.	0.	40.
E	18.	0.	0.	0.	0.	0.	18.
ESE	14.	2.	0.	0.	0.	0.	16.
SE	23.	5.	0.	0.	0.	0.	28.
SSE	20.	41.	0.	0.	0.	0.	61.
S	24.	37.	3.	0.	0.	0.	64.
SSW	16.	11.	1.	0.	0.	0.	28.
SW	6.	3.	0.	0.	0.	0.	9.
WSW	7.	6.	0.	0.	0.	0.	13.
W	3.	4.	0.	0.	0	0.	7.
WNW	2.	18.	2.	0.	0.	0.	22.
NW	1.	15.	3.	0.	0.	0.	19.
NNW	4.	21.	10.	0.	0.	0.	35.
TOTAL	277.	306.	63.	13.	5.	0.	664.
CALM	0.						

Table 2.6-4
(Sheet 12 of 28)

JOINT FREQUENCY DISTRIBUTION
INDIAN POINT APR-JUNE 1986
10 METER WIND SPEED & DIR. WITH 61-10 METER DELTA T
PASQUILL CLASS E

WIND DIRECTION	WIND SPEED (MPH) 01-03	04-07	08-12	13-18	19-24	>24	TOTAL
N	37.	41.	7.	0.	0.	0.	85.
NNE	44.	32.	7.	0.	0.	0.	83.
NE	72.	15.	0.	0.	0.	0.	87.
ENE	47.	3.	1.	0.	0.	0.	51.

E	15.	0.	0.	0.	0.	0.	15.
ESE	12.	0.	0.	0.	0.	0.	12.
SE	26.	2.	0.	0.	0.	0.	28.
SSE	37.	28.	0.	0.	0.	0.	65.
S	35.	39.	0.	0.	0.	0.	74.
SSW	15.	19.	1.	0.	0.	0.	35.
SW	6.	3.	0.	0.	0.	0.	9.
WSW	4.	2.	0.	0.	0.	0.	6.
W	7.	7.	1.	1.	0	0.	15.
WNW	5.	7.	0.	0.	0.	0.	12.
NW	1.	10.	0.	0.	0.	0.	11.
NNW	9.	13.	3.	0.	0.	0.	25.
TOTAL	372.	221.	20.	0.	0.	0.	613.
CALM	1.						

Table 2.6-4
(Sheet 13 of 28)

JOINT FREQUENCY DISTRIBUTION
INDIAN POINT APR-JUNE1986
10 METER WIND SPEED & DIR. WITH 61-10 METER DELTA T
PASQUILL CLASS F

WIND DIRECTION	WIND SPEED (MPH)						
	01-03	04-07	08-12	13-18	19-24	>24	TOTAL
N	13.	1.	0.	0.	0.	0.	14.
NNE	48.	7.	0.	0.	0.	0.	55.
NE	59.	16.	0.	0.	0.	0.	75.
ENE	25.	0.	0.	0.	0.	0.	25.
E	18.	0.	0.	0.	0.	0.	18.
ESE	5.	0.	0.	0.	0.	0.	5.
SE	5.	1.	0.	0.	0.	0.	6.
SSE	8.	0.	0.	0.	0.	0.	8.
S	12.	3.	6.	0.	0.	0.	15.
SSW	6.	0.	0.	0.	0.	0.	6.
SW	1.	0.	0.	0.	0.	0.	1.
WSW	0.	0.	0.	0.	0.	0.	0.
W	1.	0.	0.	0.	0	0.	1.
WNW	1.	0.	0.	0.	0.	0.	1.
NW	3.	0.	0.	0.	0.	0.	3.
NNW	1.	0.	0.	0.	0.	0.	1.
TOTAL	206.	28.	0.	0.	0.	0.	234.
CALM	0.						

Table 2.6-4
(Sheet 14 of 28)

JOINT FREQUENCY DISTRIBUTION
INDIAN POINT APR-JUNE 1986
10 METER WIND SPEED & DIR. WITH 61-10 METER DELTA T
PASQUILL CLASS G

WIND DIRECTION	WIND SPEED (MPH)						
	01-03	04-07	08-12	13-18	19-24	>24	TOTAL
N	3.	1.	0.	0.	0.	0.	4.
NNE	13.	0.	0.	0.	0.	0.	13.
NE	12.	4.	0.	0.	0.	0.	16.
ENE	1.	0.	0.	0.	0.	0.	1.
E	3.	0.	0.	0.	0.	0.	3.
ESE	1.	0.	0.	0.	0.	0.	1.
SE	0.	0.	0.	0.	0.	0.	0.
SSE	0.	0.	0.	0.	0.	0.	0.
S	2.	0.	0.	0.	0.	0.	2.
SSW	1.	0.	0.	0.	0.	0.	1.
SW	0.	0.	0.	0.	0.	0.	0.
WSW	0.	0.	0.	0.	0.	0.	0.
W	0.	0.	0.	0.	0	0.	0.
WNW	0.	0.	0.	0.	0.	0.	0.
NW	0.	0.	0.	0.	0.	0.	0.
NNW	2.	0.	0.	0.	0.	0.	2.
TOTAL	38.	5.	0.	0.	0.	0.	43.
CALM	0.						

Table 2.6-4
(Sheet 15 of 28)

JOINT FREQUENCY DISTRIBUTION
INDIAN POINT JULY-SEPT 1986
10 METER WIND SPEED & DIR. WITH 61-10 METER DELTA T
PASQUILL CLASS A

WIND DIRECTION	WIND SPEED (MPH)						
	01-03	04-07	08-12	13-18	19-24	>24	TOTAL
N	4.	67.	6.	0.	0.	0.	77.

NNE	1.	9.	1.	0.	0.	0.	11.
NE	1.	4.	2.	0.	0.	0.	7.
ENE	2.	1.	0.	0.	0.	0.	3.
E	0.	0.	0.	0.	0.	0.	0.
ESE	1.	0.	0.	0.	0.	0.	1.
SE	5.	2.	0.	0.	0.	0.	7.
SSE	11.	14.	0.	0.	0.	0.	25.
S	19.	72.	3.	0.	0.	0.	94.
SSW	7.	25.	8.	0.	0.	0.	40.
SW	3.	13.	0.	0.	0.	0.	16.
WSW	1.	7.	0.	0.	0.	0.	8.
W	6.	16.	0.	0.	0	0.	22.
WNW	2.	5.	0.	0.	0.	0.	7.
NW	2.	16.	4.	0.	0.	0.	22.
NNW	5.	26.	6.	0.	0.	0.	37.
TOTAL	70.	277.	30.	0.	0.	0.	377.
CALM	0.						

Table 2.6-4
(Sheet 16 of 28)

JOINT FREQUENCY DISTRIBUTION
INDIAN POINT JULY-SEPT 1986
10 METER WIND SPEED & DIR. WITH 61-10 METER DELTA T
PASQUILL CLASS B

WIND DIRECTION	WIND SPEED (MPH)						
	01-03	04-07	08-12	13-18	19-24	>24	TOTAL
N	4.	18.	1.	0.	0.	0.	23.
NNE	3.	9.	1.	0.	0.	0.	13.
NE	0.	1.	0.	0.	0.	0.	1.
ENE	1.	0.	0.	0.	0.	0.	1.
E	1.	0.	0.	0.	0.	0.	1.
ESE	1.	0.	0.	0.	0.	0.	1.
SE	1.	0.	0.	0.	0.	0.	1.
SSE	1.	1.	0.	0.	0.	0.	2.
S	8.	18.	1.	0.	0.	0.	27.
SSW	2.	4.	0.	0.	0.	0.	6.
SW	1.	5.	0.	0.	0.	0.	6.
WSW	1.	1.	0.	0.	0.	0.	2.
W	3.	2.	0.	0.	0	0.	5.
WNW	1.	2.	0.	0.	0.	0.	3.
NW	2.	0.	0.	0.	0.	0.	2.
NNW	1.	0.	0.	0.	0.	0.	1.

TOTAL	31.	61.	3.	0.	0.	0.	95.

CALM 0.

Table 2.6-4
(Sheet 17 of 28)

JOINT FREQUENCY DISTRIBUTION
INDIAN POINT JULY-SEPT1986
10 METER WIND SPEED & DIR. WITH 61-10 METER DELTA T
PASQUILL CLASS C

WIND DIRECTION	WIND SPEED (MPH)						
	01-03	04-07	08-12	13-18	19-24	>24	TOTAL
N	3.	21.	0.	0.	0.	0.	24.
NNE	2.	2.	0.	0.	0.	0.	4.
NE	4.	4.	1.	0.	0.	0.	9.
ENE	1.	0.	0.	0.	0.	0.	1.
E	1.	1.	0.	0.	0.	0.	2.
ESE	2.	0.	0.	0.	0.	0.	2.
SE	2.	0.	0.	0.	0.	0.	2.
SSE	3.	3.	0.	0.	0.	0.	6.
S	9.	15.	0.	0.	0.	0.	24.
SSW	3.	4.	1.	0.	0.	0.	8.
SW	1.	1.	0.	0.	0.	0.	2.
WSW	0.	1.	0.	0.	0.	0.	1.
W	3.	4.	0.	0.	0	0.	7.
WNW	0.	0.	0.	0.	0.	0.	0.
NW	0.	2.	0.	0.	0.	0.	2.
NNW	4.	3.	0.	0.	0.	0.	7.
TOTAL	38.	61.	2.	0.	0.	0.	101.

CALM 0.

Table 2.6-4
(Sheet 18 of 28)

JOINT FREQUENCY DISTRIBUTION
INDIAN POINT JULY-SEPT 1986
10 METER WIND SPEED & DIR. WITH 61-10 METER DELTA T
PASQUILL CLASS D

WIND DIRECTION	WIND SPEED (MPH)

	01-03	04-07	08-12	13-18	19-24	>24	TOTAL
N	11.	77.	7.	0.	0.	0.	95.
NNE	21.	36.	2.	0.	0.	0.	59.
NE	34.	22.	0.	0.	0.	0.	56.
ENE	34.	5.	0.	0.	0.	0.	39.
E	20.	6.	0.	0.	0.	0.	26.
ESE	5.	2.	0.	0.	0.	0.	7.
SE	22.	0.	0.	0.	0.	0.	22.
SSE	13.	4.	0.	0.	0.	0.	17.
S	43.	86.	5.	0.	0.	0.	134.
SSW	15.	39.	4.	0.	0.	0.	58.
SW	11.	3.	0.	0.	0.	0.	14.
WSW	12.	2.	0.	0.	0.	0.	14.
W	6.	8.	0.	0.	0	0.	14.
WNW	2.	3.	1.	0.	0.	0.	6.
NW	2.	8.	1.	0.	0.	0.	11.
NNW	5.	13.	1.	0.	0.	0.	19.
TOTAL	256.	314.	21.	0.	0.	0.	591.
CALM	0.						

Table 2.6-4
(Sheet 19 of 28)

JOINT FREQUENCY DISTRIBUTION
INDIAN POINT JULY-SEPT 1986
10 METER WIND SPEED & DIR. WITH 61-10 METER DELTA T
PASQUILL CLASS E

WIND DIRECTION	WIND SPEED (MPH)						
	01-03	04-07	08-12	13-18	19-24	>24	TOTAL
N	31.	41.	0.	0.	0.	0.	72.
NNE	52.	49.	0.	0.	0.	0.	101.
NE	56.	38.	0.	0.	0.	0.	94.
ENE	26.	3.	0.	0.	0.	0.	29.
E	23.	2.	0.	0.	0.	0.	25.
ESE	19.	0.	0.	0.	0.	0.	19.
SE	36.	0.	0.	0.	0.	0.	36.
SSE	31.	2.	0.	0.	0.	0.	33.
S	76.	95.	2.	0.	0.	0.	173.
SSW	55.	42.	2.	0.	0.	0.	99.
SW	18.	3.	1.	0.	0.	0.	22.
WSW	9.	3.	0.	0.	0.	0.	12.
W	11.	4.	0.	0.	0	0.	15.
WNW	10.	4.	0.	0.	0.	0.	14.

NW	19.	6.	0.	0.	0.	0.	25.
NNW	14.	18.	0.	0.	0.	0.	32.
TOTAL	486.	310.	5.	0.	0.	0.	801.
CALM	14.						

Table 2.6-4
(Sheet 20 of 28)

JOINT FREQUENCY DISTRIBUTION
INDIAN POINT JULY-SEPT 1986
10 METER WIND SPEED & DIR. WITH 61-10 METER DELTA T
PASQUILL CLASS F

WIND DIRECTION	WIND SPEED (MPH)						
	01-03	04-07	08-12	13-18	19-24	>24	TOTAL
N	21.	2.	0.	0.	0.	0.	23.
NNE	32.	6.	0.	0.	0.	0.	38.
NE	43.	24.	1.	0.	0.	0.	68.
ENE	15.	0.	0.	0.	0.	0.	15.
E	17.	0.	0.	0.	0.	0.	17.
ESE	6.	0.	0.	0.	0.	0.	6.
SE	8.	0.	0.	0.	0.	0.	8.
SSE	12.	0.	0.	0.	0.	0.	12.
S	6.	1.	0.	0.	0.	0.	7.
SSW	4.	0.	0.	0.	0.	0.	4.
SW	2.	0.	0.	0.	0.	0.	2.
WSW	1.	0.	0.	0.	0.	0.	1.
W	2.	0.	0.	0.	0	0.	2.
WNW	2.	0.	0.	0.	0.	0.	2.
NW	5.	0.	0.	0.	0.	0.	5.
NNW	2.	0.	0.	0.	0.	0.	2.
TOTAL	178.	33.	1.	0.	0.	0.	212.
CALM	4.						

Table 2.6-4
(Sheet 21 of 28)

JOINT FREQUENCY DISTRIBUTION
INDIAN POINT JULY-SEPT 1986
10 METER WIND SPEED & DIR. WITH 61-10 METER DELTA T
PASQUILL CLASS G

WIND DIRECTION	WIND SPEED (MPH)					
	01-03	04-07	08-12	13-18	19-24	>24	TOTAL
N	2.	0.	0.	0.	0.	0.	2.
NNE	1.	0.	0.	0.	0.	0.	1.
NE	3.	3.	0.	0.	0.	0.	6.
ENE	1.	0.	0.	0.	0.	0.	1.
E	0.	0.	0.	0.	0.	0.	0.
ESE	1.	0.	0.	0.	0.	0.	1.
SE	0.	0.	0.	0.	0.	0.	0.
SSE	0.	0.	0.	0.	0.	0.	0.
S	2.	0.	0.	0.	0.	0.	2.
SSW	0.	0.	0.	0.	0.	0.	0.
SW	0.	0.	0.	0.	0.	0.	0.
WSW	0.	0.	0.	0.	0.	0.	0.
W	0.	0.	0.	0.	0	0.	0.
WNW	0.	0.	0.	0.	0.	0.	0.
NW	0.	0.	0.	0.	0.	0.	0.
NNW	0.	0.	0.	0.	0.	0.	0.
TOTAL	10.	3.	0.	0.	0.	0.	13.
CALM	0.						

Table 2.6-4
(Sheet 22 of 28)

JOINT FREQUENCY DISTRIBUTION
INDIAN POINT OCT-DEC 1986
10 METER WIND SPEED & DIR. WITH 61-10 METER DELTA T
PASQUILL CLASS A

WIND DIRECTION	WIND SPEED (MPH)						
	01-03	04-07	08-12	13-18	19-24	>24	TOTAL
N	1.	22.	6.	0.	0.	0.	29.
NNE	0.	0.	0.	0.	0.	0.	0.
NE	0.	0.	0.	0.	0.	0.	0.
ENE	0.	0.	0.	0.	0.	0.	0.
E	0.	0.	0.	0.	0.	0.	0.
ESE	0.	1.	0.	0.	0.	0.	1.
SE	0.	0.	0.	0.	0.	0.	0.
SSE	3.	9.	0.	0.	0.	0.	12.
S	6.	16.	4.	0.	0.	0.	26.
SSW	1.	4.	5.	0.	0.	0.	10.
SW	0.	8.	0.	0.	0.	0.	8.

WSW	0.	1.	0.	0.	0.	0.	1.
W	1.	6.	1.	0.	0	0.	8.
WNW	0.	11.	1.	0.	0.	0.	12.
NW	0.	16.	6.	0.	0.	0.	22.
NNW	1.	12.	2.	0.	0.	0.	15.
TOTAL	13.	106.	25.	0.	0.	0.	144.
CALM	0.						

Table 2.6-4
(Sheet 23 of 28)

JOINT FREQUENCY DISTRIBUTION
INDIAN POINT OCT-DEC 1986
10 METER WIND SPEED & DIR. WITH 61-10 METER DELTA T
PASQUILL CLASS B

WIND DIRECTION	WIND SPEED (MPH)						
	01-03	04-07	08-12	13-18	19-24	>24	TOTAL
N	0.	16.	6.	0.	0.	0.	22.
NNE	0.	1.	1.	0.	0.	0.	2.
NE	0.	0.	0.	0.	0.	0.	0.
ENE	0.	1.	0.	0.	0.	0.	1.
E	0.	0.	0.	0.	0.	0.	0.
ESE	0.	2.	0.	0.	0.	0.	2.
SE	1.	0.	0.	0.	0.	0.	1.
SSE	1.	2.	0.	0.	0.	0.	3.
S	4.	10.	1.	0.	0.	0.	15.
SSW	2.	2.	1.	0.	0.	0.	5.
SW	0.	1.	0.	0.	0.	0.	1.
WSW	2.	0.	0.	0.	0.	0.	2.
W	0.	0.	0.	0.	0	0.	0.
WNW	1.	3.	0.	0.	0.	0.	4.
NW	1.	4.	5.	1.	0.	0.	11.
NNW	2.	8.	5.	0.	0.	0.	15.
TOTAL	14.	50.	19.	1.	0.	0.	84.
CALM	0.						

Table 2.6-4
(Sheet 24 of 28)

JOINT FREQUENCY DISTRIBUTION
INDIAN POINT OCT-DEC 1986
10 METER WIND SPEED & DIR. WITH 61-10 METER DELTA T

PASQUILL CLASS C

WIND DIRECTION	WIND SPEED (MPH)					
	01-03	04-07	08-12	13-18	19-24	>24	TOTAL
N	3.	14.	6.	1.	0.	0.	24.
NNE	1.	4.	0.	0.	0.	0.	5.
NE	0.	1.	0.	0.	0.	0.	1.
ENE	1.	1.	0.	0.	0.	0.	2.
E	0.	1.	0.	0.	0.	0.	1.
ESE	0.	0.	0.	0.	0.	0.	0.
SE	0.	0.	0.	0.	0.	0.	0.
SSE	2.	1.	0.	0.	0.	0.	3.
S	5.	3.	0.	0.	0.	0.	8.
SSW	7.	3.	1.	0.	0.	0.	11.
SW	1.	0.	0.	0.	0.	0.	1.
WSW	0.	0.	0.	0.	0.	0.	0.
W	1.	1.	1.	0.	0	0.	3.
WNW	1.	2.	3.	0.	0.	0.	6.
NW	2.	2.	3.	0.	0.	0.	7.
NNW	4.	7.	5.	0.	0.	0.	16.
TOTAL	28.	40.	19.	1.	0.	0.	88.
CALM	0.						

Table 2.6-4
(Sheet 25 of 28)

JOINT FREQUENCY DISTRIBUTION
INDIAN POINT OCT-DEC1986
10 METER WIND SPEED & DIR. WITH 61-10 METER DELTA T
PASQUILL CLASS D

WIND DIRECTION	WIND SPEED (MPH)						
	01-03	04-07	08-12	13-18	19-24	>24	TOTAL
N	45.	127.	41.	8.	0.	0.	221.
NNE	29.	91.	26.	1.	0.	0.	147.
NE	19.	32.	2.	0.	0.	0.	53.
ENE	16.	13.	0.	0.	0.	0.	29.
E	8.	2.	0.	0.	0.	0.	10.
ESE	9.	0.	0.	0.	0.	0.	9.
SE	10.	3.	0.	0.	0.	0.	13.
SSE	14.	2.	0.	0.	0.	0.	16.

S	33.	48.	0.	0.	0.	0.	81.
SSW	28.	13.	1.	0.	0.	0.	42.
SW	13.	1.	0.	0.	0.	0.	14.
WSW	8.	4.	1.	0.	0.	0.	13.
W	10.	15.	5.	0.	0	0.	30.
WNW	4.	21.	7.	2.	0.	0.	34.
NW	7.	46.	28.	2.	0.	0.	83.
NNW	14.	47.	28.	4.	0.	0.	93.
TOTAL	267.	465.	139.	17.	0.	0.	888.

CALM 0.

Table 2.6-4
(Sheet 26 of 28)

JOINT FREQUENCY DISTRIBUTION
INDIAN POINT OCT-DEC1986
10 METER WIND SPEED & DIR. WITH 61-10 METER DELTA T
PASQUILL CLASS E

WIND DIRECTION	WIND SPEED (MPH)						
	01-03	04-07	08-12	13-18	19-24	>24	TOTAL
N	41.	26.	0.	0.	0.	0.	67.
NNE	59.	37.	2.	0.	0.	0.	98.
NE	58.	28.	0.	0.	0.	0.	86.
ENE	17.	4.	1.	0.	0.	0.	22.
E	17.	1.	0.	0.	0.	0.	18.
ESE	13.	1.	0.	0.	0.	0.	14.
SE	22.	0.	0.	0.	0.	0.	22.
SSE	33.	2.	0.	0.	0.	0.	35.
S	60.	55.	2.	0.	0.	0.	117.
SSW	30.	17.	0.	0.	0.	0.	47.
SW	23.	10.	0.	0.	0.	0.	33.
WSW	22	4.	1.	0.	0.	0.	27.
W	18.	32.	1.	0.	0	0.	51.
WNW	16.	19.	0.	0.	0.	0.	35.
NW	14.	19.	4.	0.	0.	0.	37.
NNW	20.	10.	4.	0.	0.	0.	34.
TOTAL	463.	265.	15.	0.	0.	0.	743.

CALM 0.

Table 2.6-4
(Sheet 27 of 28)

JOINT FREQUENCY DISTRIBUTION
INDIAN POINT OCT-DEC 1986
10 METER WIND SPEED & DIR. WITH 61-10 METER DELTA T
PASQUILL CLASS F

WIND DIRECTION	WIND SPEED (MPH) 01-03	04-07	08-12	13-18	19-24	>24	TOTAL
N	25.	2.	0.	0.	0.	0.	27.
NNE	47.	4.	0.	0.	0.	0.	51.
NE	46.	30.	0.	0.	0.	0.	76.
ENE	13.	2.	0.	0.	0.	0.	15.
E	9.	0.	0.	0.	0.	0.	9.
ESE	0.	0.	0.	0.	0.	0.	0.
SE	5.	0.	0.	0.	0.	0.	5.
SSE	6.	0.	0.	0.	0.	0.	6.
S	10.	2.	0.	0.	0.	0.	12.
SSW	6.	0.	0.	0.	0.	0.	6.
SW	3.	0.	0.	0.	0.	0.	3.
WSW	6.	0.	0.	0.	0.	0.	6.
W	3.	0.	0.	0.	0	0.	3.
WNW	2.	0.	0.	0.	0.	0.	2.
NW	4.	0.	0.	0.	0.	0.	4.
NNW	11.	0.	0.	0.	0.	0.	11.
TOTAL	196.	40.	0.	0.	0.	0.	236.
CALM	0.						

Table 2.6-4
(Sheet 28 of 28)

JOINT FREQUENCY DISTRIBUTION
INDIAN POINT OCT-DEC 1986
10 METER WIND SPEED & DIR. WITH 61-10 METER DELTA T
PASQUILL CLASS G

WIND DIRECTION	WIND SPEED (MPH) 01-03	04-07	08-12	13-18	19-24	>24	TOTAL
N	6.	0.	0.	0.	0.	0.	6.
NNE	3.	1.	0.	0.	0.	0.	4.
NE	4.	5.	0.	0.	0.	0.	9.
ENE	1.	0.	0.	0.	0.	0.	1.
E	0.	0.	0.	0.	0.	0.	0.

ESE	0.	0.	0.	0.	0.	0.	0.
SE	0.	0.	0.	0.	0.	0.	0.
SSE	1.	0.	0.	0.	0.	0.	1.
S	0.	0.	0.	0.	0.	0.	0.
SSW	1.	0.	0.	0.	0.	0.	1.
SW	0.	0.	0.	0.	0.	0.	0.
WSW	0.	0.	0.	0.	0.	0.	0.
W	0.	0.	0.	0.	0	0.	0.
WNW	1.	0.	0.	0.	0.	0.	1.
NW	0.	0.	0.	0.	0.	0.	0.
NNW	2.	0.	0.	0.	0.	0.	2.
TOTAL	19.	6.	0.	0.	0.	0.	25.
CALM	0.						

Table 2.6-5

(Sheet 1 of 14)
[Historical Information]

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND DIRECTION FOR PASQUILL STABILITY CATEGORY A

Indian Point B(3) Using a delta t correction factor of 0.605

Jan 1 1970 to Dec 31 1972 (Jan-July), Nov-Dec. 1970, Aug 1971, Sept-Oct 1972)

WIND DIRECTION		WIND SPEED (MPH)							
		01-03	04-07	08-12	13-18	19-24	Greater than 24	MISS	TOTAL
349-11	N	.0001	.0031	.0039	.0008	.0001	.0000	.0000	.0080
12-33	NNE	.0000	.0011	.0007	.0002	.0000	.0000	.0000	.0021
34-56	NE	.0000	.0003	.0003	.0000	.0000	.0000	.0000	.0007
57-78	ENE	.0000	.0005	.0000	.0001	.0001	.0000	.0000	.0007
79-101	E	.0000	.0000	.0000	.0001	.0000	.0000	.0000	.0001
102-123	ESE	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
124-146	SE	.0000	.0002	.0002	.0010	.0002	.0000	.0000	.0017
147-168	SSE	.0001	.0011	.0056	.0037	.0002	.0000	.0000	.0107
169-191	S	.0000	.0026	.0026	.0010	.0000	.0000	.0000	.0063
192-213	SSW	.0001	.0019	.0015	.0001	.0000	.0000	.0000	.0037
214-236	SW	.0000	.0015	.0009	.0001	.0000	.0000	.0000	.0025
237-258	WSW	.0002	.0008	.0008	.0000	.0000	.0000	.0000	.0018
259-281	W	.0002	.0014	.0016	.0001	.0001	.0000	.0001	.0035
282-303	WNW	.0001	.0002	.0015	.0023	.0009	.0000	.0008	.0058
304-326	NW	.0000	.0009	.0022	.0026	.0019	.0001	.0003	.0081
327-348	NNW	.0000	.0018	.0027	.0021	.0006	.0000	.0000	.0072
CALM		.0000							.0000
MISS		.0000	.0000	.0000	.0000	.0000	.0000		.0016
TOTAL		.0009	.0176	.0246	.0143	.0042	.0001	.0029	.0646

Percentage of hours of temperature difference present in this stability category = 6.5 Numbers of hours in this stability category = 565

Table 2.6-5 (Sheet 2 of 14)

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND DIRECTION FOR PASQUILL STABILITY CATEGORY B

Indian Point B(3) Using a delta t correction factor of 0.605

JAN. 1 1970 1 DEC. 31 1972

WIND DIRECTION		WIND SPEED (MPH)							
		01-03	04-07	08-12	13-18	19-24	Greater than 24	MISS	TOTAL
349-11	N	.0002	.0010	.0014	.0005	.0001	.0000	.0000	.0032
12-33	NNE	.0001	.0006	.0006	.0003	.0000	.0000	.0000	.0016
34-56	NE	.0000	.0005	.0001	.0000	.0000	.0000	.0000	.0006
57-78	ENE	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
79-101	E	.0000	.0003	.0001	.0001	.0000	.0000	.0000	.0006
102-123	ESE	.0000	.0000	.0001	.0000	.0000	.0000	.0000	.0001
124-146	SE	.0000	.0002	.0002	.0001	.0000	.0000	.0000	.0006
147-168	SSE	.0000	.0008	.0017	.0014	.0000	.0000	.0000	.0039
169-191	S	.0002	.0017	.0014	.0002	.0000	.0000	.0000	.0035
192-213	SSW	.0001	.0010	.0006	.0000	.0000	.0000	.0000	.0017
214-236	SW	.0002	.0003	.0005	.0001	.0000	.0000	.0000	.0011
237-258	WSW	.0002	.0000	.0003	.0000	.0000	.0000	.0001	.0007
259-281	W	.0001	.0003	.0005	.0005	.0000	.0000	.0001	.0015
282-303	WNW	.0001	.0003	.0003	.0002	.0000	.0000	.0002	.0015
304-326	NW	.0001	.0002	.0003	.0008	.0002	.0001	.0001	.0018
327-348	NNW	.0000	.0006	.0014	.0005	.0003	.0001	.0000	.0029
CALM		.0000							.0000
MISS		.0000	.0000	.0000	.0000	.0000	.0000		.0002
TOTAL		.0015	.0080	.0095	.0047	.0009	.0002	.0007	.0255

Percentage of hours of temperature difference present in this stability category = 2.5 Numbers of hours in this stability category = 223

Table 2.6-5 (Sheet 3 of 14)

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND DIRECTION FOR PASQUILL STABILITY CATEGORY C

Indian Point B(3) Using a delta t correction factor of 0.605

JAN. 1 1970 1 DEC. 31 1972

WIND DIRECTION		WIND SPEED (MPH)							
		01-03	04-07	08-12	13-18	19-24	Greater than 24	MISS	TOTAL
349-11	N	.0005	.0008	.0010	.0005	.0000	.0000	.0001	.0025
12-33	NNE	.0001	.0003	.0008	.0002	.0000	.0000	.0000	.0015
34-56	NE	.0002	.0003	.0001	.0000	.0000	.0000	.0000	.0007
57-78	ENE	.0001	.0000	.0000	.0000	.0001	.0000	.0000	.0002
79-101	E	.0000	.0000	.0001	.0000	.0000	.0000	.0000	.0001
102-123	ESE	.0000	.0002	.0000	.0001	.0000	.0000	.0000	.0003
124-146	SE	.0000	.0000	.0002	.0001	.0000	.0000	.0000	.0003
147-168	SSE	.0000	.0014	.0013	.0007	.0001	.0000	.0000	.0034
169-191	S	.0000	.0013	.0010	.0001	.0000	.0000	.0000	.0024
192-213	SSW	.0000	.0008	.0003	.0000	.0000	.0000	.0000	.0011
214-236	SW	.0001	.0002	.0001	.0000	.0000	.0000	.0000	.0005
237-258	WSW	.0001	.0002	.0003	.0000	.0000	.0000	.0000	.0007
259-281	W	.0002	.0000	.0010	.0002	.0001	.0000	.0000	.0016
282-303	WNW	.0001	.0000	.0005	.0005	.0002	.0002	.0000	.0015
304-326	NW	.0003	.0002	.0005	.0007	.0006	.0003	.0000	.0026
327-348	NNW	.0001	.0006	.0007	.0007	.0002	.0000	.0000	.0023
CALM		.0000							.0000
MISS		.0000	.0000	.0000	.0000	.0000	.0000		.0001
TOTAL		.0019	.0064	.0080	.0038	.0014	.0006	.0002	.0223

Percentage of hours of temperature difference present in this stability category = 2.2 Numbers of hours in this stability category = 195

Table 2.6-5 (Sheet 4 of 14)

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND
DIRECTION FOR PASQUILL STABILITY CATEGORY D

Indian Point B(3) Using a delta t correction factor of 0.605

JAN. 1 1970 1 DEC. 31 1972

WIND DIRECTION		WIND SPEED (MPH) 01-03	04-07	08-12	13-18	19-24	Greater than 24	MISS	TOTAL
349-11	N	.0022	.0152	.0159	.0068	.0005	.0001	.0009	.0414
12-33	NNE	.0033	.0163	.0110	.0029	.0007	.0001	.0007	.0350
34-56	NE	.0027	.0072	.0019	.0006	.0002	.0000	.0006	.0133
57-78	ENE	.0023	.0018	.0005	.0002	.0000	.0000	.0005	.0053
79-101	E	.0026	.0013	.0009	.0001	.0000	.0000	.0001	.0056
102-123	ESE	.0022	.0021	.0014	.0005	.0000	.0000	.0002	.0063
124-146	SE	.0027	.0056	.0061	.0009	.0000	.0000	.0001	.0154
147-168	SSE	.0025	.0130	.0138	.0054	.0001	.0000	.0002	.0351
169-191	S	.0037	.0136	.0072	.0021	.0000	.0000	.0003	.0269
192-213	SSW	.0024	.0055	.0039	.0013	.0000	.0000	.0001	.0131
214-236	SW	.0026	.0025	.0009	.0010	.0001	.0000	.0003	.0075
237-258	WSW	.0018	.0019	.0009	.0015	.0000	.0000	.0001	.0063
259-281	W	.0011	.0021	.0037	.0030	.0009	.0002	.0007	.0117
282-303	WNW	.0018	.0014	.0053	.0119	.0071	.0019	.0003	.0297
304-326	NW	.0015	.0017	.0056	.0103	.0087	.0033	.0002	.0313
327-348	NNW	.0022	.0054	.0077	.0087	.0024	.0003	.0001	.0267
CALM		.0000							.0000
MISS		.0000	.0000	.0000	.0000	.0005	.0001		.0015
TOTAL		.0377	.0971	.0865	.0568	.0211	.0062	.0065	.3119

Percentage of hours of temperature difference present in this stability category = 31.2 Numbers of hours in this stability category = 2730

Table 2.6-5 (Sheet 5 of 14)

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND
DIRECTION FOR PASQUILL STABILITY CATEGORY E

Indian Point B(3) Using a delta t correction factor of 0.605

JAN. 1 1970 1 DEC. 31 1972

WIND DIRECTION		WIND SPEED (MPH)							
		01-03	04-07	08-12	13-18	19-24	Greater than 24	MISS	TOTAL
349-11	N	.0065	.0143	.0149	.0046	.0009	.0000	.0007	.0418
12-33	NNE	.0063	.0281	.0183	.0035	.0011	.0001	.0014	.0594
34-56	NE	.0059	.0123	.0043	.0006	.0005	.0000	.0014	.0250
57-78	ENE	.0031	.0032	.0008	.0000	.0000	.0000	.0002	.0073
79-101	E	.0025	.0041	.0018	.0001	.0001	.0000	.0001	.0088
102-123	ESE	.0029	.0041	.0021	.0001	.0000	.0000	.0002	.0094
124-146	SE	.0043	.0062	.0027	.0003	.0006	.0000	.0001	.0143
147-168	SSE	.0038	.0114	.0089	.0023	.0007	.0002	.0001	.0274
169-191	S	.0041	.0162	.0102	.0014	.0003	.0003	.0001	.0327
192-213	SSW	.0049	.0101	.0077	.0008	.0000	.0000	.0000	.0234
214-236	SW	.0042	.0078	.0038	.0007	.0001	.0001	.0001	.0168
237-258	WSW	.0030	.0038	.0029	.0013	.0007	.0002	.0001	.0119
259-281	W	.0023	.0025	.0055	.0019	.0015	.0005	.0007	.0149
282-303	WNW	.0022	.0017	.0072	.0079	.0059	.0016	.0007	.0272
304-326	NW	.0013	.0030	.0098	.0120	.0043	.0010	.0009	.0323
327-348	NNW	.0024	.0071	.0089	.0070	.0010	.0000	.0022	.0286
CALM		.0000							.0001
MISS		.0009	.0009	.0010	.0008	.0013	.0000		.0062
TOTAL		.0606	.1375	.1107	.0452	.0191	.0041	.0103	.3875

Percentage of hours of temperature difference present in this stability category = 38.7 Numbers of hours in this stability category = 3391

Table 2.6-5 (Sheet 6 of 14)

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND
DIRECTION FOR PASQUILL STABILITY CATEGORY F

Indian Point B(3) Using a delta t correction factor of 0.605

JAN. 1 1970 1 DEC. 31 1972

WIND DIRECTION		WIND SPEED (MPH) 01-03	04-07	08-12	13-18	19-24	Greater than 24	MISS	TOTAL
349-11	N	.0043	.0033	.0007	.0000	.0000	.0000	.0000	.0083
12-33	NNE	.0053	.0143	.0051	.0005	.0000	.0000	.0001	.0253
34-56	NE	.0050	.0094	.0008	.0000	.0000	.0000	.0001	.0153
57-78	ENE	.0031	.0014	.0000	.0000	.0000	.0000	.0000	.0045
79-101	E	.0011	.0006	.0000	.0000	.0000	.0000	.0000	.0017
102-123	ESE	.0009	.0008	.0000	.0000	.0000	.0000	.0000	.0017
124-146	SE	.0016	.0016	.0001	.0000	.0000	.0000	.0000	.0033
147-168	SSE	.0029	.0041	.0005	.0001	.0000	.0000	.0000	.0075
169-191	S	.0022	.0043	.0007	.0000	.0000	.0000	.0000	.0072
192-213	SSW	.0031	.0058	.0006	.0000	.0000	.0000	.0001	.0096
214-236	SW	.0033	.0041	.0007	.0000	.0000	.0000	.0001	.0082
237-258	WSW	.0019	.0013	.0003	.0000	.0000	.0000	.0005	.0040
259-281	W	.0021	.0009	.0007	.0000	.0001	.0000	.0001	.0039
282-303	WNW	.0016	.0005	.0008	.0001	.0000	.0000	.0001	.0031
304-326	NW	.0017	.0006	.0007	.0002	.0000	.0000	.0000	.0032
327-348	NNW	.0024	.0015	.0006	.0001	.0000	.0000	.0000	.0046
CALM		.0000							.0000
MISS		.0002	.0000	.0000	.0000	.0000	.0000		.0011
TOTAL		.0427	.00544	.0122	.0010	.0001	.0000	.0021	.1125

Percentage of hours of temperature difference present in this stability category = 11.3 Numbers of hours in this stability category = 985

Table 2.6-5 (Sheet 7 of 14)

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND
DIRECTION FOR PASQUILL STABILITY CATEGORY G

Indian Point B(3) Using a delta t correction factor of 0.605

JAN. 1 1970 1 DEC. 31 1972

WIND DIRECTION		WIND SPEED (MPH)							
		01-03	04-07	08-12	13-18	19-24	Greater than 24	MISS	TOTAL
349-11	N	.0010	.0005	.0001	.0000	.0000	.0000	.0000	.0016
12-33	NNE	.0015	.0042	.0001	.0000	.0000	.0000	.0000	.0058
34-56	NE	.0018	.0034	.0000	.0000	.0000	.0000	.0000	.0053
57-78	ENE	.0008	.0001	.0000	.0000	.0000	.0000	.0000	.0009
79-101	E	.0010	.0002	.0000	.0000	.0000	.0000	.0000	.0013
102-123	ESE	.0005	.0002	.0000	.0000	.0000	.0000	.0000	.0007
124-146	SE	.0014	.0001	.0000	.0000	.0000	.0000	.0000	.0015
147-168	SSE	.0007	.0005	.0000	.0000	.0000	.0000	.0000	.0011
169-191	S	.0009	.0008	.0000	.0000	.0000	.0000	.0000	.0017
192-213	SSW	.0013	.0016	.0001	.0000	.0000	.0000	.0001	.0031
214-236	SW	.0016	.0005	.0000	.0000	.0000	.0000	.0000	.0021
237-258	WSW	.0009	.0002	.0001	.0000	.0000	.0000	.0000	.0013
259-281	W	.0008	.0003	.0000	.0000	.0000	.0000	.0000	.0011
282-303	WNW	.0010	.0002	.0000	.0000	.0000	.0000	.0000	.0013
304-326	NW	.0011	.0002	.0000	.0000	.0000	.0000	.0000	.0014
327-348	NNW	.0010	.0003	.0000	.0001	.0000	.0000	.0000	.0015
CALM		.0000							.0000
MISS		.0000	.0000	.0000	.0000	.0000	.0000		.0001
TOTAL		.0174	.0135	.0005	.0001	.0000	.0000	.0002	.0316

Percentage of hours of temperature difference present in this stability category = 3.2 Numbers of hours in this stability category = 277

Table 2.6-5 (Sheet 8 of 14)

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND DIRECTION FOR PASQUILL STABILITY CATEGORY A

Indian Point B(3) Using a delta t correction factor of 0.605

JAN. 1 1970 1 DEC. 31 1972

WIND DIRECTION		WIND SPEED (MPH)							
		01-03	04-07	08-12	13-18	19-24	Greater than 24	MISS	TOTAL
349-11	N	.0010	.0070	.0080	.0025	.0002	.0000	.0001	.0189
12-33	NNE	.0005	.0042	.0027	.0010	.0000	.0000	.0001	.0086
34-56	NE	.0003	.0019	.0006	.0001	.0001	.0000	.0001	.0032
57-78	ENE	.0002	.0005	.0001	.0002	.0002	.0000	.0000	.0013
79-101	E	.0002	.0005	.0002	.0002	.0000	.0000	.0000	.0011
102-123	ESE	.0001	.0002	.0005	.0002	.0000	.0000	.0000	.0010
124-146	SE	.0005	.0006	.0014	.0016	.0002	.0000	.0000	.0042
147-168	SSE	.0002	.0050	.0103	.0071	.0003	.0000	.0000	.0230
169-191	S	.0010	.0090	.0066	.0017	.0000	.0000	.0000	.0184
192-213	SSW	.0003	.0054	.0029	.0002	.0000	.0000	.0001	.0089
214-236	SW	.0007	.0024	.0017	.0003	.0000	.0000	.0000	.0051
237-258	WSW	.0007	.0013	.0016	.0002	.0000	.0000	.0002	.0040
259-281	W	.0007	.0021	.0038	.0016	.0007	.0001	.0003	.0093
282-303	WNW	.0006	.0010	.0032	.0047	.0024	.0007	.0011	.0137
304-326	NW	.0008	.0015	.0037	.0055	.0038	.0007	.0005	.0163
327-348	NNW	.0007	.0041	.0056	.0042	.0013	.0001	.0000	.0160
CALM		.0000							.0000
MISS		.0000	.0000	.0000	.0000	.0000	.0000		.0022
TOTAL		.0086	.0466	.0528	.0315	.0093	.0016	.0048	.1552

Percentage of hours of temperature difference present in this stability category = 15.5Numbers of hours in this stability category = 1358

Table 2.6-5 (Sheet 9 of 14)

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND
DIRECTION FOR PASQUILL STABILITY CATEGORY B

Indian Point B(3) Using a delta t correction factor of 0.605

JAN. 1 1970 1 DEC. 31 1972

WIND DIRECTION		WIND SPEED (MPH) 01-03	04-07	08-12	13-18	19-24	Greater than 24	MISS	TOTAL
349-11	N	.0002	.0007	.0005	.0001	.0000	.0000	.0000	.0015
12-33	NNE	.0001	.0005	.0005	.0000	.0000	.0000	.0000	.0010
34-56	NE	.0005	.0003	.0000	.0000	.0000	.0000	.0000	.0008
57-78	ENE	.0001	.0002	.0000	.0000	.0000	.0000	.0001	.0005
79-101	E	.0002	.0000	.0000	.0000	.0000	.0000	.0000	.0002
102-123	ESE	.0003	.0005	.0000	.0001	.0000	.0000	.0000	.0009
124-146	SE	.0000	.0003	.0006	.0000	.0000	.0000	.0000	.0009
147-168	SSE	.0002	.0005	.0010	.0007	.0001	.0000	.0000	.0025
169-191	S	.0005	.0008	.0006	.0005	.0000	.0000	.0000	.0023
192-213	SSW	.0000	.0002	.0006	.0000	.0000	.0000	.0000	.0008
214-236	SW	.0003	.0001	.0000	.0000	.0000	.0000	.0000	.0005
237-258	WSW	.0006	.0002	.0000	.0001	.0000	.0000	.0000	.0009
259-281	W	.0000	.0000	.0007	.0001	.0000	.0000	.0000	.0008
282-303	WNW	.0002	.0000	.0000	.0005	.0003	.0000	.0000	.0010
304-326	NW	.0001	.0002	.0003	.0002	.0003	.0003	.0000	.0016
327-348	NNW	.0001	.0001	.0006	.0002	.0000	.0001	.0000	.0011
CALM		.0000							.0000
MISS		.0000	.0000	.0000	.0000	.0000	.0000		.0000
TOTAL		.0035	.0047	.0053	.0025	.0008	.0005	.0001	.0174

Percentage of hours of temperature difference present in this stability category = 1.7 Numbers of hours in this stability category = 152

Table 2.6-5 (Sheet 10 of 14)

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND
DIRECTION FOR PASQUILL STABILITY CATEGORY C

Indian Point B(3) Using a delta t correction factor of 0.605

JAN. 1 1970 1 DEC. 31 1972

WIND DIRECTION		WIND SPEED (MPH) 01-03	04-07	08-12	13-18	19-24	Greater than 24	MISS	TOTAL
349-11	N	.0000	.0009	.0009	.0008	.0000	.0000	.0000	.0026
12-33	NNE	.0001	.0007	.0008	.0003	.0001	.0000	.0000	.0021
34-56	NE	.0002	.0005	.0001	.0001	.0000	.0000	.0000	.0009
57-78	ENE	.0001	.0000	.0002	.0000	.0000	.0000	.0000	.0003
79-101	E	.0000	.0001	.0000	.0001	.0000	.0000	.0000	.0002
102-123	ESE	.0000	.0001	.0000	.0000	.0000	.0000	.0000	.0001
124-146	SE	.0002	.0005	.0007	.0001	.0000	.0000	.0000	.0015
147-168	SSE	.0003	.0013	.0018	.0008	.0000	.0000	.0000	.0042
169-191	S	.0001	.0016	.0006	.0001	.0000	.0000	.0000	.0024
192-213	SSW	.0001	.0006	.0003	.0001	.0000	.0000	.0000	.0011
214-236	SW	.0003	.0005	.0002	.0003	.0000	.0000	.0000	.0014
237-258	WSW	.0001	.0000	.0002	.0003	.0000	.0000	.0000	.0007
259-281	W	.0000	.0002	.0005	.0008	.0001	.0000	.0000	.0016
282-303	WNW	.0005	.0002	.0002	.0013	.0011	.0005	.0000	.0038
304-326	NW	.0000	.0001	.0001	.0001	.0009	.0006	.0000	.0018
327-348	NNW	.0003	.0007	.0005	.0008	.0006	.0000	.0000	.0029
CALM		.0000							.0000
MISS		.0000	.0000	.0000	.0000	.0002	.0001		.0006
TOTAL		.0025	.0079	.0072	.0062	.0031	.0011	.0002	.0282

Percentage of hours of temperature difference present in this stability category = 2.8 Numbers of hours in this stability category = 247

Table 2.6-5 (Sheet 11 of 14)

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND
DIRECTION FOR PASQUILL STABILITY CATEGORY D

Indian Point B(3) Using a delta t correction factor of 0.605

JAN. 1 1970 1 DEC. 31 1972

WIND DIRECTION		WIND SPEED (MPH) 01-03	04-07	08-12	13-18	19-24	Greater than 24	MISS	TOTAL
349-11	N	.0023	.0149	.0182	.0066	.0008	.0001	.0014	.0442
12-33	NNE	.0037	.0166	.0120	.0035	.0006	.0001	.0014	.0378
34-56	NE	.0030	.0066	.0026	.0003	.0003	.0000	.0006	.0135
57-78	ENE	.0022	.0016	.0001	.0001	.0000	.0000	.0006	.0046
79-101	E	.0024	.0019	.0010	.0000	.0000	.0000	.0002	.0056
102-123	ESE	.0023	.0016	.0013	.0002	.0000	.0000	.0003	.0057
124-146	SE	.0025	.0055	.0047	.0006	.0002	.0000	.0001	.0136
147-168	SSE	.0022	.0114	.0103	.0032	.0000	.0000	.0003	.0274
169-191	S	.0026	.0101	.0056	.0013	.0000	.0000	.0003	.0199
192-213	SSW	.0030	.0035	.0029	.0011	.0000	.0000	.0000	.0105
214-236	SW	.0021	.0019	.0009	.0006	.0001	.0000	.0003	.0059
237-258	WSW	.0011	.0018	.0008	.0011	.0000	.0000	.0000	.0049
259-281	W	.0013	.0018	.0022	.0017	.0008	.0001	.0006	.0085
282-303	WNW	.0015	.0010	.0050	.0097	.0063	.0011	.0002	.0249
304-326	NW	.0011	.0019	.0059	.0117	.0078	.0026	.0001	.0312
327-348	NNW	.0011	.0057	.0071	.0079	.0022	.0002	.0002	.0245
CALM		.0000							.0000
MISS		.0000	.0001	.0000	.0001	.0005	.0000		.0011
TOTAL		.0343	.0881	.0806	.0498	.0195	.0043	.0072	.2838

Percentage of hours of temperature difference present in this stability category = 28.4 Numbers of hours in this stability category = 2484

Table 2.6-5 (Sheet 12 of 14)

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND
DIRECTION FOR PASQUILL STABILITY CATEGORY E

Indian Point B(3) Using a delta t correction factor of 0.605

JAN. 1 1970 1 DEC. 31 1972

WIND DIRECTION		WIND SPEED (MPH) 01-03	04-07	08-12	13-18	19-24	Greater than 24	MISS	TOTAL
349-11	N	.0046	.0096	.0073	.0022	.0006	.0000	.0002	.0243
12-33	NNE	.0035	.0178	.0117	.0019	.0009	.0001	.0006	.0366
34-56	NE	.0034	.0070	.0026	.0006	.0002	.0000	.0009	.0147
57-78	ENE	.0023	.0026	.0006	.0000	.0000	.0000	.0000	.0055
79-101	E	.0014	.0035	.0016	.0001	.0001	.0000	.0000	.0067
102-123	ESE	.0015	.0034	.0018	.0001	.0000	.0000	.0001	.0070
124-146	SE	.0035	.0043	.0021	.0002	.0003	.0000	.0001	.0106
147-168	SSE	.0025	.0078	.0066	.0016	.0005	.0002	.0000	.0192
169-191	S	.0026	.0093	.0064	.0010	.0003	.0003	.0001	.0201
192-213	SSW	.0031	.0074	.0063	.0007	.0000	.0000	.0000	.0175
214-236	SW	.0025	.0045	.0029	.0007	.0001	.0001	.0000	.0107
237-258	WSW	.0019	.0029	.0018	.0009	.0007	.0002	.0000	.0085
259-281	W	.0014	.0018	.0046	.0014	.0008	.0005	.0001	.0105
282-303	WNW	.0010	.0009	.0048	.0058	.0041	.0014	.0002	.0183
304-326	NW	.0008	.0019	.0069	.0085	.0021	.0006	.0005	.0211
327-348	NNW	.0019	.0042	.0057	.0042	.0005	.0000	.0013	.0178
CALM		.0000							.0000
MISS		.0005	.0007	.0010	.0007	.0010	.0000		.0049
TOTAL		.0385	.0897	.0746	.0306	.0122	.0034	.0051	.2542

Percentage of hours of temperature difference present in this stability category = 25.4 Numbers of hours in this stability category = 2225

Table 2.6-5
(Sheet 13 of 14)

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND
DIRECTION FOR PASQUILL STABILITY CATEGORY F

Indian Point B(3) Using a delta t correction factor of 0.605

JAN. 1 1970 1 DEC. 31 1972

WIND DIRECTION		WIND SPEED (MPH) 01-03	04-07	08-12	13-18	19-24	Greater than 24	MISS	TOTAL
349-11	N	.0040	.0026	.0026	.0007	.0000	.0000	.0000	.0099
12-33	NNE	.0040	.0134	.0075	.0008	.0002	.0000	.0000	.0259
34-56	NE	.0031	.0082	.0017	.0000	.0000	.0000	.0005	.0135
57-78	ENE	.0016	.0013	.0002	.0000	.0000	.0000	.0000	.0031
79-101	E	.0013	.0003	.0001	.0000	.0000	.0000	.0000	.0017
102-123	ESE	.0013	.0007	.0000	.0000	.0000	.0000	.0000	.0019
124-146	SE	.0009	.0014	.0002	.0000	.0000	.0000	.0000	.0025
147-168	SSE	.0022	.0041	.0015	.0001	.0002	.0000	.0000	.0081
169-191	S	.0022	.0071	.0029	.0002	.0000	.0000	.0000	.0123
192-213	SSW	.0021	.0056	.0014	.0000	.0000	.0000	.0000	.0090
214-236	SW	.0027	.0049	.0009	.0000	.0000	.0000	.0001	.0087
237-258	WSW	.0021	.0013	.0009	.0000	.0000	.0000	.0005	.0047
259-281	W	.0011	.0010	.0011	.0001	.0003	.0000	.0007	.0045
282-303	WNW	.0010	.0008	.0022	.0009	.0001	.0001	.0006	.0057
304-326	NW	.0011	.0007	.0022	.0006	.0009	.0001	.0005	.0061
327-348	NNW	.0013	.0016	.0024	.0016	.0001	.0000	.0008	.0078
CALM		.0000							.0001
MISS		.0007	.0001	.0000	.0000	.0000	.0000		.0013
TOTAL		.0326	.0552	.0279	.0050	.0019	.0002	.0040	.1268

Percentage of hours of temperature difference present in this stability category = 12.7 Numbers of hours in this stability category = 1110

Table 2.6-5 (Sheet 14 of 14)

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND
DIRECTION FOR PASQUILL STABILITY CATEGORY G

Indian Point B(3) Using a delta t correction factor of 0.605

JAN. 1 1970 1 DEC. 31 1972

WIND DIRECTION		WIND SPEED (MPH) 01-03	04-07	08-12	13-18	19-24	Greater than 24	MISS	TOTAL
349-11	N	.0027	.0025	.0003	.0000	.0000	.0000	.0000	.0056
12-33	NNE	.0047	.0125	.0014	.0000	.0000	.0000	.0001	.0186
34-56	NE	.0053	.0089	.0000	.0000	.0000	.0000	.0000	.0142
57-78	ENE	.0029	.0008	.0000	.0000	.0000	.0000	.0000	.0037
79-101	E	.0018	.0007	.0000	.0000	.0000	.0000	.0000	.0025
102-123	ESE	.0009	.0009	.0000	.0000	.0000	.0000	.0000	.0018
124-146	SE	.0024	.0014	.0000	.0000	.0000	.0000	.0000	.0038
147-168	SSE	.0023	.0023	.0002	.0000	.0000	.0000	.0000	.0048
169-191	S	.0021	.0027	.0005	.0000	.0000	.0000	.0000	.0053
192-213	SSW	.0033	.0040	.0003	.0000	.0000	.0000	.0002	.0079
214-236	SW	.0034	.0026	.0002	.0000	.0000	.0000	.0001	.0064
237-258	WSW	.0017	.0008	.0003	.0000	.0000	.0000	.0001	.0030
259-281	W	.0024	.0006	.0001	.0000	.0000	.0000	.0000	.0031
282-303	WNW	.0022	.0003	.0001	.0000	.0000	.0000	.0000	.0026
304-326	NW	.0021	.0005	.0000	.0001	.0000	.0000	.0000	.0026
327-348	NNW	.0026	.0008	.0001	.0001	.0000	.0000	.0000	.0037
CALM		.0000							.0000
MISS		.0000	.0000	.0000	.0000	.0000	.0000		.0008
TOTAL		.0427	.0423	.0037	.0002	.0000	.0000	.0014	.0903

Percentage of hours of temperature difference present in this stability category = 9.0 Numbers of hours in this stability category = 790

2.7 GEOLOGY [Historical Information]

Indian Point 3 is located approximately two miles southwest of the city of Peekskill, Westchester County New York, on the east bank of the Hudson River. Geologically, it is located in the central part of the Peekskill Quadrangle. The complete geologic description is divided into two broad sections:regional geology, physiography and tectonics; and geology of the area surrounding the site.

2.7.1 Regional Geology, Physiography and Tectonics

The general landscape of the region (see Figure 2.7-1 consists of bedrock-supported ridges following generally northeasterly structural trends and rather steep and broad swampy valleys. The highest elevation in the region is 1,000 ft, and elevations range from 50 to 300 ft above mean sea level in low-lying areas. At the plant site the ground is level, about 15 feet above sea level and is covered with fill. The surface is artificially leveled and bedrock lies very close to the surface.

The eastern part of the United States has gone through tectonism since the Precambrian age (Figure 2.7-2 and is known as the Appalachian Orogen. The plant is situated within the Manhattan Prong of the Appalachian Mountains. It is estimated that the earliest tectonic activity in the Appalachian Orogen was in Precambrian age and was a result of continental rifting and associated intrusive activity. A striking characteristic of the region is the high degree of metamorphism exhibited by the rocks. This has resulted from their long and complex history (Precambrian through the mid- Ordovician time) which included extensive thrust faulting, folding, instrusion, etc. The Taconic Orogeny was intense in the Manhattan Prong region and produced most of the structures evident in the map today. Essentially, the rocks in the plant site area belong to three tectonic provinces, e.g., the Hudson Highlands, the Manhattan Prong and the Newark Basin. The geology of these provinces follows:

The Hudson Highlands

The Hudson Highlands are a part of the much larger Blue Ridge - New Jersey Highlands Province. Here the northeast trending ridges are underlain by complexly folded granitoid gneisses and schists. These also involve granodioritic intrusives. Prevailing dips in the entire region are steep towards the southeast. The bulk of the Highland rocks represent a sequence of Precambrian aged miogeosynclinal and engeosynclinal deposits, however those in the areas of concern are in faulted and in-folded strata of Cambro - Ordovician age.

Helenock and Mose[(2)] recognized a mappable sequence of five rock units in the Lake Carmel, New York, area of the Highlands. These rocks were metamorphosed to granulite facies, and were multiply deformed in the Greenville Orogeny. There was recrystallization to amphibolite facies accompanied by folding during the Taconic Orogeny (mid-Ordovician).

The Ramapo Fault Zone (Section 2.8) separates the Highlands from the Manhattan Prong and the Newark Basin.

The Manhattan Prong

The Manhattan Prong is bounded on the east by Cameron's line, on the west by the Newark Basin border fault and the Hudson River. It covers the geographic areas of New York City (Manhattan), Westchester County, New York and parts of Fairfield County, Connecticut.

The uppermost formation of sedimentary origin is called a Phyllite or Schist known as the Manhattan Schist. This is the most recent geologic formation. In order of increasing age and depth are the Inwood Marble, the Lowerre Quartzite, the Yonkers-Pound Ridge Granite and the Fordham Gneiss. Due to the extremely complicated nature of the region's geology this stratigraphy varies with location.

The Manhattan Formation was deposited in a miogeosyncline. It was metamorphosed, deformed and intruded during the Taconic and the Acadian episodes(3).
The Inwood Marble, consisting of dolomite and calcite marbles with interlayed calc - silicate schists, were deposited during the Cambrian - Ordovician period. It is widespread in the Appalachian Orogen.

The Lowerre Quartzite underlies the Inwood Marble. It is a relatively thin, discontinuous unit representing an arkosic sandstone. The Lowerre consists mainly of quartz with potassium feldspar and biotite. It is always found underlying the Cambro-Ordovician aged rocks.

The Yonkers and Fordham formations are Precambrian in age and are separated from the Lowerre. Inwood and Manhattan formations are joined by an angular unconformity.The Fordham formation was deformed and metamorphosed to granulite facies during the Greenville Orogeny. The Yonkers - Pound Ridge Granite, emplaced during the opening of the Proto-Atlantic in late Precambrian age, is mostly a metamorphosed rhyolite.

The Newark Basin

The Newark Basin formation, west of the Hudson River, extends from York County, Pennsylvania to Rockland County, New York. The northern tip of this basin very closely approaches the Indian Point Site on the opposite side of the Hudson River near Stony Point. This is an assemblage of conglomerates, sandstones and shales with their intercalated beds of basaltic lava and the well known intrusive sill of the "Palisades". Deposition was continuous from the late Triassic through the upper Jurassic ages(4). The boundary fault between this basin and older crystalline rocks is the well known Ramapo Fault.

2.7.2 Geology of the Area Surrounding the Site

The geology of the area surrounding the site is shown in Figure 2.7-3. The Ramapo Fault System passes through the area surrounding the site. The Ramapo is a series of N - NE trending faults, with minimum age for movement being Greenville (5). The faults surrounding the plant site have been studied utilizing radiometric age determination, cross-cutting lighologic relationships and textural evidences, by Ratcliff (5); Dames & Moore(4) studied age based on geothermometry of fluid inclusions in calcite.

The most prominent faults that separate the Manhattan Prong from Hudson Highlands are the Thiells fault, the Annsville fault, the Peekskill fault and the Croton Falls fault. These are all believed to be of Paleozoic age. The N-S faults at Tomkins Cove across the Hudson River from Indian Point are the youngest in the area, presumably of Masezoic age. Dames & Moore (4) mapped a group of faults at the Indian Point site with displacements no more than a few feet. They are filled with undeformed euthedral calcite crystals (Figure 2.7-4)

Radiometric age determination of these, and the lack of fault related deformation of Pleistocene deposits and surface features, prove that the faults surrounding the site have not moved in the last 2 million years, although predominant movements took place in Precambrian time. Examination of recent core drills in the area indicate that the dip is consistently to the S-E and that the dominant latest motion in the fault was right oblique normal faulting.

References

1) Van Eysing, F. W. B, 1978 Geologic Time Scale

2) Heleneck, H. L. & Mose, D. G. 1978, Geology and Geochronology of Precambrian Rocks in the Lake Carmel Region Hudson Highlands, N. Y., Geological Society of America, V. 10. No. 2.

3) a) Brock, P.W.G and Mose, D. G., 1979, Taconic and Younger Deformations in the Croton Falls Area, S-Eastern N.Y., Bulletin of Geo. Soc of Am. Pt II V. 90.
b) Mose, D G. and Hall, L. M., 1979, Rb - Sr Whole-Rock Age Determination of Member C of the Manhattan Schist and its Bearing on Allochthony in the Manhattan Prong, SE New York Geo. Soc. of Am, Abstracts with Programs, V. II No. 1.

4) Dames & Moore, 1977, Geotechnical Investigation of the Ramapo Fault System in the Region of the Indian Point Generating Station.

5) Ratcliffe, N. M., 1976, Final Report on Major Fault Systems in the Vicinity of Tomkins Cove - Buchanan, N. Y. Report for Consolidated Edison Company, Inc. of N. Y.

6) Dames & Moore, Nov. 1975. Supplemental Geological Investigation of the Indian Point Generating Station for Consolidated Edison Co. of N. Y. Inc.

2.8 SEISMOLOGY [Historical Information]

2.8.1 Background and Seismic Design Bases

Geographic areas of the continental United States have been subdivided into regions of known or assigned seismic probability or risk and this has served as a useful basis for generating code provisions for earthquake - resistant structures. The Seismic Risk Map adopted by the International Conference of Building Officials for inclusion in the 1970 edition of Uniform Building Code, divides the United States into four (4) major zones of seismic risk or probability. The Indian Point Site is located in Zone I of this map with intensities limited to V and VI on the Modified Mercalli Intensity Scale of 1931 (Figure 2.8-1) and only slight earthquake activity can be expected.

However, the Indian Point 3 facility was actually built per requirements of Zone 2 of the Uniform Building Code i.e., corresponding to an intensity VII of the Modified Mercalli Scale. The range of expected horizontal acceleration of ground motion for earthquakes of this intensity is 70-150 cm/sec2 near the epicenter or about 0.15 g max. At a distance of 100 miles from the epicenter, the acceleration drops to 50%. The nearest event larger than intensity VII occurred near Cape Ann, Massachusetts, a distance of more than 200 miles from the site, in 1755. This event was classified as intensity VIII on the Modified Mercalli Scale. It was believed, therefore, that the plant's structural design, allowing for safe shutdown in the event of an earthquake of intensity VII on the Modified Mercalli Scale, was adequate. A list of known earthquakes which have occurred in the vicinity of the plant having intensities of V through VII on the modified Mercalli Scale is provided by Table 2.8-1.

The Reverend Joseph Lynch, S. J., while Director of the Fordham University Seismic Observatory stated:

> ".... that the probability of a serious shock occurring in this area for the next several hundred years is practically nil. The area therefore would certainly seem to be as safe as any area at present known."

Captain Elliott B. Roberts, while Chief of the Geophysics Division of the Department of Commerce, substantially agreed with the conclusions of Rev. Lynch.

Rev. Lynch also stated that the "estimated maximum ground acceleration of 0.03 g is reasonably conservative for the area." This has been established as the basis for design of the plant. Rev. Lynch stated further that the "safety factor for a horizontal stress of 0.1 g is therefore.... more than adequate. "For earthquakes having a horizontal acceleration of 0.1 g and a vertical acceleration of 0.05 g acting simultaneously at zero period, the plant is designed to have no loss of function of systems important to safety, although in some cases, the stresses may reach or slightly exceed yield points.

2.8.2 Public Concerns and Resolutions

Subsequent to the plant's construction, public concerns were raised on the following issues:

1) A series of N-NE trending faults pass through the area surrounding the site - collectively known as the Ramapo Fault System (Section 2.7). The concern raised was whether the Ramapo Fault is "capable" of causing an earthquake at the site.(1)
2) Because of the lack of historical records of earthquakes (Table 2.8-1) in the plant area - older than the Cape Ann earthquake, concerns were raised whether the safe shutdown earthquake (SSE) for the plant's design should be greater than intensity VII on the Modified Mercalli Scale.
3) As stated earlier, the plant was designed for 0.15 g max base shear for safe shutdown. Concern was that, if the SSE ground acceleration be raised from intensity VII to VIII on the Modified Mercalli Scale, would the 0.15 g base shear still be adequate?
4) An extended micro-monitoring system for measuring magnitude, accurately determining the location and even focal mechanism behavior of small magnitude earthquake near the plant and Ramapo Fault Zone was required by condition 2.C.4(c) of Amendment No. 2 to the Operating License of the plant. Concern was raised whether this extended micro-seismic-measuring instrumentation was considered as a licensing requirement.

An 18 month proceeding was held on the above concerns before a U.S. Nuclear Regulatory Commission (NRC) Atomic Safety and Licensing Board. Also, there have been numerous extensive studies on these concerns. The findings by the board and these studies may be summarized as follows:

1) In testimony before the board, Charles F. Richter, who developed the Richter Scale, stated that the earthquakes in the Ramapo region are "of minor magnitude and relatively trivial". Radiometric age determination of undeformed minerals that have grown within fault zones was studied by Ratcliffe(2) and fault related deformation of Pleistocene deposits and surface features by Dames & Moore(3) and Ratcliffe(5). Both prove that the faults in the Indian Point area have not moved in at least the last 2 million years. The Ramapo Fault, therefore, is considered to be old, inactive and not a "capable" fault under Appendix A to 10 CFR 100.

2) The unanimous ruling by the board was - "In accordance with Appendix A to 10 CFR 100, neither the Cape Ann earthquake nor any other historic event requires the

assumption of a safe shutdown earthquake for the Indian Point Site of greater than a Modified Mercalli intensity of VII(6)". Hearings were also held before the Advisory Committee on Reactor Safeguards. Despite some controversy over appropriate tectonic divisions, the Committee, scientists in the TERA Corp.(8)(9) and Dames & Moore(4) concluded that an event of intensity VII on the Modified Mercalli Scale is adequate as the design earthquake for the Indian Point Site.

3) Consistent with the above ruling, the board also ruled - "The ground acceleration value used for the design of Indian Point Units 2 and 3 should remain at 0.15 g".(6)

4) Amendment 2 to the Technical Specifications stated in Section 2(c)(4)(c) that an extended microseismic instrumentation network must be operated for at least two years following complete installation of all stations. The Atomic Safety and Licensing Appeal Board repealed this decision in hearings held on October 12, 1977(6) and the NRC issued Technical Specification Amendment 9 to reflect this. However, a network was operated from 1975-1990 and a final report(10) was published.

2.8.3 Conclusions

Microseismic activity recorded by the seismic monitoring network is evidence of minor crustal adjustments due to regional stresses. However, neither the readings from the network nor the bore-hole experiment(11) at Kent Cliffs show the evidence of any contemporary (geologic) movement along faults exposed at the surface as was suggested by Aggarwal and Sykes.(1) On the contrary, the last movement of the region (Mesozoic Period) was in a direction normal to the proposed direction.

It is therefore concluded that the seismic design criteria for structural analysis at the Indian Point site is satisfactory and that the plant is set on solid bedrock. No public hazard can be expected from the plant due to a probable earthquake in the region.

References

1) Aggarwal, Y. P. and Sykes, L. R., 1978, Earthquakes, Faults and Nuclear Power Plants in Southern New York and Northern New Jersey in "Science," V. 200.

2) Ratcliffe, N. M. 1976, Final Report on Major Fault Systems in the Vicinity of Tomkins Cove - Buchanan, New York: Report for Consolidated Edison Company of New York, Inc.

3) Dames & Moore, 1977, Geotechnical Investigation of Ramapo Fault System in the Region of the Indian Point Generating Station.

4) Dames & Moore, 1980, Seismic Ground Motion Hazard at Indian Point Nuclear Power Plant Site - A report prepared for Pickard, Lowe & Garrick, Inc.

5) Ratcliffe, N. M. 1981, Brittle Faults (Ramapo Fault) and Phyllonitic Ductile Shear Zones in the Basement Rocks of the Ramapo Seismic Zone, New York and New Jersey, and their Relationship to Current Seismicity published in: Manspeizer, W., editor, Field Studies of New Jersey Geology and Guide to Field Trips, 52nd Annual Meeting of the New York State Geological Association, Newark, New Jersey, Rutgers University.

6) United States Nuclear Regulatory Commission/Atomic Safety and Licensing Appeal Board, Farrar, M. C. - Chairman, Buck, J. H. and Quarles L. R. - Members at a Hearing cited as 6 NRC 547 (1977), ALAB -436.

7) Gutenberg, B. and Ritchter C. F. Earthquake Magnitude, Intensity, Energy and Acceleration, BSSA, 32,(3) July 1942.

8) TERA Corporation, Seismic Hazard Analysis - A Methodology for the Eastern United States, NUREG/CR-1582, 2, 1980.

9) TERA Corporation, Seismic Hazard Analysis Solicitation of Expert Opinion, NUREG/CR-1582, 3, 1980.

10) Woodward-Clyde; Scientific Results of Seismic Monitoring Network near the Indian Point Nuclear Generating Facilities, Final Report (10/27/92); R&D Project 92284

11) Woodward-Clyde, 1986: Kent Cliffs Bore-hole Research Project: A Determination of the Magnitude and Orientation of Tectonic Stress in Southeastern New York; Research Report EP 84-27, Empire State Electric Energy Research Corporation, New York, New York.

TABLE 2.8-1

LIST OF EARTHQUAKES OF INTENSITIES GREATER THAN OR EQUAL TO V ON MODIFIED MERCALLI SCALE
(SOURCE: EPRI)

DATE	TIME (HOUR)	GEOGRAPHIC COORIDINATES LATIITUDE (DEG. N)	GEOGRAPHIC COORIDINATES Longitude (Deg. W)	DEPTH (KM)	INTENSITY MODIFIED MERCALI SCALE	REMARKS
12-19-1737	04:00:00	40.80	74.00	0	VII	
11-30-1783	03:50:00	41.00	74.50	0	VI	
09-29-1847	00:00:00	40.50	74:00	0	V	Felt Over 2,000 Sq. Miles
12-11-1874	03:25:00	40.90	73.80	0	V	
10-04-1878	07:30:00	41.50	74.00	0	V	
08-10-1884	19:07:00	40.60	74.00	0	VII	Felt Over 31,000 Sq. Miles
09-01-1895	11:09:00	40:70	74.80	0	VI	
06-01-1927	12:23:00	40.30	74.00	0	Between VI and VII	
10-08-1952	21:40:00	41.70	74.00	0	V	
03-23-1957	19:02:00	40.60	74.80	0	VI	
11-17-1964	17:08:00	41.20	73:70	0	V	
11-21-1967	22:10:00	41.20	73.80	0	V	
03-11-1976	21:07:00	41.00	74.40	0	V	
04-13-1976	15:39:00	40.80	74.00	0	VI	
01-30-1979	16:30:52	40.32	74.26	5	VI	
03-10-1979	04:49:40	40.72	74.72	3	V	
12-30-1979	14:15:12	41.14	73.69	5	V	

(For location Map see Figure No. 2.7-3)

2.9 ENVIRONMENTAL MONITORING PROGRAM

2.9.1 General

A program to determine the environmental radioactivity in the vicinity of Indian Point Station was instituted in 1958, four years prior to the initial operation of Consolidated Edison's Indian Point Unit No. 1. The purpose of this survey was to determine the natural background radioactivity and to show the variations in the activities that may be expected from natural sources, fallout from bomb tests and other sources in the vicinity. This program has been continued to the present so that changes in the environment, resulting from station operations, could be accounted for. The results of these surveys are reported annually to the Nuclear Regulatory Commission.

In addition, the New York State Department of Health has conducted surveys throughout the State of New York since 1955, including extensive surveys in the vicinity of the Indian Point Station since 1958. In 1965 and 1966, they reported the findings in the vicinity of the Indian Point Station in two special reports. Since that time, their reporting has been on a statewide basis in quarterly bulletins and in annual reports.

In 1964, the New York University Medical Center began a research program on the ecology of the Hudson River. The New York University studies include the biology of the Hudson River, the distribution and abundance of fish in the river, pesticides and radio-ecological studies. The results of this program, supported by the United States Public Health Service, the New York State Department of Health, and the Consolidated Edison Company have been submitted in several program reports.

The various studies mentioned above included measurements of radioactivity in fresh water, river water, river bottom sediments, fish, aquatic vegetation, soil, vegetation and air in the vicinity of the Indian Point Station. The results of these monitoring programs have shown that the operation of the Indian Point Units 1, 2, and 3 have had no deleterious effects on the environment.

2.9.2 Survey Programs

The survey of environmental radioactivity in the vicinity of Indian Point Station provides an indication of the integrity of the in-plant radiation monitoring instrumentation and can reveal any buildup of long lived radionuclides.

By determining the activity of filterable air particulate, vegetation, drinking water and above ground gamma fields, an indirect monitoring of discharges to the atmosphere is provided by the environmental survey program.

The effect of liquid effluents on the Hudson River is monitored by measuring the activity of the cooling water inlet to and discharge from the station, discharges from the plant, activity analysis of river shoreline soils and river fish and invertebrates.

A detailed description of the media sampled in accordance with plant Environmental Monitoring Program and the ODCM is given below:

Air Particulate and Organic Iodide

Concentration of radioactive particles in the air is measured weekly from 5 stations.

Membrane filters precede charcoal impregnated filters. The particulate filters are assayed for gross beta activity and are composited for quarterly gamma spectral analysis. Charcoal filters have gamma spectral analysis for I-131 performed weekly.

Reservoir Water

Drinking water is sampled monthly from an area reservoir. The water sampled is analyzed for gross beta activity, and for other nuclides via gamma spectral analysis. A quarterly composite sample is analyzed for tritium.

Hudson River Water

Continuous flow samples of the condenser inlet cooling water and discharge water are collected and composited. Samples are taken, at a frequency specified in the ODCM, from continuous samples and composited for a monthly gamma spectroscopy analysis, and for a quarterly tritium analysis.

Hudson River Shoreline Soil

Twice a year, at least 90 days apart, samples of river shoreline soil are taken at two locations. Gamma spectral analysis is performed on each sample.

Hudson River Fish and Shellfish

Fish and invertebrates are caught seasonally (semi-annually if not seasonal) where available near the site and analyzed by gamma spectral analysis.

Vegetation

Samples of broad leaf vegetation are collected monthly, if available, in the critical wind sections within several miles of the plant. Gamma spectral and Iodine-131 analyses are performed on these samples.

Milk

Milk samples are obtained, when available, on a monthly basis (semi-monthly when animals are on pasture) from dairy farms, located within 5 miles of the site. The samples are analyzed for Iodine-131 content, and for other nuclides by gamma spectral analysis.

Direct Gamma (Continuous)

At 40 locations near the site and out to about 5 miles, the background gross gamma radiation is continuously monitored. The measuring devices consist of two sets of thermoluminescent dosimeters (TLDs). The TLDs are removed at quarterly intervals and the amount of absorbed background radioactivity is recorded.

2.9.3 Summary

The environmental monitoring program conducted by Entergy supplies sufficient data to determine the compliance of Indian Point Unit Nos. 1, 2 and 3 with the requirements of 10 CFR 20. The environmental survey program which monitors air, water, river shoreline sediments, terrestial vegetation, milk and selected aquatic biota provides an indication of the cumulative amounts of radioactivity in the environment.

Results of the environmental monitoring program are reported on an annual basis to the nuclear Regulatory Commission, who are thereby advised of the short and long-term trends in the environment. In addition, discharges of radioactive liquids and gases are reported to the Nuclear Regulatory Commission.

In the event that the Indian Point Station Environmental Monitoring Program detects increases in the background radiation levels above the reporting levels specified in the Offsite Dose Calculation Manual (ODCM), Entergy will notify the Nuclear Regulatory Commission.

Although the design of Indian Point 3 and administrative controls are such that liquid and gaseous effluents are released in accordance with the requirements of 10 CFR 20, the environmental monitoring program conducted by IP3 and IP2 provides a redundant means of insuring that the operation of this facility does not pose any undue risk to the health and safety of the public.

An Analysis

of the Con Edison and AEC-DRL

Accident Meteorology Models

as Applied to the Indian Point Site

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Summary

The conservatism of the Con Edison and AEC-DRL accident meteorology models has been investigated for the initial 0-2 hr, 0-8 hr and 0-24 hr periods following the inception of a postulated leak through the containment structure.

Values of å c/Q for each model were compared to values calculated for 15 observed wind sequences which represented the longest persistences of lightest winds under inversion conditions in a two-month period. In computing the å c/Q for each observed hour, the Con Edison system of assigning stability classifications and selecting diffusion model equations, as described in the Unit 2 FSAR, was employed.

A tabulation of the results may be found in table 7. Roughly speaking, both models gave the same results. The 0-2hr å c/Q could expected to be exceeded about 1% of the time, the 0-24 hr å c/Q would never be exceeded. Stated another way, the factor of safety at the 1% probability level was about 1 for the 0-2 hr period and 2 for the 0-24 hr period. At the 5% probability level the factors of safety were about 10 for the 0-2hr period and 3 for the 0-24 period. These values varied by about ± 25%, depending upon distance from the source.

Additional conservatism, apart from the ratios of å c/Q cited above, was found to exist in the models, but did not show up in the ratios because the same assumption was used for both the accident models and the observed sequences. This had to do with enhanced lateral diffusion within the Indian Point building complex. Wind tunnel tests were cited to show that lateral diffusion corresponding to about Pasquill C stability, coupled with restricted vertical diffusion,

2.6.L-6

would accurately predict measured concentrations in the wind tunnel. Also, bivane measurements at Indian Point under low speed inversion conditions showed that lateral turbulence increased as vertical temperature gradients increased from 0 to about 6^{o} C/100 m (3^{o} F/88 ft), while vertical turbulence was largely unaffected. A diffusion calculation based on the bivane measurements showed that the maximum value of å $\bar{u}/Q$ would occur at a vertical temperature gradient of about 1^{o} C/100 m (0.5^{o} F/88 ft), and the corresponding Pasquill stability class at that gradient would be C-D.

Therefore, it appears that the Pasquill F or Con Edison Inversion categories in the accident models should be replaced by Pasquill D or Con Edison Neutral, if the indications of the wind tunnel and field bivane measurements are to be taken as representing the true diffusion condition. Retaining the Inversion category rather than changing to the Neutral in the Con Edison model introduces a factor of safety of about 7 at the 520 m distance.

A detailed analysis of meteorological experiments conducted in 1969 and 1970 at Indian Point to assess the long term variability of annual wind statistics and the details of wind persistence and diurnal reversal of direction confirmed that the 1956 data, used as the basis for the Con Edison accident meteorology model, are still valid today.

2.6.L-7

Introduction

The report reviews and analyzes the meteorological model used in dose calculations for a postulated loss of coolant accident at the Indian Point site, with respect to questions raised by AEC staff meteorologists and consultants and by the AEC Atomic Safety and Licensing Board. Some of these questions have been answered in writing, some were answered orally during the Indian Point Unit 3 Construction permit hearings, and some could not be answered then owing to the lack of on-site experimental data, but can be answered now with data from meteorological experiments performed at Indian Point subsequent to the Unit 3 hearings.

In this report, the questions which appear to require further response are extracted from the record, and the answers are clarified or supplemented as indicated.

Because the documents in this record were filed over a 17 year period in a number of proceedings, a chronological review has been provided in Appendix A for the convenience of the reader.

Frequent reference will be made to the ConEdison and AEC-DRL accident meteorology models. These are presented in Tables 1 and 2.

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1. Conservatism of the ConEdison and AEC-DRL Accident Meteorology Models As Applied to Indian Point.

1.1 General Comments on Quantification of Conservatism

The application of conservatism in the choice of an accident meteorology model involves two concepts; one is probability, the other is factor of safety.

If one anticipates that a series of events of varying severity will occur, and can estimate the probabilities of their occurrences, then one can develop a functional inverse relationship between severity and probability, select an acceptably low probability, and design for the indicated severity.

There may, however, be uncertainty about the magnitudes of various quantities which enter into the specification of the indicated severity, and it may be desirable to design for an arbitrarily greater severity to provide a reserve against this uncertainty. The ratio of the arbitrarily greater severity to the severity dictated by the specified probability may be termed the factor of safety, or often, the factor if ignorance.

The selection of this arbitarily greater severity may be based on the selection of a lower specified probability to yield a higher indicated severity, but, as frequently happens, sufficient data may not be available to allow extrapolation in this direction. One may then postulate a series of increasingly severe events, each having an intuitively or demonstrably zero probability of occurrence, and base the estimate of degree of conservatism on the magnitude of the factor of safety. This approach carries the implication that an event with the indicated severity will occur at some time, and that the arbitrarily chosen factor of safety will harden the design specifications to compensate for the inherent uncertainties in the design parameters.

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2.2 Data Analysis

In the use of Con Edison and AEC-DRL accident meteorology models, it will be shown that according to past records, neither model has ever occurred in its 30-day entirety. Also, by considering the physics of fluid motions over irregular, non-isothermal terrain, it seems likely that the probability of their ever occurring is virtually zero. However, there is a finite probability that the initial stages of the models will occur. Much of the questioning on meteorology during the IP 3 Construction License hearings was directed toward ascertaining the probabilities and factors of safety during these initial stages, but the answers were inconclusive.

In the following sections, attention will be focused on the 0-2 hr, 0-8 hr, and 0-24 meteorological sequences. The contribution of the 1-30 day sequence to the total dose is so small as not to warrant a detailed investigation.

The general procedure will be to calculated from observed data a curve of the total dilution factor $\Sigma X/Q$ vs cumulative probability of occurrence for 0-2 hr, 0-8 hr and 0-24 hr meteorological sequences, and to calculated the values of $\sum X/Q$ according to the Con Edison and AEC-DRL models for the same time periods. From these computations, two estimates of conservatism will be presented:

a) the probability of occurrence of each of the model predictions,

and

b) the factor of safety of each of the model predictions, using as a reference the $\Sigma X/Q$ vale corresponding to arbitrary indicated low probability levels of 1% and 5%.

The techniques used in the calculations, and important results, are presented in the following sections.

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2.21 The Con Edison Accident Diffusion Model

For each hour of the first 24 hours, the dilution factor χ/Q is given by Eq 8, Pg 14.3.5-9 of the Unit 2FSAR[1], with y = 0:

$$\frac{\chi}{Q} = \frac{2}{\pi C_y C_z (X+X_o)^{2-n}\bar{u}} \qquad \text{(steady wind model)} \qquad (1)$$

where

$$\left.\begin{aligned} C_y &= 0.40 \ m^{n/2} \\ C_z &= 0.07 \ m^{n/2} \\ n &= 0.5 \end{aligned}\right\} \text{inversion stability}$$

$\bar{u}$ = 1 m/s for first 2 hours

= 2 ms for next 22 hours

X = 350 m for Unit 3 site boundary

= 520 m for Unit 2 site boundary

= 1100 m for Units 2 and 3 low population zone

X_o = $(A/8C_yC_z)^{1/(2-n)}$ = 430 m when A = 2000 m^2

For each sequence in the model, the values of χ/Q for each hour calculated by Eq (1), may be summed to yield å X / Q for the sequence. Sequence values of å X / Q are given in Table 3.

2.22 The AEC-DRL Accident Diffusion Model

For each hour of the first 8 hours, the dilution factor χ/Q is given in AEC-DRS Safety Guide 4[(2)] as

$$\frac{\chi}{Q} = \frac{1}{(\pi \ \sigma_y \ \sigma_z + cA)\ \bar{u}} \qquad \text{(steady wind model)} \qquad (2)$$

With the restriction that ¶ $\sigma_y \sigma_z$ + cA must not exceed 3 ¶ $\sigma_y \sigma_z$. Under conditions of F stability and cA = 1000 m^2, this restriction becomes effective at distances between 0 and 500 meters from the source.

In Eq (2),

$(\sigma_y \sigma_z)$ = 83 m^2 at X = 350 m

165 m^2 at X = 520m

570 m^2 at X = 1100m

in F stability

$\bar{u}$ = 1 m/s

c = 0.5

A = 2000 m^2

For the 9th through the 24th hours, the dilution factor is given by

$$\frac{X}{Q} = \frac{\sqrt{2/p}}{S_z X \quad b} \qquad \text{(meandering wind model)} \qquad (3)$$

where σz = 6.4 m at X = 350 m

8.8 m at X = 520 m

15.0 m at X = 1100 m

β = 22.5/57.3 = 0.393

Values of $\sum \chi/Q$ are given in Table 3.

2.23 Values of Hourly Average χ/Q from Observed Data

The data used for this calculation are contained in Figs 10-11 of N.Y.U. Report TR 71-3[3].

The meteorological parameters cited therein are as follows:

© $\bar{u}$ = wind speed (m/s)

θ = wind direction (deg from N)

ΔT = T_{95}-T_7 (^oF), temperature difference between the 95 and 7 ft elevations above the base of the meteorology tower IP3 whose location is shown in Fig 1 of the above report.

The calculation of χ/Q or each hour was performed using Sutton Diffusion models with diffusion parameters given on Pg 14.3.5-9 of the Unit 2 FSAR. Two diffusion models are involved; one is the steady wind model described by Eq (1). The other is a meandering wind model similar to Eq (3) but written in the Sutton form and including a wide-plume factor. This wide-plume factor has not appeared in the various Con Edison submittals to the AEC-DRL. Its omission leads to illogical concentration predictions for meandering plumes, noted in the ESSA-AREL Comment on Pg 95 in Appendix C to the AEC-DRL Safety Evaluation of Unit 2[4].

The meandering wind model used in this report is that shown on Pg Q11.10-1 of the Unit 2 FSAR multiplied by the wide-plume factor. Under conditions such that the plume centerline meanders uniformly within the sector boundaries during one entire hour, the equation for the hourly average χ/Q (using present notation) may be written as

$$\frac{X}{Q} = \frac{2}{\beta\sqrt{\pi}} \frac{W}{\bar{u} C_z X (X + X_O)^{(1-n/2)}} \qquad (4)$$

where

$$W = \frac{1}{\sqrt{2\pi}} \int_{-P}^{+P} exp\left(-p^2/2\right)dp$$

and $P = X\beta/2\sigma_y = x\beta/\sqrt{2}\,C_y(X + X_O)^{1-n/2}$

Values of W may be found in Fig A-4 of "Workbook of Atmospheric Dispersion Estimates" by D. B. Turner[5] for corresponding values of P.

The factor W has a value of unity when the plume width is less than the sector width. For example, assuming that the plume width is equal to $5\sigma_y$,

2.6.L-13

and the sector width is equal to Xβ then P is greater than 2.5 when $5\sigma_y$ is less than Xβ. From Turner's Workbook, W is seen to have a value of .987 at P = 2.5 and to approach unity as P increases (corresponding to a narrowing plume).

When the plume is wider than the sector, P becomes less than 2.5 and W becomes increasingly smaller as the plume width increases.

The omission of W from published versions of the sector-averaged diffusion equation carries with it the implication that the plume width is smaller than the sector width. This is often true for inversions, particularly at large distances from the source, but it is generally not true for volume sources with wakes in neutral and lapse conditions.

Values of P and W for various stability and distances are given below:

X(m	350			520			1100		
Stability	I	N	L_1	I	N	L_1	I	N	L_1
X_o(m)	430	92	43	430	92	43	430	92	43
X + X_o(m)	780	449	393	950	612	563	1530	1192	1143
n	.5	.4	.2	.5	.4	.2	.5	.4	.2
C_y	.40	.47	.60	.40	.47	.60	.40	.47	.60
P	1.48	1.41	.67	1.76	1.62	.67	2.08	2.03	.81
W	.87	.85	.50	.92	.90	.50	.99	.95	.59

Inclusion of the wide plume factor in the meandering plume equation eliminates the anomaly of yielding a sector-average concentration that is greater than the steady concentration for all stabilities and wake corrections.

In applying the diffusion models to the observed data, it is necessary to translate ΔT into stability, and to specify whether the plume is steady or meandering during a given hour.
2.6.L-14

The Con Edison system for defining stability in terms of temperature gradients was defined by B. Davidson in N. Y. U. Tech. Rep. 372.3, which is included as Pgs. Q26-Q43 of Sec. 2.6 of the Unit 2 FSAR[1]. Davidson divided the temperature gradient spectrum into three parts using the isothermal (0°F/1000 ft) and adiabatic (-5.5°F/1000 ft) temperature gradients as dividers. This yielded three stability categories lapse (L), neutral (N) and inversion (I). The L category was further subdivided into a light wind lapse (L_1) for wind speeds of 1-3 m/s and a strong wind lapse (L_2) for wind speeds > 4 m/s.

For the data from the IP3 tower, reported in N.Y.U. Rep. TR71-3[3], the ΔT corresponding to an adiabatic lapse rate over a height interval of 95-7=88 ft was –0.48°F. Accordingly, observations were assigned stability categories as follows:

inversion I	$0 < \Delta T$
neutral N	$-0.5°F < \Delta T < 0$
lapse L_1	$\Delta T < -0.5°F$

The L_1 lapse was assumed since most of the wind speeds of interest were less than 4 m/s.

Meander was estimated from the progression of wind directions in Fig 11 of N.Y.U. Rep. TR-71-3[3]. Each data point represents an average wind direction for an even-numbered hour. The average wind directions during the odd-numbered hours were estimated by arithmetical averaging of adjacent even-hour values. If the difference between the two odd-hour wind directions straddling a given even hour was greater than 20°, the wind was presumed to have meandered during the even hour, and the dilution factor was calculated according to Eq (4).

2.6.L-15

If the odd-hour difference was less than 20°, the wind was assumed to have been steady, and Eq (1) was used. Steady plumes occurred 75% of the time.

On several occasions, ΔT data were missing. These hours were assigned I stability if the wind speed was less than 2 m/s and the wind meandered. N stability was used for all other cases. (See Section 4 for a discussion of stability under low speed inversion conditions.)

2.24 Cumulative Probability of Observed $\Sigma X/Q$.

Cumulative probability is derived by summing the frequency distribution from the tail end, representing zero or small probability, toward the center. The small probabilities are associated with very high values of hourly χ/Q, and these in turn are associated with light winds and inversion stability.

Inasmuch as we are interested in obtaining information regarding $\Sigma X/Q$ values at very low probabilities, it is not necessary to obtain the complete frequency distribution under all wind conditions. It is sufficient to make sure that all the high $\Sigma X/Q$ hours appear in the data base.

Figs 10a-d of N.Y.U. Rep TR 71-3[3] show the wind behavior at IP 3 on all of the days in a two-month period when the geostrophic wind was virtually zero. This means that the wind-driven circulation in the valley was absent, and the resulting wind motions were due primarily to density currents originating along the valley slopes. These are the lightest winds possible in a valley system. Therefore, a cumulative frequency distribution in the low probability range, based on wind behavior during these days, should be essentially the same as a distribution which includes the days on which the wind-driven circulation is strong.

2.6.L-16

Fig 11 shows only those hours of Figs 10a-d when the wind fell into the 000°-045° sector. The joint values of speed $\bar{u}$, direction θ, and temperature difference ΔT are given in the Figure at each data point.

The data base in Fig 11 includes 15 days of light winds during the two month period of July 15-Sept 15, 1970. This period was chosen because the tests were designed to duplicate, as closely as possible, similar experiments conducted at Indian Point during Sept.-Oct. 1955 and included in Sec 2.62 of the Unit 2 FSAR[1].

An analysis which surveys only two months out of a year may be criticized as being unrepresentative. However, to quote from Pg 2.6-3 of the Unit 2 FSAR[1]: "In general, these local winds are most frequent under clear sky and relatively light prevailing wind conditions such as occur mostly in the fall of the year." Again, on Pg 2.6-6: "It is concluded that the "worst meteorological conditions are associated with the nocturnal down-valley flow which is most frequent during September and October."

The 1955 data include 12 days in which the large scale flow was virtually zero and 35 days in which the large scale flow was less than 16 mph. It is not known whether these are overlapping statements, but it seems likely that at least 50 days were included in the sample. Thus, the frequency of occurrence for the virtually zero flow was 12/50 = 25% in 1955.

The 1970 data also show a frequency of 15/60 = 25%. Therefore, it seems likely that the 1970 tests were made in a period of high frequency of light winds, and that a sample taken throughout the year would show a lower average frequency. Consequently, use of the 1970 2-month data base is a conservative procedure.

2.6.L-17

The first step in the calculation was determine the individual values of χ/Q for each of the data points in Fig 11 of N.Y.U. Rep TR71-3[3], according to the diffusion models and classification procedures described in the previous section. Tables 4, 5 and 6 show the results. A zero value of χ/Q was assigned to those hours not shown as data points, since the wind was out of the critical sector during those hours.

The next step was to accumulate these single-hour values χ/Q in 2 hr, 8 hr and 24 hr sequences, starting at each of the 24 hours in a day. For the 2 hr and 8 hr sequences, values of χ/Q for the odd hours were estimated by arithmetically averaging the adjacent even hour values. Then individual sequence accumulations of χ/Q, designated S_I^{I+n-1} where I = starting hour and n= number of hours in the sequence, were calculated, and a cumulative probability distribution made based on a sample population of 24 x 60 = 1440 possible sequence accumulations in the two month period. For the 24 hr sequences, S_I^{I+n-1} was assumed equal to twice the sum of the even-hour χ/Q values for a particular day, and 24 of these equal sums appeared each day.

An element of conservatism was introduced into the above calculation by simply adding the hourly χ/Q values in each sequence. In actuality, these hourly χ/Q values occur only on the plume centerline for steady plumes, although they occur anywhere in the sector for meandering plumes. However, the plume centerlines do not lie in the same direction during each hour of the 60 day period. Therefore, a fixed sampling point at the site boundary or low population zone would experience concentrations varying from the hourly χ/Q shown in Table 3 to zero, depending upon the direction of the steady plume axis.

2.6.L-18

Examination of the 15 useful cases in Fig 11 of N.Y.U. Rep TR 71-3[3] shows that the mean wind direction for all data points was 026 ^{o} and the standard deviation was 4.7 ^{o}. If the distribution of wind directions were normal, this would correspond to a probability of 96% that the winds would fall in a ± 10 ^{o} sector about the mean. Thus, the plume centerlines appear to have been normally distributed in a 20 ^{o} sector centered on 026 ^{o}.

The reduction factor for long time plume centerline concentrations with short time centerline fluctuations may be estimated by Eq 3.120 in "Meteorology and Atomic Energy" by D. H. Slade, ed, 1968[6], as $Y^2/(Y^2 + D^2)$, where Y^2 may be taken as the lateral variance of the steady inversion plume and D^2 the variance of the centerline fluctuations. These quantities are given for the inversion case by:

$$\overline{Y^2} \approx \sigma_y^2 = C_y^2 (X + X_o)^{2-n} /2 = 0.08 (X + 430)^{1.5}$$

and

$$\overline{D^2} = (4.7 X/57.3)^2 = 0.0067 X^2$$

the reduction factor is tabulated below for several distances:

dist X (m)	350	520	1100
reduction factor	0.67	0.56	0.37

Figs 1 and 2 show the cumulative probability distribution of for three time periods and å c / Q three distances from the source. They represent the simple addition of plume centerline values for the steady plumes, and sector average values for the meandering plumes. Application of the centerline fluctuation factor described in the preceding paragraph would lower the å c / Q values at the

2.6.L-19

same probability levels. The amount of the reduction would be somewhat less than the values shown because only 75% of the plumes were steady.

2.3 Accident Diffusion Model Conservatism Estimates

When the Con Edison and AEX-DRL accident model å c / Q values from Table 3 are marked on the appropriate curves of Figs 1 and 2, one may estimate the probability levels and factors of safety which were described in Sec 2.1. Table 7 lists these values.

For the 0-2 hr period, the Con Edison model å c / Q is exceeded about 0.7% of the time at the average site boundary distance. The AEC-DRL model å c / Q , being lower, is exceeded about 1.1% of the time. At the low population zone, x = 1100m, both model å c / Q values are exceeded at about the 0.5% probability level.

For the 0-8 hr period at the low population zone boundary, the Con Edison model å c / Q is exceeded 0.7% of the time, whereas the AEC-DRL model is never exceeded. This is due primarily to the AEC-DRL specification of a 1 m/s wind speed during the 2-8 hr period.

For the 0-24 hr period, neither the Con Edison nor the AEC-DRL model å c / Q values are ever exceeded. This is due primarily to the omission in both models of the diurnal wind direction reversal which removes the plume from the design sector for a large part of each day. According to Fig 11 of N.Y.U. Rep. TR 71-3[3], the longest occasion of wind duration in the sector during a given day was 10 hours, with an average of 7.0 hrs and a standard deviation of 2.65 hrs. Thus the observed 10 hr maximum duration may be expected to occur 75% of the time, and a maximum duration of 14 hrs may be expected to occur 99% of the time if the durations are assumed to be normally distributed.

2.6.L-20

The factors of safety in Table 7 are defined by

$$\text{factor of safety} = \frac{\text{model } \sum\chi/Q}{\text{observed } \sum\chi/Q \text{ at specified probability level}}$$

The two models show about the same factors of safety at the 1% probability level for all distances and time periods, except that the AEC-DRL model shows more conservatism in the 0-8 hr period, due to its use of a 1 m/s wind speed in the 2-6 hr interval while the Con Edison model uses a 2 m/s speed for the same interval. Similar points of similarity and disagreement between the models occur at the 5% probability level.

At the 5% probability level, the two models show about the same agreement and disagreement with respect to factors of safety as appears at the 1% probability level. However, the factors of safety are higher by about 6-10 times for the 0-2 hr period. This reduces to about 2.5 times for the 0-8 hr period and 1.5 times for the 0-24 hr period.

It should be remembered that these factors of safety apply to χ/Q values, or to concentrations if the source strength remains constant throughout the sampling period and no decay occurs in transit from source to sample point. In theory, the source strength will be a rapidly decreasing function of time, due to operations within containment, initiated at the time of occurrence of the accident. Therefore, if the curves of Fig 2 and the corresponding data in Table 7 are to be used for predicting probabilities and factors of safety for cumulative concentration $\sum\chi$, appropriate adjustment must be made for the behavior of Q with time.

2.6.L-21

3. Building Wake Effects in Diffusion

3.1 Physical Appearance of Diffusion Model Wake Plumes

It is often useful to visualize plume behavior by determining the plume boundary in space. This is particularly helpful in wake diffusion estimates. The Con Edison and AEC-DRL accident meteorology models both incorporate wake diffusion parameters whose effect on the plume shape is not immediately seen. In this section, the wake plumes as described by the diffusion models, and as observed in wind tunnel and field tests, will be compared.

The Con Edison wake diffusion model employed a virtual displacement of the point source to a location upwind of the building. This has the effect of increasing the distance available for the plume to grow laterally and vertically before it reaches a given station downwind of the building. The increased transverse dimensions allow the matter in the plume to be distributed over a wider cross-sectional area, and the average concentration is thereby reduced at that station.

The calculation of the vertical point displacement distance for the Con Edison diffusion model is described on Pgs 14.3.5-8 and 14.3.5-9 of the Unit 2 FSAR[1]. Analytically its value is given by

$$X_o = (A/8\ C_y C_z)^{1/(2-n)} \qquad (5)$$

where A is the building area projected in the direction of the wind and C_y, C_z and n are the Sutton diffusion parameters.

The shape of the plume boundary between the virtual source and the building is irrelevant since the plume is non-existent in this region. The shape of the boundary at large distances downwind of the building is that of a simple plume from a ground level continuous point source, i.e. elliptical in cross-section with the horizontal axis at the ground surface. At the building location

2.6.L-22

the real plume boundary must be that of the building wake. Between the building and some distance downwind, the plume boundary changes shape gradually from that of the building wake to that of the undisturbed plume.

The virtual source displacement method does not define the plume boundary at the building, and its characterization of the boundary shape takes on increasing reality with distance downwind from the building. However, it is interesting to compare the hypothetical elliptical plume dimensions with the building dimensions at the building location.

Although the Con Edison diffusion model utilizes the Sutton equations, the concentration distribution in any cross-section is bi-gaussian, and the Sutton parameters are related to the gaussian plume parameters by

$$\sigma_y = C_y X^{(2-n)/2} / \sqrt{2} \qquad (6)$$

and

$$\sigma_z = C_z X^{(2-n)/2} / \sqrt{2}$$

Eq (5) derives from Eq (6) when the assumption

$$4\sigma_y = 4\sigma_z = \sqrt{A} \qquad (7)$$

is made. However, the equality of σ_y and σ_z implied by (7) indicates that the plume boundary at the building is a circle, as a special case of an ellipse. Since no specification is made of the building shape, one may assume that the projected cross-section of the building is an approximate semi-circle, with radius R_B equal to the building height. This leads to $A = \pi R_B^2/2$ and

2.6.L-23

$\sigma_y = \sigma_z = \sqrt{A}/4 = 0.31R_B$. Now assuming as is commonly done, that a gaussian plume boundary lies at 2.5 , the plume boundary radius becomes $R_P = 2.5\ (0.3/R_B) \cong 0.08R_B$.

This result is not far from reality for a hemispherical building, as may be seen by examining the photographs in Fig 5.23 of Meteorology and Atomic Energy. [6] For rounded building surfaces, the wake seems to form downwind of the building centerline, especially so for high Reynolds Numbers, and the effective wake radius is smaller than the building radius.

For sharp-edged buildings, however, the wake boundary is larger than the building radius. For example, Fig 5.18 of Meteorology and Atomic Energy shows the cavity and wake downwind of a sharp-edged square plate with the cavity boundary at about 2 plate half-widths above the plate half-widths above the plate centerline. Other experiments described in the same chapter of Meteorology and Atomic Energy[6] show that the wake boundary for a sharp edged building is about 1.5 building heights above the ground. If considerable roughness exists around the base of the building so that flow is retarded in the lower layers near the ground, the wake height is reduced.

Therefore, it seems that the virtual point source displacement calculation given by Eq (1) is realistic for rounded buildings and conservative for sharp-edged buildings, with respect to replication of the plume dimensions at the building.

The AEC-DRL[(2)] wake model sets $\sigma_y \sigma_z = cA/\pi$ at the building. If A =

$\pi R_B{}^2/2$ and $\sigma_y = \sigma_z$ as before, $\sigma_y = \sigma_z = \sqrt{c/2}\, R_B{}^2$ and $R_p = 2.5\sqrt{c/2}\, R_B = 1.77\sqrt{c}\, R_B$.

Using the AEC- DRL [(2)] shape c = 0.5, we obtain $R_P = 1.25\ R_B$.

Therefore, the AEC-DRL wake plume boundary at the building appears to be less conservative but more realistic than the Con Edison wake plume boundary at the same location.

2.6.L-24

The greatest element of conservatism in both wake models is the use of the cross-sectional area of the reactor building alone as the numerical value of A. At a real site, the reactor building may have contiguous auxiliary buildings and lie in a complex of other buildings. Matter released into the atmosphere from the reactor building will diffuse into the composite wake of the reactor and auxiliary buildings, and may disperse laterally and even upwind into the wakes of other buildings in the complex. Therefore, the effective cross-sectional area which characterizes the wake at the building should be based on the probable plume boundary dimensions near the reactor building rather than the area of the building alone. Unfortunately, no systematic study of this aspect is available as a basis for formulating a more liberal rule.

3.2 <u>Wind Tunnel Test of Diffusion in the Indian Point Complex</u>.

Some information regarding the behavior of gas released in a building complex with restricted vertical diffusion potential was obtained in a wind tunnel test of a topographical model of the Indian Point reactor complex. The study was reported in

"Wind Tunnel Test of Gas Dispersion From Indian Point Unit 1" by James Halitsky, June 29, 1971,[(7)]

and was submitted to the AEC-DRL in connection with a safety analysis of the Unit 1 stack under tornado wind loadings.

In this test, the model was oriented in the 020 ^{o} wind direction, the tunnel wind stream was isothermal, and had low turbulence (<1%) and uniform mean velocity everywhere except in the floor, ceiling and wall boundary layers. Tests were conducted at prototype wind speeds of 6.7 and 11.1 m/s which, with

2.6.L-25

Froude Number scaling and model linear scale of 1:360, mandated tunnel velocities of 0.35 m/s and 0.58 m/s, respectively. SO_2 tracer gas was released at ground level just downwind of the Unit 1 reactor shell, and sampled on the tunnel centerline at prototype downwind distances of 305m, 610m, and 1100m from the Unit 1 Stack, and at prototype elevations of 2m, 16m, 30 m, 45m, 74m, 103m, 131m, and 160m above local ground at each downwind location.

From the vertical concentration traverses (fig 18 of the reference) it was found that the gas had diffused upward through the entire surface boundary layer by the time it reached the first longitudinal station (305m), and did not diffuse farther vertically with additional distance downwind. The boundary layer height (and plume depth) was about 90m above local ground. Suppression of further upward diffusion was caused by the existence of very low turbulence (equivalent to Pasquill Type E stability) in the free stream.

The lateral plume boundary distance may be calculated from knowledge of the measured value of $\chi \bar{u}$/Q at the ground and the value of σ_Z, which may be assumed to have been 90/2.5 = 36m since the vertical concentration profiles were approximately gaussian in shape. The observed data for the test, and the corresponding σ_y values, calculated with

$$\chi u/Q = [\pi \sigma_y \sigma_Z]^{-1} \quad (8)$$

are shown below:

distance (m)	305	610	1100
$\chi \bar{u}$/Q (m^{-2}) (observed)	2.2×10^{-4}	1.0×10^{-4}	0.8×10^{-4}
σ_z (= 90/2.5)m (observed)	36	36	36
σ_y (m) calculated	39	38	111

2.6.L-26

The above observed values may be replicated by calculation with fair accuracy by using Eq (8) with a fixed σ_Z, and assuming a Pasquill stability class slightly more unstable than C, for σ_y. The following table shows the calculated values:

distance (m)	305	610	1100
$X\bar{u}/Q(m^{-2})$(calculated)	2.3×10^{-4}	1.2×10^{-4}	0.7×10^{-4}
σ_z (m) (assumed)	36	36	36
σ_y (m) (assumed C^+)	39	73	130

The plume behavior in this test is an example of the mixed type of diffusion process which takes place in the atmosphere when vertical diffusion is suppressed by inversion temperature gradients while horizontal diffusion is uninhibited and, in fact, may be augmented by slope density currents and mechanical turbulence generated by wind passage between buildings and other structures in a reactor complex.

Attempts were made to fit the Con Edison and AEC-DRL wake models to the observed tunnel plume behavior, but were unsuccessful. It appears that neither the vertical source displacement nor the volume wake correction, which work well for discrete buildings, is applicable to diffusion in building complexes.

2.6.L-27

4. Turbulence Characteristics Under Low Wind Speed Inversion Conditions.

4.1 Observations at the IP 2 Tower in 1969

From September 1968 to May 1969, wind data were measured at three locations in the Hudson River Valley at and near Indian Point, to aid in selection of a site for proposed new nuclear power units. A report of this investigation is contained in NYU Report TR 70-3[8].

The locations of the three towers, designated by the symbols IP2, MP and BP respectively, are shown in Figs 1-5 or TR 70-3. Of particular significance to the present study is an analysis, contained in TR 70-3, of the wind behavior at the IP 2 tower during periods of upper air stagnation.

The IP2 tower was located 305m southwest of Unit 3 on a cleared strip of ground which sloped down in a westerly direction to the Hudson River. The base of the 100 ft tower was at el 60 ft above river elevation. The tower was equipped with sensitive wind speed and direction sensors at 100 ft above local ground, and temperature difference sensors at 95 ft and 5 ft above local ground.

Figs 11-13 of TR 70-3 show simultaneous measurements of hourly mean wind speed, wind direction, temperature difference and wind direction range/4.3 for three periods of upper air stagnation during April and May 1969. During seven nights of these periods, the wind behavior approached the type assumed in both the Con Edison and AEC-DRL 2-hr accident meteorology models, exhibiting low wind speed and temperature inversion, but did not exhibit directional steadiness either within an hour or from hour to hour.

Directional steadiness within an hour is usually identified by a

2.6.L-28

small standard deviation of 15-sec average directional angles about the hourly mean direction. It has been observed by Slade in "Meteorology and Atomic Energy, 1968", Sec. 3-3.4.1, [6] that the Pasquill stability categories may be associated with typical numerical values of the standard deviation as follows:

Pasquill Stability Class	Std. Deviation σ_θ
A	25°
B	20°
C	15°
D	10°
E	5°
F	2.5°

Applying Slade's classification system to observations of direction range/ 4.3 . σ_θ) during the seven night periods under consideration, it was found that the observed hourly stabilities varied from A to F, with an average of about C or D, even though a temperature inversion was always present. Thus, on the average, a considerable amount of unsteadiness was present during an average hour.

The effect of wind unsteadiness (high σ_θ) on dispersion is to reduce concentrations to values below those which would occur under steady conditions (low σ_θ). For example, using Slade's Fig A.2, it is readily seen that a change from F stability to D stability would increase σ_y by a factor of 2 at a distance of 1100 m and thereby decrease, $\chi\bar{u}/Q = \left[\pi\sigma_y\sigma_z\right]^{-1}$ a factor of 2, even

if σ_z should remain unchanged. By comparing σ_z values in F and D stabilities, it is seen that an additional reduction in concentration by a factor of 2 would occur if the vertical unsteadiness changed in proportion to the lateral unsteadiness.

Thus, it is important in evaluating conservatism to establish realistic stability classifications in both the vertical and horizontal directions.

4.2 Observations at the IP 3 Tower in 1970

4.3

When the Indian Point meteorology tower was moved from the IP 2 location to the IP 3 location in late 1969, a bivane was added at IP 3 to provide additional data on low speed inversion stability characteristics. Data were reported in NYU Report TR 71-10[9].

The bivane was equipped with a switching device which allowed the strip chart recorder to run at 3 in/min for 10 minutes each hour, and at 3 in/hr during the remainder of the hour. The recorder was operated at various intervals from May to December 1970. The intervals, which were 30 hours long, were determined according to the expectation of low wind speeds and by the availability of personnel. A total of 153 hours for which concurrent speed, direction and vertical temperature difference were available, and the speed was less than 2 m/s, formed the data base.

The most important conclusions of the study were:

a) Under inversion conditions, the standard deviation of azimuth angles (σ_θ) increased with wind speed to 1.4 m/s and then decreased, with the following values of σ_θ and equivalent Pasquill stability class according to Slade's system: (from Fig 1 of TR 71-10):

2.6.L-30

No. of Observ.	Speed Range (m/s)	5-Sec Smoothing σ_θ(deg)	5-Sec Smoothing Stability	200-Sec Smoothing σ_θ(deg)	200-Sec Smoothing Stability
3	0-0.4	9	D	7	DE
5	0.4-0.8	12	CD	10	D
12	0.8-1.2	16	C	15	C
17	1.2-1.6	18	BC	13	CD
17	1.6-2.0	9	D	4	E

The 5-sec smoothing period better approximates Slade's procedure than the 200 sec smoothing period.

b) Under the same inversion and speed categories as above, the standard deviation of elevation angles, σ_θ, remained about constant at 6-7° over the entire range of speeds.

c) A breakdown of the above according to wind direction sectors showed a minimum σ_θ of 8° (DE stability) to occur in the 340° - 040° wind sector, with a corresponding σ_ϕ of 6°, for 5 sec smoothing.

d) Under inversion conditions, both σ_θ and s_f increased with $\Delta T = T_{95}-T_5$ (°F) in the 340° - 040° wind sector as follows:

No. of Observ.	ΔT Range (°F)	σ_θ (deg)	5-Sec Smoothing Stability Class	σ_ϕ (deg)
10	0-1	8	DE	6
7	1-2	9	D	7

6	2-3	20	B	9

e) A calculation of $\chi \bar{u}/Q$ by the method of Hay and Pasquill in Advances in Geophysics (6) Academic Press pp 345-365[10], using data for the 340° - 040° wind sector, shows the following at a distance of 1100 m and $\bar{u}$ = 1.85 m/s:

ΔT(°F)	Values of $10^4 \chi \bar{u}/Q$ Steady	Meander
0	0.7	0.4
0.5	0.8	0.4
1.0	0.7	0.4
1.5	0.4	0.3
2.0	0.2	0.2

A concentration maximum occurs at a ΔT of 0.5 °F.

4.4 Interpretation of Bivane Observations

The behavior of the bivane can be attributed to the development of katabatic winds, or down-slope currents, under inversion conditions. These currents are caused by cooling of slope air near the ground. This air becomes more dense than air over the floor of the valley, resulting in local pressure differences which cause drainage of air down the slopes.

Over irregular terrain, non-parallel descending currents impinge on one another, creating irregular motions in both the horizontal and vertical directions. However, the horizontal motions are unconstrained, while the vertical motions are suppressed by the inversion. Therefore the effects of the slope currents show up strongly in enhanced horizontal turbulence (σ_θ) but have only a small effect on vertical turbulence (σ_ϕ).

The bivane observations show that, in the 340^o - 040^o sector, maximum turbulence levels of $\sigma_\phi \approx 18^o$ and $\sigma_\phi \approx 9^o$ occur with wind speeds near 1.4 m/s and vertical temperature differences of about 2.5 ^{o} F. Turbulence levels fall off at wind speeds above and below 1.4 m/s. The temperature difference accompanying speeds greater than 1.4 m/s must have been equal to or less than 2.5 ^{o} F since Table IV of TR 71-10 shows no observations >3.0 ^{o} F in this sector. This may indicate the existence of extra-valley influences in creating higher wind speeds that are not thermally-generated.

The high turbulence level at a wind speed of 1.4 m/s and $\Delta T = 2.5\,^o$ produces a $\chi \bar{u}/Q$ equivalent to a Pasquill type A-B stability (See Table XI).

As ΔT reduces, $\bar{u}$, σ_θ and σ_ϕ also reduce, causing an increase in $\chi \bar{u}/Q$ to a maximum at $\Delta T = 0.5\,^o$ where the equivalent Pasquill stability is C-D.

Thus, the bivane data confirm the applicability of the Pasquill C-D stability class at near-neutral temperature stratification, even in irregular terrain. However, the customary increase of stability class with increasing temperature gradient, which is valid for level ground, is not valid at this site. On the contrary, the effective stability decreases with increasing temperature gradient at least up to a $\Delta T = 3\,^o$ F. Similar behavior was observed at the IP 2 tower during the previous year.

It appears that the assignment of I stability to those hours which exhibited inversion temperature stratification in the calculation of hourly average χ/Q from observed data in Sec 2.23 in extremely conservative, and that a more realistic rule would be to use N stability instead.

2.6.L-33

5. <u>Wind Persistence</u>

Wind persistence is a measure of steadiness of wind direction with increasing time. It is important in accident meteorology because invariance of wind direction exposes a stationary subject on the plume centerline to the highest concentration for the longest period, under given conditions of source strength and wind speed.

Both the Con Edison and AEC-DRL accident meteorology models postulate that the hypothetical accident occurs at the onset of a 24-hr period of strong wind persistence. Both models assume a steady hourly mean wind direction for the first 8 hours. For the next 16 hours, the Con Edison model retains steadiness, while the AEC-DRL model permits uniform direction meander within a 22½ ^{o} wind sector. Although the Con Edison model is more stringent with respect to direction in the latter period, it also assumes a higher wind speed, thereby making the hourly average χ/Q values about the same for both models.

It is generally recognized that a correlation exists between strong wind speeds and long periods of persistence. [15] However, the accident models assume light wind speeds (<2 m/s) and inversion conditions during the 0-24 hr period. The probability of occurrence of long periods of persistence under these conditions is examined in the following sections.

5.1 <u>Persistence Data Taken in 1955</u>

On Pg 2.6-4 of the Unit 2 FSAR, experiments are cited to show that " … a consecutive 24 hour down-valley flow with light wind speeds and inversion conditions is extremely improbable …", that the " ... duration of the down-valley flow is about 12 hours rather than 24 hours, …" and that "… it must be concluded that the safety analysis for the first 24 hours is conservative to within a factor of about two …", since the average speeds assumed in the model are about the same as those observed at the site.

2.6.L-34

The experiments referred to above were measurements taken in Sept.-Oct. 1955 with an Aerovane mounted 70 ft above the Hudson River on the mast of a ship, the Jones, whose location is shown in Figs 1-3 of NYU Rep TR 71-3[3] at the point marked J. The data taken during the experiments were presented in Figs 2.6-1, 2.6-2 and 2.6-3 of the Unit 2 FSAR.

Figs 2.6-1 and 2.6-2 are wind hodographs, or polar coordinate representations of the behavior of the wind velocity vector during a 24 hour diurnal cycle. Fig 2.6-3 is a graph of wind steadiness, defined as the ratio (mean vector wind speed)/(mean scalar wind speed) vs time of day.

All three Figures present the averages of measurements taken on 12 days and 35 days during the test period. The procedure of basing the accident meteorology model on the average condition rather than on some less frequent but more severe condition has been criticized as not being sufficiently conservative (see Appendix C to the AEC-DRL Safety Evaluation for the Unit 2 Operational License hearing, and Appendix C to the AEC-DRL Safety Evaluation for the Unit 3 Construction License Hearing).[11]

Also somewhat misleading is the placement of the coordinate center for the hodographs in Figs 2.6-1 and 2.6-2 at the Unit 2 reactor building, rather than at the Jones, which was located about a mile to the north.

The questions regarding individual, rather than average, wind characteristics, and possible differences between characteristics at the Jones and at the plant site, could not be resolved because the original data were no longer available.

2.6.L-35

5.2 Persistence Data Taken in 1970

5.21 Hodographs

In 1970, experiments were undertaken to duplicate as closely as possible the 1955 experiments at the Jones and to obtain concurrent data for the ship and plant locations. Results were presented in NYU Report TR 71-3.[3]

An Aerovane was mounted on the Cape Charles, which was anchored about 350 m SW of the former location of the Jones (at point CC in Figs 2-4 of Report TR 71-3). A sensitive Climet cup and vane anenometer was mounted on the IP 3 tower.

It was found that the average wind hodograph at the ships, separated by 14 years in time, were very similar, exhibiting the same diurnal reversal pattern and the same 2.5 m/sec nighttime downvalley speed in both years. It seems clear that the diurnal reversal of direction is a characteristic of the site and can be expected to occur every year.

The average wind hodograph at IP 3 showed a similar pattern of diurnal reversal, but the average nighttime downvalley speed was somewhat less than 2 m/s.

All sixteen daily wind hodographs at IP 3 showed the diurnal reversal, and exhibited considerable variability in speed and direction from day to day through a complete cycle. The longest period of direction persistence was 8 hrs, occurring on each of 5 nights, and the average wind speed during each period was least 2 m/s.

Further analysis of wind behavior during these and other low speed inversion days is presented in Sections 2.23 and 4.2 of this report.

2.6.L-36

5.22 numeration of Occurrences of Persistence

Also presented in Report TR 71-3 is a listing of the number of occurrences that wind of specified characteristics persisted for a specified number of consecutive hours during the entire 10-month test period in 1969-70. The data of Table 1 of Report TR 71-3, showing persistence under inversion conditions for the 002^{o} - 022^{o} and 022^{o} - 042^{o} sectors* taken separately, are shown combined in Table 8 of this report. Also shown are comparable data taken during 12 months of calendar year 1971 (previously unpublished).

The two sets of data show that:

a) persistence increases with wind speed,

b) the probability of finding 2 consecutive hours when the wind speed is 1 m/s or less in the 40^{o} wide sector under inversion conditions is extremely small (2 hours in 5989 or 0.03% during 1969-70 and 0 in 1971),

c) the probability of finding 2 consecutive hours when the wind speed is 2 m/s or less under the same conditions as above was 7 hours in 5989 or 0.11% during 1969-70, and 60 hours in 5560 or 1.1% in 1971,

d) no persistences longer than 7 consecutive hours occurred in either year, for wind speeds less than 3 m/s.

The above figures are based on occurrences of wind direction somewhere in a 40 ^{o} sector. It is likely that a more stringent definition of steadiness would reduce the probabilities even further.

*Wind direction sectors are identified in this report by the wind angles at the sector boundaries; the difference between these boundary angles is the sector width of 20 ^{o}. In the Unit 2 FSAR and early NYU reports, sectors are identified according to the range of wind observations as read to the nearest 5 ^{o} from wind charts. Thus, wind sector 002^{o} - 022^{o} in this report corresponds to wind sector 005^{o}-020^{o} in FSAR.

2.6.L-37

6. Recurrence of Annual Wind Statistics

Fig 3 shows the annual distribution of wind direction for 1956, 1957 (from NYU Report 372.4, Pg R-6 of Sec 2.6 of the Unit 2 FSAR) and for 1970 and 1971 (unpublished).

The bi-modal character of the distribution occurs in each of the years, and the fraction of time that the wind is in the critical 002 ^{o} - 022^{o} sector remains relatively constant in all four years.

The most noticeable change is a re-distribution of some southerly (160^{o} - 200^{O}) and north-westerly (290^{O} - 360^{O}) winds to the easterly and westerly sectors. Neither of these changes is a matter for concern since easterly winds carry releases over the river and westerly winds carry them toward a more distant site boundary.

Fig 4 shows the wind direction distribution in each temperature gradient class for 1956, 1970 and 1971. The 1956 curves represent data in Table 3.3 of NYU Rep. 372.3, (Pgs Q-39 and 40 of Sec 2.6 of the Unit 2 FSAR). The 1970 and 1971 data have not been published as yet. The 1956 wind statistics appear to be representative of a typical year. Therefore the stability mix used in the 1-30 day period of the accident model, which was selected on the basis of the 1956 statistics, is still valid.

Fig 5 shows a small decrease in the frequency of light wind speeds (02- m/s) in the critical sector. The shift of southerly winds toward the southeast seems to be accompanied by an increase in wind speed.

2.6.L-38

7. Plume Behavior Beyond the Site Boundary

7.1 Steady Wind

Material released as a leak at the surface of a building under non-zero wind speed conditions will diffuse initially around the building surfaces in a complicated pattern determined by the building shape and the location of the leak. Typical dispersion pattern around a wind tunnel model of a reactor building having rounded surfaces are shown in Figs 5.29a-c of Meteorology and Atomic Energy[(6)].

It is seen that concentrations decrease rapidly with distance from the release point, although the direction of the minimum rate of decrease is not always readily foreseen. This results in irregular, non-gaussian concentration distributions in planes normal to the mean wind direction at stations within several building diameters downwind from the building.

However, at about 5 building diameters downwind, the concentration profiles become more-or-less regular in that a concentration maximum occurs at the ground on the building centerline and concentrations decrease radially outward in the y-z plane to zero at the wake boundary. With increasing distance downwind, atmospheric turbulence acts to further smooth the profiles in an asymptotic approach to the familiar bi-gaussian distribution.

A non-dimensional distance of 5 diameters is equivalent to 5 x 43 = 215 meters for the Unit No. 2 reactor building. Since the site boundary distance is 350 m from Unit No 2, it seems reasonable to assume that the maximum concentration in a plane normal to the wind at the site boundary will indeed be at the ground. If this is so, then ground level concentrations beyond the

2.6.L-39

site boundary will always be lower than at the boundary by virtue of continued plume expansion with distance downwind.

7.2 Wind Reversal

If the wind should reverse 180^{o} after a steady plume has been established, the time history of concentration at the former downwind site boundary after reversal will be a continued reduction of concentration with time because successively more diffuse portions of the plume will cross the boundary as the plume "backs up."

If the wind reversal persists long enough, the backed-up plume will become the ambient environment within which a new plume has formed in the opposite direction from the original. The total plume concentrations will then be the sum of the residual old plume concentrations plus the new plume concentrations. The critical site boundary will be located 180^{o} from the previous one.

At the Indian Point site, 180^{o} wind reversals occur occasionally during periods of upper air stagnation, between about 1000 and 1200 hours and between 2000 and 2200 hours (see individual hodographs in Figs 10a-d of NYU Report TR 71-3[3]).

In the daytime reversal, the wind has been in the downvalley direction with inversion conditions, but the change to the upvalley direction is accompanied by a change to a lapse temperature gradient. Therefore both the newly-established plume and the residual old plume have light concentrations compared to those assumed in the accident model.

In the nighttime reversal, the accident meteorology must be assumed to occur at some time prior to the reversal in order to create the residual old plume. This old plume is generated under lapse conditions during the period of greatest source strength. After reversal, these concentrations pass over the downvalley site boundary, but the source strength is now weaker. Therefore the

2.6.L-40

sum of the new and residual concentrations is probably not greater than obtained with the accident model.

2.6.L-41

List of References

1. Con Edison Final Facility Description and Safety Analysis Report (FSAR), as Amended, for Indian Point Nuclear Generating Unit No 2 (U.S.A.E.C. Docket 50-247).

2. U.S.A.E.C.-D.R.S. (1970): Safety Guides for Water Cooled Nuclear Power Plants.

3. Halitsky, J., E. J. Kaplin and J. Laznow (1971): Wind Observations at Indian Point, 26 November 1969-1 October 1970. N.Y.U. Dept of Met. and Ocean. GLS Report TR 71-3.

4. U.S.A.E.C.-D.R.L. (1970): Safety Evaluation of Con Edison Indian Point Nuclear Generating Unit No. 2 (Docket 50-247), with Appendix C prepared by Air Resources Environmental Laboratory, Environmental Science Services Administration, Nov. 29, 1968 and Feb. 17, 1970.

5. Turner, D. B. (1969): Workbook of Atmospheric Dispersion Estimates, Air Resources Field Research Office, ESSA.

6. Slade, D. H., ed. (1968): Meteorology and Atomic Energy. U.S.A.E.C. Div. Of Tech. Inform. CFST1 TID-24190.

7. Halitsky, J. (1971): Wind Tunnel Test of Gas Dispersion from Indian Point Unit 1. Unpublished report prepared for Con Ed dated June 29, 1971.

8. Halitsky, J., J. Laznow and D. M. Leahey (1970): Wind Observations at Indian Point, Montrose and Bowline Point, 31 Aug 1968-30 June 1969. NYU Dept. of Met. and Ocean. GSL Report TR 70-3.

9. Leahey, D. M. and J. Halitsky (1971): Low Wind Speed Turbulence and Related Diffusion Estimates for Indian Point, N. Y. N.Y.U. Dept. of Met. and Ocean. GSL Report TR 71-10. Also Atmospheric Environment Vol. 7 page 49-61 January, 1973.

10. Hay, J.S. and F. Pasquill (1959): Diffusion from a Continuous Source in Relation to the Spectrum and Scale of Turbulence. Advances in Geophysics 6. Academic Press pp. 345-365.

11. U.S.A.E.C.-D.R.L. (1969): Safety Evaluation of the Con Edison Indian Point Nuclear Generating Unit No. 3 (Docket No. 50-286), with Appendix C prepared by Air Resources Environmental Laboratory, ESSA January 2, 1968.

12. DiNunno, J. J., R. E. Baker, F. D. Anderson and R. L. Waterfield (1962): Calculation of Distance Factors for Power and Test Reactor Sites, U.S.A.E.C. DTI document TID-14844.

13. Con Edison PSAR for Indian Point Nuclear Generating Unit No. 1 (U.S.A.E.C. Docket 50-3).

14. U.S.A.E.C.- Atomic Safety and Licensing Board (1969): Transcript of Testimony in the Matter of Con Edison Application for a Construction License for Indian Point Nuclear Generating Unit No. 3.

15. Van der Hoven, I. (1969): Wind Persistence Probability ESSA Tech. Mem. ERLTM-ARL 10.

2.6.L-43

Table 1. The Con Edison Meteorological Model

Period	Wind Speed ©	Wind Direction U Variability in 20 ^{o} Sector	Frequency in 20 ^{o} Sector	Stability* Class	Stability* Frequency	Building** Wake Effect
0-2 hrs	1 m/s	steady	100%	Inversion, I	100%	yes
2-24 hrs	2 m/s	steady	100%	Inversion, I	100%	yes
1-30 days	1.74 m/s			Lapse, L_1	13.7%	
	5.23 m/s			Lapse, L_2	6.1%	
	2.79 m/s			Neutral, N	37.8%	
	2.03 m/s			Inversion, I	42.4%	
	all	meander	35%	all	100%	no

*Sutton parameters C_y, C_z and n for stability classes L_1, L_2 and N were derived from site meteorological experiments. For stability class I, the model adopted the Inversion parameters from USAEC TID-14844 “Calculation of Distance Factors for Power and Test Reactor Sites” by J. J. di Nunno, F. D. Anderson, R. E. Baker and R. L. Waterford, dated March 23, 1962.[12]

**Employs virtual point source displacement $X_o = (A/8\ C_y\ C_z)^{1/(2-n)}$ where A = building area of 2,000 m^2 for periods 0-2 hrs and 1-30 days.

2.6.L-44

Table 2. The AEC-DRL Meteorological Model

for Pressurized Water Reactors

Period	Wind Speed ©	Wind Direction, U: Variability in 22.5 ^{o} Sector	Wind Direction, U: Frequency in 22.5 ^{o} Sector	Stability*: Class	Stability*: Frequency	Building Wake Effect**
0-8 hrs	1 m/s	steady	100%	Pasquill F	100%	yes
8-24 hrs	1 m/s	meander	100%	Pasquill F	100%	no
1-4 days	3 m/s			Pasquill D	40%	
	2 m/s			Pasquill F	60%	
	all	meander	100%	all	100%	no
4-30 days	3 m/s			Pasquill C	33.3%	
	3 m/s			Pasquill D	33.3%	
	2 m/s			Pasquill F	33.3%	
	all	meander	33.3%	all	100%	no

*Volumetric building wake correction as defined in Section 3-3.5.2 of AEC TID-24190 Meteorology and Atomic Energy 1968[(6)] with shape factor c = 0.5 and minimum across sectional area of reactor building only.

Table 3. Values of $10^4\ \sum\chi/Q$ (sec/m^3)

According to the Con Edison and AEC-DRL Models
for Various Time Periods and Distances

	Con Edison			AEC-DRL		
Distance from Source (m)	350	520	1100	350	520	1100
Time Period						
0-2 hr	20.8	15.8	7.6	25.6	13.2	7.2
2-8 hr	--	--	11.4	--	--	21.6
8-24 hrs	--	--	30.4	--	--	19.6
0-8 hrs	--	--	19.0	--	--	28.8
0-24 hrs	--	--	49.4	--	--	48.4

Values of $\sum\chi/Q$ for sequences longer than 2 hours at x = 350 and 520 are not given because standards at the site boundary are specified for the 0-2 hr period only.

2.6.L-46

Table 4. Values of Hourly Average $10^4\ \chi/Q$ at 350 m from the Source

During 15 Selected Light Wind Days, July 15 – Sept 15, 1970

	Hour					
Date	22	24	02	04	06	08
7/23-24		3.5	4.7	0.6		
7/24-25		21.7	13.5	1.5	0.3	
7/25-26	0.7	1.2	0.3	1.0	0.4	
7/26-27				1.0		
7/27-28		9.8	8.3		0.5	
7/28-29			0.8	1.0	0.3	
8/8-9	5.1	1.1	1.0	0.7	0.6	
8/13-14	5.9	7.2	2.9	0.4	0.9	
8/14-15		2.4	21.7	5.2	0.6	
8/15-16		12.8	4.7		1.0	6.4
8/25-26	5.5	0.9	.8			
8/26-27				15.4	10.8	
8/27-28		5.5	0.9	0.6		
9/12-13	3.7	7.2	6.8	7.7	0.5	
9/13-14	15.4	9.9	0.7	0.4	0.8	

Table 5. Values of Hourly Average 10^4 χ/Q at 520 m from the Source

During 15 Selected Light Wind Days, July 15 – Sept 15, 1970

	Hour					
Date	22	24	02	04	06	08
7/23-24		2.2	3.4	0.4		
7/24-25		15.7	9.8	0.9	0.1	
7/25-26	0.3	0.7	0.1	0.6	0.2	
7/26-27				0.6		
7/27-28		7.1	6.1		0.2	
7/28-29			0.4	0.6	0.2	
8/8-9	3.2	0.6	0.6	0.4	0.3	
8/13-14	3.6	5.2	2.1	2.5	0.5	
8/14-15		1.5	15.2	3.7	0.3	
8/15-16		7.9	3.4		0.6	4.6
8/25-26	3.4	0.5	0.5			
8/26-27				9.5	7.9	
8/27-28		3.4	0.5	0.4		
9/12-13	2.3	5.2	4.9	0.5	0.3	
9/13-14	9.5	7.1	0.4	0.3	0.5	

Table 6 Values of Hourly Average 10^4 χ/Q at 1100 m from the Source

During 15 Selected Light Wind Days, July 15 – Sept 15, 1970

	Hour					
Date	22	24	02	04	06	08
7/23-24		0.8	1.7	0.1		
7/24-25		7.8	4.8	0.3	0	
7/25-26	0.1	0.3	0	0.2	0.1	
7/26-27				0.2		
7/27-28		3.5	3.0		0.1	
7/28-29			0.1	0.2	0	
8/8-9	1.1	0.2	0.2	0.2	0.1	
8/13-14	1.3	2.6	1.1	0.1	0.2	
8/14-15		0.5	7.8	1.9	0.1	
8/15-16		2.8	1.7		0.2	2.3
8/25-26	1.2	0.2	0.2			
8/26-27				3.4	3.9	
8/27-28		1.2	0.2	0.1		
9/12-13	0.8	2.6	2.4	0.2	0.1	
9/13-14	3.4	3.5	0.1	0.1	0.2	

2.6.L-49

Table 7. Probabilities and Safety Factors in the
Con Edison and AEC-DRL Accident Meteorology Models

	Con Edison Model			AEC-DRL Model		
	350m	520m	1100 m	350m	520m	1100m
% Probability of Exceeding Accident Model $\sum\chi/Q$						
0-2 hr	0.8	068	0.4	1.3	1.0	0.5
0-8 hr	-	-	0.7	-	-	0
0-24 hr	-	-	0	-	-	0
Factor of Safety at 1% Probability Level						
0-2 hr	1.1	1.2	1.3	1.3	1.0	1.2
0-8 hr	-	-	1.2	-	-	1.8
0-24 hr	-	-	2.0	-	-	1.9
Factor of Safety at 5% Probability Level						
0-2 hr	7.0	7.9	12.6	8.5	6.6	12.0
0-8 hr	-	-	3.2	-	-	4.8
0-24 hr	-	-	3.1	-	-	3.0

Note

350m = distance from Unit 3 to site boundary

520m = distance from Unit 2 to site boundary

1100m = distance from Units 2 and 3 to low population zone.

Table 8. Wind Persistence at IP 3 under Inversion Conditions

In Combined Sectors 002^{o} - 022^{o} - 042^{o}**
(The body of the table shows number of hourly occurrences of the designated duration and speed class)

Number of Consec. Hours	Maximum Speed in Sequence (m/s)* 0.3	0.5	1.0	1.5	2.0	3.0
During 10 months (26 Nov 1969 – 1 Oct 1970)						
1	1	2	38	83	139	270
2			2	5	7	23
3				3	1	4
4				1	0	2
7						1
During 12 months (1 Jan 1971 – 31 Dec 1971)						
1	1	6	66	115	217	431
2				18	60	181
3				2	23	89
4					4	39
5					2	19
6					1	5
7						1

*mph notation for speed in Table 1 of Rep TR 71-3[(3)] should be m/s.

**see note on page 32 regarding sector notations.

Appendix A

Chronological Review of Events Relating to the Accident Meteorology Models

The Con Edison meteorological model was developed by Dr. Ben Davidson of New York University, using meteorological data collected at Indian Point during the period 1955-1957. That meteorological investigation, conducted in support of the licensing application for Indian Point Unit No. 1, yielded three reports which contained not only the meteorological summaries, but various dose calculations for postulated releases at stack height and at ground level. These reports we submitted to the AEC in their entirety as Exhibits L-1, L-5 and L-6 of Docket 50-3, Indian Pont Unit No. 1[(3)]

Exh. L-1: N.Y.U. Tech. Rep. 372.1 "A Micrometeorological Survey of the Buchanan, N.Y. Area", dated Nov. 1955,

Exh. L-5: N.Y.U. Techn. Rep. 372-3 " Evaluation of Potential Radiation Hazard Resulting from Assumed Release of Radioactive Wastes to the Atmosphere from Proposed Buchanan Nuclear Power Plant", dated April 1957, and

Exh. L-6: N.Y.U. Tech. Rep. 372.4 "Summary of Climatological Data at Buchanan, N.Y., 1956-1957", dated March 1958.

The meteorological data acquired during the foregoing study were synthesized into the Con Edison meteorological model which first appeared in the PSAR for Indian Point Unit No. 2 (Docket No. 50-247). The supporting documentation for the model included:

Exhibits L-1 and L-6 (described above) in their entirety,

Meteorology Sections 2 and 3 of Exhibit L-5 (described above),

U.S.W.B. Tech. Paper No. 15: Maximum Station Precipitation for 1, 2, 3, 6, 12 and 24 Hours, Part X: New York, dated Dec. 1954, and

Additional meteorological data concerning wind behavior at an elevation of 70 ft above the Hudson River near the Indian Point site when the wind speed above the valley ridge lines fell into two classes; virtually zero and less than 16 mph. Those data had been collected during 1955, and had been used in generating the dose calculations in Exhibit L-5, but had not been presented previously in this form (low speed hodographs).

The Con Edison meteorological model postulated the sequence of wind condition shown in Table 1, beginning at the time of the postulated accident. The AEC did not comment directly as to the acceptability of the Con Edison model. It did request justification for the inversion frequency used in the 1-30 day period (question No. 16 in letter to Applicant dated Feb 28, 1966). This was provided in the First Supplement to the PSAR for I.P. 2 (Docket No. 50-247, Exhibit B-1).

The Con Edison meteorological model was used again in the PSAR for Indian Point Unit No. 3 (Docket 50-286). During evaluation of the dose calculations, the AEC-DRL made known its own meteorological model which has subsequently been published formerly in Safety Guide No. 4 of:

U.S. A.E.C.-D.R.S.: "Safety Guides for Water Cooled Nuclear Power Plants", dated Nov. 2, 1970.

The meteorological sequence in the AEC-DRL model is shown in Table 2.

When the AEC-DRL model became known to Con Edison, the latter compared the two models for comparable time periods, and it was found that the Con Edison model yielded higher χ/Q values than did the AEC-DRL model in all periods except 2-8 hrs after the postulated accident. However, in a submittal to the AEC in the Fifth Supplement to the I.P. 3 PSAR, it was argued in Item 8 that the AEC-DRL assumption of a 1 m/s wind speed during this six-hour period did not in fact occur, and that the Con Edison assumption of 2 m/s for the same period was adequately conservative. This factor of 2 on wind speed, if applied with the AEC-DRL model in the 2-8 hr period, would reduce the AEC-DRL value of χ/Q for the period to below the Con Edison value, thereby rendering the Con Edison model more conservative in all categories. The argument also called upon experimental evidence to show that the wind meandered during the 2-8 hr period, following a directional pattern of wind rotation. The omission of wind meander in both the AEC and Con Edison models during this period introduces conservatism into each model.

In preparation for the I.P. 3 Construction Permit hearings, Con Edison submitted to the AEC Licensing Board:

(1) Summary of Application (Docket 50-286) dated Feb. 20, 1969, and the AEC-DRL Technical Staff submitted:

(2) AEC-DRL Safety Evaluation dated Feb. 20, 1969,

(3) Appendix C to AEC-DRL Safety Evaluation: Comments on PSAR and Fifth Supplement for I.P.3, dated May 24, 1968 and Jan. 2, 1969 (included as Pgs. 75 and 76 of (2) above).

(4) AEC-DRL Summary Statement dated March 20, 1969.

During the pre-hearing conference for I.P. 3 on March 11, 1969, Board Member Pigford requested clarification from both AEC-DRL and Con Edison regarding (a) the statement in Document 2 (above) that Con Edison did not have long-term data on stability-speed-direction persistence, and (b) the differences between the Con Edison and AEC-DRL meteorological models (transcript pgs. 70-71).

During the I.P. 3 Construction Permit hearings, on March 27, 1969, oral testimony was given by both Con Edison and AEC-DRL representatives in response to Dr. Pigford's questions. It was established that "the absence of long-term data" referred to the fact that the original experimental data, taken in 1955, 1956 and 1957, were no longer available (transcript pg. 170), leaving only the statistical summaries in the various reports and submittals previously cited. The details of the Con Edison and AEC-DRL accident meteorologies, shown in Table 1 and 2, were also enumerated, and major points of discrepancy between the models were discussed.

Of particular interest to the Board was the rationale behind the AEC-DRL statement that the 0-8 hr meteorology used in the AEC model was conservative (pg. 660). This led to an extended discussion of the low wind speed hodographs shown in Figs. 2.6-1 and 2.6-2 of Section 2 of the Unit 3 PSAR, which provide the justification for both the Con Edison and AEC-DRL 0-8 hr meteorology models.

These hodographs show the progression of wind speed and direction during a typical day when the wind speed above ridge elevation is zero or small. The hodographs were constructed by averaging the measurements taken on 12 days (zero wind speed) and 35 days (small wind speed) during September and October, 1955.

The Con Edison positions was that the combination of average hodographs plus conservative assumptions regarding the persistence of wind speed, direction and stability provide adequate conservatism in its 0-24 hr meteorology model.

The AEC-DRL position was that an average hodograph implied the existence of some individual hodographs which might exhibit longer persistence of lower speed winds in the critical direction with strong stability, and that a more conservative model was called for in the absence of evidence to the contrary. In its Safety Evaluation, Document (2) above, the AEC-DRL stated that its standard meteorological model, given in Table 2, "conservatively characterizes the meteorology of the Indian Point site" in the absence of long-term data "on the specific joint frequency of stability-wind speed-wind direction persistence." The AEC-DRL consultant (ESSA Air Resources Environmental Laboratory) concurred in Document (3) above that the 1-8 hr meteorology in the AEC-DRL model was a " --reasonably conservative meteorological assumption…" in view of the absence of join-frequency data.

Board member Pigford then attempted to obtain a numerical estimate of the probability of occurrence of the Con Edison and AEC-DRL 0-8 hr models in the critical wind direction sector by questioning AEC-DRL staff meteorologist Spickler and Con Edison consultant Halitsky. Mr. Spickler reasoned that it would probably be less than 1%, since the combination of inversion stability and 1 m/s wind speed occurred approximately 5% of the time for all directions combined. However, in view of the lack of persistence data for individual cases, Mr. Spickler characterized his estimate of 1% as an "educated guess" (page 67).

Dr. Pigford then requested that Dr. Halitsky estimate the probability of occurrence of the 0-8 hr meteorological sequences as defined by AEC-DRL and Con Edison (pgs. 675-676). Dr. Halitsky requested some time to consider his reply. Further questioning was directed toward the validity of diffusion coefficients (pg 677), the need for diffusion testing to validate the Sutton diffusion model (pg 678ff and 745ff), plans for continued meteorological testing to generate the data needed to clarify the 0-8 hr assumptions (pg 682ff) and topographical effects on diffusion (pgs 749ff).

Prior to adjournment the Board posed several questions of a meteorological nature to both Con Edison and the AEC-DRL staff (pg 1671). Those were responded to at a resumption of the hearings on 13 May 1969. Mr. Spickler placed into the record the AEC-DRL standard meteorological model as shown in Table 2 (pg 1756). Two questions were directed to Dr. Halitsky (pgs. 1671 and 1672):

a) Present a technical justification for the conclusion that the frequency spectrum of wind speeds and the range of air and low wind speeds is now and will continue to be the same as that measured in 1956.

b) Present a technical justification for the meteorological conditions used in the applicant's accident analysis indicating the estimated probability of occurrence of these conditions.

Question (a) was answered by reviewing the substance of NYU Report 372.4 which compared 1956 and 1957 meteorological data. Question b) was answered in part, but the discussion veered toward temperature gradients and plume rise, not returning to probabilities that day. Mr. Jensch raised the question again of conducting diffusion tests. Dr. Halitsky recommended against such test as being unnecessary.

On the following day Dr. Halitsky continued his reply to Question b) pgs. 1795ff). He stated that the Con Edison 30-day meteorology was conservative because inversions were assumed to occur twice as frequently as the average for the year, and the meander was assumed to occur in a 20^{o} sector whereas the actual meander angle was more like 40^{o}. The combination of those two effects introduced a factor of about 4 in the χ/Q calculation.

Turning to the first day meteorology, considerable discussions then ensued regarding the interpretation of the hodographs in Figs. 2.6-1 and 2.6-2, particularly with respect to lapse rates during different hours of the day. Dr. Halitsky pointed out that the Con Edison model ignored meander and reversal during the first day, each of which would introduce a factor of 2 for a total of 4 on the calculated χ/Q. Furthermore, the increase of wind speed from 1 m/s to 2 m/s during the 2-24 hr period appeared justified according to the average hodograph.

Dr. Pigford then brought the questioning back to the probability of occurrence of the Con Edison assumed meteorology (pg 1815). Dr. Halitsky offered an opinion of the probabilities of the first-day meteorology specified in the AEC-DRL and Con Edison models, based on Mr. Spickler's previous estimate (pg 670) of "probably less than 1%" and Dr. Halitsky's estimate of two orders of magnitude less than Mr. Spickler's for the same model; i.e., assuming that the average hodograph occurs 100 days/yr, the AEC-DRL "anomalous" hodographs would occur .01x100 = 1 day/yr according to Spickler and 1 day/100yrs according to Halitsky. The Con Edison "anomalous" hodograph, which is not as severe as the AEC-DRL version, would have an intermediate frequency, say 1 day/10yrs.

Dr. Pigford then requested a statement of probability for each of the three time periods in the Con Edison model. Dr. Halitsky was unable to furnish this information.

Dr. Pigford subsequently questioned Mr. Grob regarding the possibility of return flows over the site causing an accumulation of concentrations (pg 1846), the possibility of accumulation of concentrations due to simultaneous operation of Units 1, 2 and 3 (pg 184) and the possibility of plumes leaving the Unit 1 stack and/or the tops of the Units 2 and 3 containment structures and impinging upon a local rise in terrain beyond the site boundary (pg 1853). Mr. Grob and Dr. Halitsky responded by giving qualitative descriptions of plume behavior and concluding that the postulated conditions would not yield higher concentrations than in the assumed accident meteorology.

Mr. Jensch then queried Mr. Spickler on wake effects with cylindrical structures at low wind speeds (pg 1862). Mr. Spickler cited various references, none of which reported tests in wind speeds as low as 1 m/s. Mr. Spickler concurred with Mr. Jensch as to the desirability of having wake concentration data at low wind speeds to justify inclusion of the wake factor in the meteorological model (pg 1864).

Dr. Halitsky completed his statement on the probability of occurrence of the Con Edison meteorological model by specifying a substantially zero probability since the first two periods in the model do not provide for wind meander which always occurs (pg 1914).

After conclusion of the hearings, both Con Edison and the AEC-DRL submitted written answers to the questions posed by Dr. Pigford at the pre-hearing conference. Con Edison concurred that data were lacking to prove that 8 hr wind persistence under low speed conditions could not occur. It also showed that the AEC-DRL value of $X\bar{u}/Q$ during the 2-8 hr period, while twice as high as the Con Edison value for the same period, would produce only a 20% increase in dose. The AEC-DRL contended that relaxation of their long-term model is not justified until joint probability of persistence with speed and stability can be examined. It also stated that their model showed a 40% increase in dose over the Con Edison model.

The AEC-DRL also provided written comments on Dr. Halitsky's hearing testimony, i.e.: (a) they agreed with his testimony, (b) they believe that the Sutton equations are valid for this type of terrain and that smoke photography and wind measurements are adequate experimental techniques in lieu of direct concentration measurements, and (c) they believe that year to year variations in meteorology will be small and that accident meteorological assumptions would still be quite conservative.

In its Initial Decision granting a Construction Permit (Aug 13, 1969) the Board took note that Con Edison had undertaken a new meteorological program in the Indian Point area, and had stated that the new data would be used in connection with the proposed operations for Unit No. 3. The Board strongly urged that

a) Definitive criteria should be developed for judging the adequacy of the meteorological program (pg 12),

b) the present continuing study should be made as comprehensive as possible (pg 13), and

c) the possibility that ground concentrations higher than those at the site boundary might occur beyond the site boundary should be given detailed consideration (pg 16).

In response to the Board's recommendations, Con Edison revised its ongoing meteorological program in the Fall of 1969 to serve two functions; one was to acquire data which could be used for the scheduling of operational releases, the other was to acquire data which would help to resolve the unanswered questions, which arose during the hearings, regarding the 0-8 hr accident meteorology models.

The portion of the revised program relating to accident meteorology included experiments to obtain data on:

a) annual wind statistics

b) low wind speed hodographs

c) turbulence characteristics under low wind speed inversion conditions

d) persistence statistics

e) building wake effects on diffusion

The results of experiments in the above categories were submitted to Con Edison in three reports. These are

1) N.Y.U. Tech. Rep. TR 71-3 "Wind Observations at Indian Point, 26 November 1969 – 1 October 1970" by J. Halitsky, E.J. Kaplin, and J. Laznow 17 May 1971.

2) N.Y.U. Tech. Rep. TR 71-10 "Low Wind Speed Turbulence Statistics and Related Diffusion Estimates for Indian Point, N.Y/" by D. M. Leahey and J. Halitsky 15 September 1971.

3) "Wind Test of Gas Dispersion from Indian Point Unit 1", by J. Halitsky 29 June 1971.

Item 1) was submitted to the AEC-DRL in Supplement 1, pgs Y-1 to Y-32 of the Unit No. 3 FSAR.

Item 3) was submitted to the AEC-DRL in support of an application to reduce the stack height of Unit No. 1 to meet structural safety requirements under tornado loadings. In preparation for the I.P.2. Operational Permit hearings, Con Edison submitted to the AEC Licensing Board:

(1) Summary of Application (Docket 50-247) dated Nov. 12, 1970, and the AEC-DRL Technical Staff submitted:

(2) AEC-DRL Safety Evaluation dated Nov. 16, 1970,

(3) Appendix C to AEC-DRL Safety Evaluation; Comments on FSAR and Amendments 12 and 14 for I.P.2., dated Nov. 29, 1969 and Feb. 17, 1970 (included as Pgs 93, 94 and 95 of (2) above).

In Item (2) pg 9, the AEC-DRL stated that the two years of (new) meteorological data presented by the applicant provide an adequate basis for selecting the meteorological parameters for both routine and accident meteorology calculations.

In Item (3), ESSA-AREL acknowledged the existence of the diurnal reversal of valley winds at the site but stated that in the absence of joint frequency data, an appropriately conservative assumption would be a steady wind of 1 m/s speed for the first 8 hrs followed by a 22½^{o} meandering wind of 1 m/s speed for the next 16 hrs in the same sector. ESSA characterized the latter assumption as “very conservative”. It also stated that the Con Edison correction for building wake effect agreed well with the AEC-DRL method, but disagreed with the Con Edison procedure of including a virtual source displacement in the long term average diffusion model.

February 1973

A STUDY OF ATMOSPHERIC DIFFUSION CONDITION PROBABILITIES USING THE COMPOSITE YEAR OF INDIAN POINT SITE WEATHER DATA [Historical Information]

by:

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2.6.M-1

Introduction

A study is presented below which compares atmospheric diffusion condition probabilities for the Indian Point site computed using various models. Several of these models realistically take into account characteristics of the Indian Point site. The cases studied account for the following: (1) effect of allowing diffusion for the distance to the nearest land not owned or controlled by Consolidated Edison in each direction (i.e., the effluent is assumed to diffuse for the real distance to the boundary, not just to the minimum site boundary radius), (2) effect of using the "split sigma" model to account for lateral wind meander, and (3) effect of averaging diffusion conditions over a two-hour period. Use of these realistic assumptions result in significant reductions in diffusion estimates.

Background

Meteorological data have been taken on a 100 ft tower at the Indian Point site for several years. To provide the most representative one-year period of data, a "composite year" was constructed using the most complete month from the total period of record available. Following is a summary of the data used.

Parameter	Measured Height	Percent Data Recovery
Wind Speed	100 ft	97.6
Wind Direction	100 ft	98.8
Temperature Difference	95 ft-7 ft	95.5

Basis Diffusion Model

From these data, centerline values of X/Q were computed for each hour of data using the following relationship:

$$Q / X = \frac{1}{\bar{u}(\pi \sigma y \sigma z + cA)}$$

where:

X = concentration $(\mu Ci / m^3)$

Q = release rate $(uCi / \sec)$

$\bar{u}$ © = average wind speed for the hour measured at 100 ft and extrapolated to 33 ft (m/sec)

s y = horizontal diffusion coefficient (m)

s z = vertical diffusion coefficient (m)

cA = building wake factor (assumed to be c = 0.5, A = 2000 m^3).

The building wake effect is limited such that no more than a factor of 3.0 reduction in dilution is obtained for any condition. The wind speed is extrapolated to the 33 ft level in accordance with recent AEC practice. The method of extrapolation is according to the following relationship:

$$\bar{u}_{33} = \bar{u}_{100}\left(\frac{h_{33}}{h_{100}}\right)^n$$

where:

$\bar{u}_{33}$ = extrapolated wind speed (m/sec)

$\bar{u}_{100}$ = measured speed (m/sec)

h_{33} = height extrapolated to (ft)

h_{100} = measured height (ft)

n = exponent based on diffusion as follows

2.6.M-3

Pasquill Diffusion Group	Value of n

A	0.25
B	0.25
C	0.25
D	0.33
E	0.5
F	0.5
G	0.5

Since ground effects influenced the lower sensor at 7 ft, measured values of ΔT are multiplied by a factor as suggested by AEC. The method used to determine this factor assumes an exponential relationship between temperature and height such that measured temperature difference between any two heights can be represented as a temperature difference between two other heights according to the following relationship:

$$f(\Delta T \text{ correction factor} = \frac{\ln\left(\frac{h_{ue}}{h_{le}}\right)}{\ln\left(\frac{h_{um}}{h_{1m}}\right)}$$

where:

h_{ue} = height of upper extrapolated temperature (ft)

h_{le} = height of lower extrapolated temperature (ft)

h_{um} = height of upper measured temperature (ft)

h_{lm} = height of lower measured temperature (ft)

For Indian Point the factor is computed as follows: $f = \frac{\ln\frac{150}{33}}{\ln\frac{95}{7}} = \frac{1.51}{2.60} = 0.58$

2.6.M-4

Thus, all vertical temperature difference values are multiplied by this factor.

In this study, either vertical temperature difference (ΔT) is used or wind direction range (R) is used to determine values of σ in the diffusion equations. The following table gives the values assumed for each Pasquill Diffusion Category.

Pasquill Category	(1) AEC ΔT Model* (ºF/100')	(2) Wind Direction Range (σ)
A	ΔT< -1.0	135 < R
B	-1.0 ≤ ΔT< -0.9	135 ≥ R > 105
C	-0.9 ≤ ΔT< -0.8	105 ≥ R > 75
D	-0.8 ≤ ΔT< -0.3	75 ≥ R > 45
E	-0.3 ≤ ΔT< -0.8	45 ≥ R > 22
F	0.8 ≤ ΔT< 2.2	22 ≥ R > 12
G	2.2 ≤ ΔT	12 ≥ R

*In conversion from °C/100 m (Safety Guide 23) to °F/100 ft, values were rounded to nearest tenth of a degree.

Two basic models are used. One is referred to as the "AEC ΔT Model" which utilizes the Safety Guide ΔT values to determine Pasquill category for use in determining both σ y and σ z. The "Split Sigma Model" determines the diffusion coefficients in the diffusion equation based on both ΔT and wind direction range. In this model the X/Q values are computed assuming that σ_z (controlling vertical diffusion) is related to ΔT as before, however,

σ_y (the horizontal diffusion coefficient) is determined using the wind direction range categories given above. The validity of decoupling σ_y and σ_z has been demonstrated in diffusion tests at Three Mile Island, Pennsylvania (Amendment 24 to FSAR) Docket No. 50-289).

As mentioned previously, one of the models used for this study ("Site Shape") assumes that the site boundary is not circular. Consolidated Edison owns or controls land in certain directions out to a distance significantly greater than the minimum assumed radius of 330 m. Additionally, for bodies of water where there are no permanent residents, it is reasonable to assume that when winds blow toward these directions the X/Q values can be computed for a distance corresponding to that of the opposite shore. The following table lists the distances used for each direction in the "Site Shape" model calculations.

Direction	Assumed Distance (meters)
N	1775
NNE	2375
NE	825
ENE	575
E	1195
ESE	585
SE	1165
SSE	1165
S	1285
SSW	1685
SW	330
WSW	1575
W	1575
WNW	1275
NW	1275
NNW	1275

Calculational Technique

Probability distributions of the X/Q values computed as described above are constructed by connecting all hours which have X/Q values equal to or greater than a selected value. The numbers of hours so obtained are then divided by the total hours in the year of data to obtain the probability that the selected X/Q value would be equaled or exceeded. This procedure is repeated for a number X/Q values which are then plotted to form a probability distribution.

For AEC licensing it is customary to pick the 5% probable hourly (0-1 hr) X/Q for use in the 0-2 hour period of loss-of-coolant accident evaluations. However, in reality, if the wind direction or diffusion conditions change during the two-hour period, a stationary receptor would not receive a dose at the same rate for the full two-hour period. To account for this effect, probability distributions have also been made using two-hour averaging of the X/Q values.

The method of averaging over longer periods is as follows. Starting with each hour of data, the computed X/Q values are added in each of 16 assumed direction sectors for the duration of the release time period being evaluated (2 hours). The maximum integrated value of all the 16 directions is stored and a new integration period is started spaced one hour later. Again, the maximum value for this next integration period is stored regardless of the direction sector in which it occurred, and so on. After processing all hours of data, cumulative probability plots are made for each release time period considered, 2 hours in this

case.

<u>Results</u>

Four cases were run using the composite year of Indian Point data. Each case was run to obtain probability distributions using hourly data (0-1 hr) and using the two-hour averaging technique. Figure 1 attached shows the probability distributions for the one-hour cases and Figure 2 shows the results for the two-hour averaging cases. The following table gives the assumptions and results for each case.

Case #	Model	Site Boundary	5% Probable X/Q (sec/m^3)* (1 hr only)	5% Probable X/Q (sec/m^3)* (2 hr averaging)
1	AEC ΔT	Circular	1.8×10^{-3}	1.3×10^{-3}
2	AEC ΔT	Site Shape	6.8×10^{-4}	5.0×10^{-4}
3	Split s	Circular	9.5×10^{-4}	6.5×10^{-4}
4	Split s	Site Shape	3.7×10^{-4}	2.9×10^{-4}
*Note: For Pasquill F and 1 m/sec wind at 330 m, X/Q = 1.3×10^{-3} sec/m^3.				

<u>Evaluation</u>

Case 1 (1 hour only) is the typical model used by the AEC for the two-hour portion of the LOCA. As shown, the X/Q value is 1.8×10^{-3} sec/m^3. If the actual distance to the site boundary is used as in Case 2, the value reduces to a X/Q of 6.8×10^{-4} sec/ m^3 resulting in a factor of reduction of 2.65. If the lateral meander is accounted for as in the “Split Sigma” Case 3 for a circular site, a X/Q value of 9.5×10^{-4} results. Another meaningful comparison is between the one-hour and two-hour averaging results for Case 1. Here there is a factor of 1.4 reduction for this effect alone.

There are many combinations which can be compared using this table, however, the thrust of this study is to demonstrate that the typical AEC model (Case 1) is not appropriate for this particular site, since it does not account for inherent site characteristics.

Department of Meteorology and Oceanography

NEW YORK UNIVERSITY

COLLEGE OF ENGINEERING

RESEARCH DIVISION

SUMMARY OF CLIMATOLOGICAL DATA AT BUCHANAN, NEW YORK

1956-1957

By

Ben Davidson

Technical Report No. 372.4

Prepared for

Consolidated Edison Co. of N. Y., Inc.

March, 1958

R-1

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R-2

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1. <u>Introduction</u>

A detailed summary of climatological data collected during 1956 is contained in Technical Report No. 372.3 – Evaluation of Potential Radiation Hazard, April 1957. The tower was run on a skeleton basis during 1957. Wind observations were made at 100 and 300 feet (200 and 400 feet above river level), while temperature was observed at 7, 150, and 300 feet above ground. Because of the relative infrequency of calibration and general maintenance during 1957 the 1956 data are considered far more accurate. The 300 ft 1957 data were processed in the same manner as the 1956 data. In the present report we summarize:

(a) The effect of climatological differences between 1956 and 1957 on the radiation calculations of Report 372.3

(b) The local wind rose as a function of height above river, and

(c) The combined 1956-1957 wind rose at 300 feet as a function of stability and wind speed.

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2

2. Comparison of 1956-1957 data

In Table I the essential features of the 1956 and 1957 300 ft data are summarized as a function of stability class. All definitions remain the same as in the previous report. In particular, Inversion conditions (I) are defined to occur when $T_{300} - T_7 \geq 0$; Isothermal-adiabatic conditions (N) when $0 > T_{300} - T_7 \geq -1.8°F$; and Lapse conditions (L) when $T_{300} - T_7 < 1.8°F$.

Table I. Frequency of Inversion (I), Neutral (N), and Lapse (L) conditions with associated mean wind speeds, $\bar{V}$ (mph) for 1956 and 1957.

Summer	I	$\bar{V}$	N	$\bar{V}$	L	$\bar{V}$
1956	0.38	6.5	0.31	10.4	0.31	11.6
1957	0.35	6.2	0.33	12.8	0.32	9.7
Winter						
1956	0.25	7.6	0.54	12.6	0.20	8.5
1957	0.33	7.1	0.48	13.1	0.19	9.0
All seasons						
1956	0.315	6.9	0.425	11.8	0.255	10.4
1957	0.340	6.6	0.405	13.0	0.255	9.4

There are minor differences, but on the whole, the data seem compatible. There were slightly more inversion hours in 1957 than in 1956 with a slightly lower wind speed. The yearly frequency for each temperature gradient condition does not vary more than 10 percent whilst the mean wind speed for each class is also within 10 percent of the 1956 figure. Almost all of the radiation calculations are inversely proportional to the mean wind speed or to the harmonic mean. There is not too great a difference between the two years and for this reasons the total integrated dosage for the area should not vary too greatly, say within 10 to 20 percent,

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which is well within the range of uncertainty of the original calculations.

The areal distribution of radiation contained in Figs. 1.1. and 1.2 of the earlier report depends in the mean on the distribution of wind direction. Fig. 1 is a comparison of the annual distribution of wind direction for 1956 and 1957. Again the differences are not great; the 1957 distribution seems a bit more peaked than the 1956 data. This may be due in part to systematic individual differences in reading the charts. Whatever the cause, the differences in the distribution are well within the limits of accuracy of the initial calculations.

R-6

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3. <u>Variation of wind direction with height</u>

Some idea of the variation of wind direction with height may be gained from the 100 and 300 ft summer wind rose (Figs. 3.1 and 3.2 of the original report). To supplement this information, we compare in Fig. 2 the distribution of wind direction for the 1956 summer season at 400 ft (300 ft tower level), 200 ft (100 ft tower level) and 70 ft above river. The 70 ft data were obtained from an anemometer mounted on the "Jones," a ship anchored in mid-river. The ship site is about 0.8 mile northwest of the tower (see map in Report 372.1). It is evident that there are systematic differences in the three distributions. The most obvious is the build-up of southerly winds with height. The Jones distribution is flat from 150^{o} to 250^{o}, while the 100 and 300 ft tower

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distributions peak fairly at 170°. On the down valley side of the distribution (about 020°), The Jones and 100 ft tower level distributions are fairly well matched. The 300 ft tower level distribution does not reach nearly the same frequency at 030° as do the other two distributions. Some of the essential differences in the two distributions are summarized in the following table.

Percent time indicated wind direction ranges were observed at

Direction Range	Jones	100 ft Tower	300 ft Tower
340-040	38	37	30
360-040	28	30	19
160-220	16	23	27
160-200	10	18	22

Part of the difference between the distributions can be explained by the tendency for light southerly winds to be observed at the 300 ft tower level when the nocturnal NNE winds have set in at the Jones and 100 ft tower locations. The remainder appears to be a daytime phenomenon and indicates that The Jones distribution is affected by the proximity of the valley walls in a rather complicated fashion.

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4. Wind rose presentation

In Fig. 3 we present wind roses based upon two years of data for inversion, neutral, and lapse conditions at the 300 ft level. The bars here are flying with the wind and pointing to the indicated meteorological wind direction. The length of the bar is proportional to the average frequency of occurrence per year of the appropriate wind direction and stability condition. For convenience in interpretation we indicate the general location of populated areas surrounding the site.

An interesting feature of the wind rose is the elongation along the axis of the valley during inversion hours. Wind trajectories towards Peekskill, the most densely populated area near the site, are relatively infrequent during neutral and lapse conditions. There is a sizeable frequency of 210° winds during inversion hours. This trajectory would just about brush the northern outskirts of Peekskill, but it is probable that terrain effects would tend to curve the trajectory so that it follows the river. In general, the inversion wind rose shows a high frequency of up and down valley wind directions.

During lapse and neutral conditions, the wind rose indicates a substantial frequency of northwest winds which are the prevailing winds over flat land in this area. Under these temperature gradient conditions, one may expect effluent concentrations on the ground. There are a substantial number of wind trajectories toward the villages of Buchanan, Montrose and Verplank during neutral and lapse conditions, and towards the village of Verplank during inversion conditions.

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8

References

Davidson, B., and J. Halitsky, 1955: A micrometeorological survey of the Buchanan, N.Y. area. - Summary of progress to 1 December 1955. Technical Report No. 372.1, Research Division, New York University, College of Engineering.

Davidson, B., and J. Halitsky, 1957: Evaluation of potential radiation hazard resulting from assumed release of radioactive wastes to atmosphere from the proposed Buchanan nuclear power plant. Technical Report No. 372.3, Research Division, New York University, College of Engineering.

R-11

Station: BEAR MOUNTAIN, NEW YORK

Drainage basin: HUDSON **County:** ORANGE

Lat. 41° 19' N **Long.** 74° 00' W **Elev.** (ft.) 1301

Period of record: 1941-1950

Month		Duration (hours)					
		1	2	3	6	12	24
Jan.	Amt.	0.38	0.60	0.85	1.29	1.49	1.51
	Date	7/1946	31/1942	31/1942	1/1945	31/1942	5-6/1949
Feb.	Amt.	0.26	0.41	0.56	0.80	1.12	1.42
	Date	14/1944	14/1944	14/1944	14-15/1944	20-21/1947	20-21/1947
Mar.	Amt.	0.35a	0.57	0.78	0.99	1.19	1.41
	Date	21/1948	3/1942	3/1942	3/1942	6-7/1944	2-3/1947
Apr.	Amt.	0.61	0.89	1.06	1.51	1.71	2.08
	Date	30/1947	30/1947	1/1948	1/1948	1/1948	18-19/1949
May	Amt.	0.70	1.21	1.35	1.77	2.51	2.87
	Date	6/1949	20/1949	30/1948	27/1946	27/1946	27/1946
Jun.	Amt.	0.67	0.83	0.88	1.01	1.50	1.82
	Date	21/1945	21/1945	21/1945	23/1942	2/1946	1-2/1946
July	Amt.	1.57	1.72	1.85	2.47	2.74	3.98
	Date	20/1945	22/1946	22/1946	22-23/1945	22-23/1945	18-19/1945
Aug	Amt.	1.25	1.44	1.71	1.93	2.30	2.47
	Date	26/1947	16/1942	16/1942	16/1942	9/1942	24-25/1945
Sep.	Amt.	0.81	1.21	1.71	2.08	2.28	2.80
	Date	30/1946	24/1946	24/1946	24/1946	24/1946	26-27/1942
Oct.	Amt.	0.59	0.86	1.03	1.53b	2.83	3.95
	Date	10/1950	26/1943	26/1942	26/1942	26-27/1943	26-27/1943
Nov.	Amt.	1.18	1.97	2.22	3.14	3.65	3.65
	Date	8/1947	8/1947	8/1947	8/1947	8/1947	8/1947
Dec.	Amt.	0.63	1.17	1.48	1.99	2.09	3.33
	Date	25/1945	25/1945	25/1945	25/1945	25-26/1945	30-31/1948

Annual	Amt. Date	1.57 7/20/45	1.97 11/8/47	2.22 11/8/47	3.14 11/8/47	3.65 11/8/47	3.98 7/18-19/45

[a] Also 23/1949 [b] Also 26/1943.

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U.S. DEPARTMENT OF COMMERCE, WEATHER BUREAU

Station, Bear Mountain County, Orange State, New York

Latitude, 41.19 Longitude, 74.00 Elevation, 1300 feet.

Data, Precipitation, Monthly and Annuals

Year	January,	February,	March.	April,	May,	June,	July,	August,	September,	October,	November,	December,	Annual
1939	3.81	3.62	3.16	5.90	1.30	5.32	3.04	3.36	3.04	4.20	1.69	3.39	41.83
1940	5.58	3.87	5.72	6.68	6.53	3.12	3.68	4.05	2.82	3.38	4.48	3.87	53.78
1941	2.77	2.87	2.22	2.00	1.79	4.46	6.31	3.33	0.25	2.35	3.18	4.47	36.00
1942	3.98	1.85	5.67	0.92	3.20	3.80	5.79	5.51	4.44	3.61	4.79	4.62	48.18
1943	2.36	1.19	2.00	3.47	4.56	3.80	3.73	2.56	2.99	12.64	4.18	[a] 1.01	44.49
1944	1.93	2.05	5.60	5.30	2.54	3.06	2.03	2.42	5.99	2.12	5.09	2.37	40.50
1945	2.97	2.46	1.79	3.79	7.18	4.28	16.87	4.73	5.36	2.13	6.53	4.46	62.55
1946	1.79	1.65	2.97	1.97	8.91	3.11	8.10	4.93	6.24	2.13	1.03	2.48	45.31
1947	2.85	3.39	3.48	4.76	9.49	6.55	7.38	2.78	[a] 1.90	2.69	8.51	3.68	57.46
1948	3.05	1.21	3.29	5.28	7.30	4.84	3.52	2.76	0.68	1.92	4.90	6.14	44.89
1949	5.08	2.27	1.88	5.47	6.53	0.96	3.45	2.94	5.60	2.52	1.91	2.79	41.40
1950	2.81	4.46	3.40	2.97	6.02	3.77	5.16	2.94	2.26	2.45	5.39	6.24	47.87
1951	4.60	4.14	8.40	2.94	4.11	3.87	5.07	5.19	2.06	5.13	7.55	5.96	59.02
1952	4.53	3.22	5.46	8.54	5.29	5.92	5.13	8.13	5.01	0.52	4.48	5.84	62.07
1953	6.75	1.89	5.71	4.72									
Sums													
Means													

REMARKS

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U. S. DEPARTMENT OF COMMERCE

WEATHER BUREAU

NATIONAL WEATHER RECORDS CENTER

JOB NO. 6729

SURFACE WIND SPEEDS VERSUS
DIRECTION WHEN SOME
FORM OF PRECIPITATION

STATION: BEAR MOUNTAIN, NEW YORK

PERIOD: JANUARY 1944 – DECEMBER 1948

Sponsored by: Consolidated Edison Company of New York,

Date October 28, 1965

FEDERAL BUILDING
ASHEVILLE, N.C.

Book 2 of 2

USCOMM. WB-ASHVILLE

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U. S. DEPARTMENT OF COMMERCE

WEATHER BUREAU

NATIONAL WEATHER RECORDS CENTER

JOB NO. 6729

OCCURRENCE OF WIND SPEED
AND DIRECTION DURING
THUNDERSTORMS

STATION: BEAR MOUNTAIN, NEW YORK

PERIOD: JANUARY 1944 – DECEMBER 1948

Sponsored by: Consolidated Edison Company of New York,

Date October 28, 1965

FEDERAL BUILDING
ASHEVILLE, N.C.

Book 1 of 2

USCOMM. WB-ASHVILLE

R-18

New York University
School of Engineering and Science
Department of Meteorology and Oceanography
Geophysical Sciences laboratory
Technical Report No. TR 71-3

Wind Observations at Indian Point
26 November 1969-1 October 1970

Prepared by

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for

Consolidated Edison Co. of N. Y., Inc.
4 Irving Place, New York, N. Y.   10003

17 May 1971

Y-1

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Summary

Wind observations made at a 100 ft meteorological tower at Indian Point and at a ship anchored in the Hudson River northwest of Indian Point in 1969-70 were compared with observations made at similar installations in 1955-56. It was found that

1) Annual average statistics of wind speed, direction and vertical temperature difference were substantially the same for 1956 and 1970. Points of difference were increased frequencies of lapses and low wind speeds and a shift in the southerly frequency maximum to the southeast in 1969-70. The low-speed inversion frequency was unchanged.

2) Average wind hodographs at the ships exhibited the same diurnal reversal pattern and the same 2.5 m/sec nighttime downvalley speed in both years. The average wind hodograph at the tower showed a similar pattern of reversal but the nighttime downvalley speed was about 2m/sec.

3) All sixteen daily wind hodographs used for calculating the aver age hodograph at the tower showed the diurnal reversal and exhibited considerable variability in speed and direction from day to day through a complete cycle.

4) Maximum persistences of low-speed inversion winds in the critical 005° - 020° sector were 2 hrs, 4 hrs and 3 hrs for 1, 1.5 and 2 m/s speeds, respectively, during the entire 10-month data record.

Y-2

1. Introduction

This is the second of two progress reports covering meteorological investigations in the Hudson River valley near Indian Point from August 1968 to the present.

The first report [Halitsky, Laznow and Leahev (1970)] described wind measurements at Indian Point, Bowline Point and Montrose until 30 June 1969, and provided details of changes in tower location and instrumentation introduced during the period July-November 1969.

This report presents an analysis of measurements taken at the present tower at Indian Point (IP 3) and at a ship, the Cape Charles (CC) anchored in the Hudson River, and compares them with similar measurements taken in 1955-1957 at approximately the same locations. The focus of this report is to evaluate whether site meteorology has changed significantly during the intervening years, and to elucidate aspects of the meteorology not reported previously.

In order to clarify the various tower locations and periods of operation, the following nomenclature was established in the first report and will be continued.

2

Date Period	Station Symbol	Station Location
1955	J	Ship “Jones” in Hudson River
1956-1957	I P 1	Indian Point, southeast of plant
1968-1969	I P 2	Indian Point, southwest of plant
1968-	B P	Bowline Point
1968-	M P	Montrose Point
1969-	I P 3	Indian Point (close to I P 1)
1970	C C	Cape Charles (close to J)

Figs 1, 2 and 3 show the station locations and local topography.

2. Data Log

Fig 4 shows the periods of data acquisition for all of the stations which were in operation in 1970. Station 3 P is included, even though its operation is now being funded by Orange and Rockland Utilities, Inc., order to show the total store of data for the region. The net radiometer (R) and ambient temperature (T) at I P 3, and the Aerovane (A) at the Cape Charles are supplementary instruments provided by N. Y. University.

All of the instruments except the bivane produced continuous records on slow-speed strip charts 91 inch, 2 inch or 3 inch per hour). The bivane chart drive was modified to run 50 minutes at 3 inch per hour followed by 10 minutes at 3 inches per minute and repeat. Thus, each chart (indicated by a dot in Fig 3) contained a 36-hr record of fast-and slow-speed data for each hour.

Y-7

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The statistical data presented in this report represent periods when simultaneous wind speed, wind direction and temperature difference were available at I P 3. The overall period selected for analysis was 26 November 1969-1 October 1970. The degree of completeness of record is as follows:

	Hours all data present	Total Hours in period	% completeness
Climet	5989	7440	80.5
Aerovane	6164	7440	82.8

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4

3. Annual Average Wind Statistics at the I P 1 and I P 3 Stations

The annual average wind statistics at I P 3 for the 10-month measurement period in 1969-1970 are shown in graphical form in Figs 5 (a-h), 6 and 7. Also included in these figures, for comparison purposes, are the equivalent I P 1 statistics originally reported by Davidson and Halitsky (9157), Table 3.3 and subsequently incorporated into Se 2.6 of the Unit 2 FSAR.

The two sets of Aerovane statistics represent observations taken about 13 years apart with similar or identical instruments at almost the same locations. As seen in Fig 1, the two towers are about 200 ft apart, and the base of the I P e tower is about 15 ft lower in elevation. The present site topography has fewer trees, more pavement, and new steel and concrete structures in the quadrant northwest of the tower.

Wind speed and direction were measured at the 100 ft elevation on each tower; therefore the absolute elevation of the I P e instrument is about 15 ft lower than that of the I P 1 instruments.

Temperature differences were measured between 95 ft and 7 ft on the 100 ft high I P 3 tower whereas 150 ft and 7 ft were used on the 310 ft high I P 1 tower. However, the isothermal and adiabatic lapse rates were used to separate the lapse, neutral and inversion categories in both cases. This was accomplished by using an adiabatic ΔT of – 0.5 F for I P 3 in place of the – 0.9 F used for I P 1.

Fig 5 shows the frequency distribution of wind directions as measured by the Aerovanes in 1956 and 1970.

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Fig 5a represents all winds irrespective of speed and temperature gradient class. The shapes of the curves are quite similar, the most important difference being a shift of the 1956 southerly maximum toward the southeast in 1970.

Fig 5 (b0d) shows the dependence on temperature gradient class. No major change is apparent in the neutral class, but the 1970 data show more frequent lapses and less frequent inversions in all directions.

Fig 5 (e-h) shows the dependence on speed class. The southeasterly shift observed in Fig 5a is seen to occur in the 5-8 mph and 9-13 mph speed classes. Low-speed winds in the 1-4 mph class were more frequent in 1970, especially for the 000°-045° direction range.

None of the above differences are sufficiently large to invalidate the 1956 wind statistics reported in Davidson and Halitsky (1957). Of the three noticeable differences, the decrease in frequency of inversions and the increase in southeasterly winds both contribute to reducing the concentrations in inhabited regions contiguous to the site. However, the increase frequency of low-speed winds from the northeasterly sector bears further examination.

Fig 6 shows a comparison of cumulative frequencies of wind speed for the two years. The 1956 curves can not be extended below 2 m/s because the published data show only two categories below that speed, i.e., calm and 1-4 mph, covering speeds from 0 – 4.5 mph. The cumulative frequency shown at a speed of 2 m/s is the sum of these two categories. The 1970 data were classified in finer groupings and yielded well-defined curves in the low speed range. The 1970 data in Fig 6 were uncorrected for speed calibration. It is not known if corrections were applied to the 1956 data.

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The upper curves, representing all temperature gradients and directions, show good agreement for the two years. The inversion curves show good agreement during calms and near 2-3/m/s, but the 1970 inversion frequencies were smaller than the 1956 frequencies at the higher speeds. This discrepancy in high speed inversions is in the direction of enhancing the atmospheric diffusion potential over that which was postulated on the basis of the 1956 data. It is not known how much of the difference between the two years is due to the absence of October and November data in 1970.

Because of the high starting speed of the Aerovane, the curves of Fig 6 show spuriously high frequencies of low wind speeds. When the 1970 data are corrected for speed calibration (see Fig 6 of Halitsky et al (1970)], the data appear as in Fig 7.

In order to check the Aerovane data, we have included in Fig 7 the corresponding curves obtained from the more sensitive Climet instrument at the same location during the same period, corrected for speed calibration.

The difference between the Aerovane and Climet curves may be attributed to the poor behavior of the aerovane at low speeds. A true speed of 1 m/s is near the starting threshold of the Aerovane. The corresponding indicated speed may be anything in the range 0-2 mph or one division of the chart. At the same time, a one-division indication may be simply a zero setting error. For these reasons, it is believed that the Climet data should be regarded as more reliable.

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4. Valley Wind Hodographs During Virtually-Zero Pressure Gradient Conditions

4.1 Average Hodographs

Average wind hodographs taken during the months of September and October 1955 are presented in FSAR Sec 2.6, Figs 2.6-1 and 2.6-2, to demonstrate that the wind reverses diurnally when the upper air (geostrophic) wind is zero or weak, thereby precluding the occurrence of protracted periods of calm or light wind.

The 1955 data were taken with an Aerovane mounted 70 ft above river elevation on the mast of a ship, the Jones, anchored in the Hudson River about one mile northwest of the tower (see Fig 2). Thirty-five days, during which weak pressure gradient conditions existed over the area, were selected for study. Of these 35 days, 12 days had virtually zero pressure gradient. The two Figures represent the average of wind vectors over the 12 or 35 day period, for each even-numbered hour during the day.

Both of the 1955 hodographs show a well-defined steady flow toward the SSW (030° winds) during the night 92000-0800 hrs), and a somewhat less steady flow toward the NNE (210° winds) during the day (1200-1600). During the transition hours (1000 and 1800 hrs) the flow was weak and variable. The average wind speeds during the night were about 2.5 m/s.

On the basis of these data, it was concluded that the accident meteorology model calling for a wind sequence of 1 m/s steady for two hours followed by 2 m/s steady in the same direction for 24 hours was conservative since the hodograph showed a wind reversal after 12 hours. However, it has been pointed out that individual hodographs for each of the days may have exhibited lower wind speeds and may have failed to show the diurnal reversal.

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To explore this aspect, an Aerovane was installed 100 ft above river level on the mast of another ship, the Cape Charles, anchored in the Hudson River close to the former location of the Jones. The instrument was in operation from March 17, 1970 to Sept. 17, 1970. It had been hoped that the period could be extended to the end of October to gather test data for the same months that were used in the 1955 study, but the instrument had to be removed prematurely because the ship was being prepared for removal.

Using the available record, we selected the two-month period July 15-Sept. 15 as having the closest seasonal correspondence to the 1955 study, and found 17 days during which virtually-zero pressure gradient conditions existed, as determined frm surface weather maps for 0700 EST. The hourly wind velocity vector was determined for each even-numbered hour and a vector average was taken over the 17 days for those hours. The average hodograph is shown in Fig 8, together with the 1955 Jones hodograph.

The important characteristics of the 1955 hodograph were confirmed by the 1970 data. A predominant, diurnally-reversing circulation exists along the 030°-210° axis. The nighttime down-valley flow was slightly weaker (~ 2.0 m/s vs ~ 2.4 m/s) and began about an hour later (2100 vs 2000 hrs) in 1970 but both terminated at ~ 0900 hrs. The up-valley daytime flow was also somewhat weaker (~ 1.5 m/s vs ~ 2 m/s), and did not show the strong southerly wind at 1400 and 1600 hrs. The latter effect may be due to the more northerly locations of the Jones, near the nose of Dunderberg Mtn., where the flow direction changes rapidly (see Fig 2).

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Before analyzing the individual hodographs for each day, it should be noted that the Jones and Cape Charles are located very close to the steep southerly side of Dunderberg Mtn. (peak el. 1120 ft), and are therefore exposed to air currents which tend to flow parallel to the hillside. This topographic influence is not present at the plant site.

To determine what differences, if any, exist between the winds at the ships and the plant site, an average hodographs for 16 of the 17 days was calculated from the records of the Climet speed and direction instrument at the 100 ft elevation on the I P 3 tower (Fig 9). The Climet instrument was inoperative on 1 of the 17 days). The most significant change from the Cape Charles hodograph is the appearance of a southeasterly wind component during the afternoon and evening hours. This component also appeared at the Jones in 1955. Apparently this is an integral part of the valley circulation, causing the hodograph vector to rotate counterclockwise with increasing time, and was not experienced at the Cape Charles due to the deflecting influence of the hillside. A northwesterly down-slope wind may also have been present during the afternoon and evening at the Cape Charles, since the hillside is in shadow at that time.

4.2 Daily Hodographs

Fig 10 (a-d) shows the 16 daily hodographs from which the average hodograph at the I P 3 tower, shown in Fig 9, was calculated. The nighttime down-valley flow appears in all 16 cases. The daytime up-valley flow is quite variable in both speed and direction, and is characterized by generally higher wind speeds and wider direction swings.

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5. Persistence of Low-Speed Winds

Fig 11 contains a time history of each of the wind conditions during the night hours of the days corresponding to the hodographs of Fig 10 (a-d). The graphs are the variation of the wind speed with time for those hours when the wind direction was between 000º and 045º. We shall assume that the wind direction was steady if it remained in this 45º sector. (This is quite conservative, since a wind which meanders uniformly in a 45º sector of 500 m radius under inversion conditions produces an average concentration about 8 times smaller than the steady wind axial concentration.)

The observed wind angle ≈ and temperature differences $\Delta T = T_{95} - T_7$ are noted under each observation. Positive values of ΔT indicate inversions. ΔT values between –0.5 and 0 indicate neutral. ΔT values smaller than – 0.5 designate lapses.

The longest period of direction persistence was 8 hrs, occurring on July 25-26, Aug. 8-9, Aug. 13-14, Sept. 12-13 and Sept. 13-14. The average winds speed in each case was at least 2 m/s. ΔT was recorded only 3 of these days, and an inversion occurred only during the first two hours of one of the days.

The period of poorest dispersion potential occurred on July 24-25. It lasted 6 hours with a gradual increase of wind speed from 0.2 to 2.0 m/s, a gradual decrease of temperature gradient from ΔT = 1.7 to – 0.8, and a gradual direction change from 007º to 043º. The occurrence of the strongest inversion during the early part of the night and its subsequent weakening and change to neutral or lapse beyond 0200 hrs seems to be a common phenomenon at the site.

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Wind persistence may also be examined by listing the number of occurrences that wind of specified characteristics persisted for a specified number of consecutive hours during the entire 10 month test period in 1969-70. The following talbe shows these data for inversion condition only.

Table 1. Wind Persistence at I P 3 Under Inversion Conditions 91969-1970)

Wind	No. of	Maximum speed in sequence (mph)							
Sector	Consec. Hrs.	0.3	0.5	1.0	1.5	2.0	3.0	4.0	6.0
005º-020º	1	1	2	22	41	64	115	141	151
	2			1	3	2	7	5	2
	3				3		2	2	3
4				1					
	10								1
005º-020º	1			16	42	75	155	189	198
	2			1	2	5	16	7	3
	3					1	2	2	1
	4						2	3	
	5							2	1
	6							1	
	7						1		

It is seen that very light winds do not persist beyond one hour, and high persistences begin to appear at about 3 m/s. For both sectors combined, the longest persistences for 1l.0, 1.5 and 2.0 m/s winds were 2 hrs, 4 hrs and 3 hrs, respectively.

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12

References

Davidson, B. and J. Halitsky (1957): Evaluation of Potential Radiation Hazard Resulting from Assumed Release of Radioactive Wastes to Atmosphere from Proposed Buchanan Nuclear Power Plant. N. Y. University Dep't. of Meteorology and Oceanography Tech. Rep't. No 372.3

Halitsky, J., J. Laznow and D. Leahey 91970): Wind Observations at Indian Point, Montrose and Bowline Point. 31 August 1968 to 30 June 1969. N. Y. University Dep't. of Meteorology and Oceanography Tech. Rep't. TR-70-3.

FSAR: Final Facility Description and Safety Analysis Report. Consolidated Edison Co. of N. Y., Inc. Nuclear Generating Unit No. 2. Exhibit B-8.

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[Historical Information]

New York University
School of Engineering and Science
Department of Meteorology and Oceanography
Geophysical Sciences Laboratory
Technical Report No. TR-73-1

Meteorological Observations at Indian Point, Trap Rock, Montrose Point and Cape Charles
1 January 1970-31 December 1971

Prepared by

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Edward J. Kaplin, Project Director

for

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4 Irving Place, New York, N.Y.   10003

15 December 1972

Z-1

Summary

Meteorological data collected at Indian Point 3, Trap Rock, Montrose Point and the Cape Charles in 1970 and 1971 were analyzed annually, seasonally and diurnally. The results were compared with observations made by Davidson in 1955-57 at similar installations.

Results

A) Comparison between two years of Indian Point 3 (1970 and 1971) data with Indian Point 1 (1956) confirm the findings of Halitsky, et al. (1970) that any apparent meteorological differences during the intervening years are such that they favor an increase in the diffusion potential at the Indian Point site. Improvements include: a) a substantial decrease of the occurrence of inversions and b) a decrease in the probability of low wind speeds in the critical quadrant during inversion conditions.

B) A shift of the secondary maximum direction to a more southeasterly orientation was found in the annual direction frequency density distribution.

C) Seasonal differences included a binodal distribution of the summer wind directions with SSE and NNE peaks. The winter curve is tri-nodal with N, WNW-NW and SSE peaks in descending order of magnitude. The percentage of low wind speeds is highest during the summer when neutral and lapse conditions are most prevalent.

D) The seasonal variation of the diurnal mean wind direction and speed is clearly defined. A direction shift occurs during the summer with upvalley winds during the day and downvalley winds at night. The IP3 diurnal curves can be expected to rotate through 360^{o}.

E) The seasonal diurnal persistence of the wind is maximum at night and minimum during the day. The winter persistence values are approximately twice those of the summer.

The winter has a single minimum occurring between 1800-2100 while the summer minimum is binodal: 1000-1300 and 1900-2000.

The winter maximum persistence occurs between 0500-0900 while the summer maximum generally occurs between 0500-0600.

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1.

1. Introduction

This is the third of three progress reports covering the continuous meteorological data collection program in the Hudson River Valley at Consolidated Edison's Indian Point nuclear generating complex in Buchanan, New York (Figure 1).

The first report (Halitsky, Laznow and Leahey (1970)) covered wind observations at the Indian Point 2 (IP2), Montrose Point (MP) and Bowline Point (BP) meteorological tower sites, and temperature difference observations at Indian Point 2 and Bowline Point for the period 31 August 1968-30 June 1969.

The second report (Halitsky, Kaplin and Laznow (1970)) covered wind and temperature difference observations at Indian Point 3 (IP3) for the period 26 November 1969-1 October 1970, and wind observations from the U.S.S. Cape Charles for the period 16 March 1970-18 September 1970. The data were compared with similar measurements taken in 1955-1957 by Davidson ((Davidson and Halitsky (1957) and FSAR)) at approximately the same locations. The analysis was made to determine if significant changes had occurred in onsite meteorology during the intervening years.

The function of the present study was to provide meteorological data to support analyses for nuclear units at Indian point as specified by the following items:

a) Turbulence analysis at Indian Point;
b) Analysis of spatial variability of winds between Indian Point 3, Trap Rock (TR) and Montrose Point;
c) Acquisition of wind statistics at Indian Point 3 and Trap Rock; and
d) Additional instrumentation at Trap Rock to develop annual wind statistics and turbulence measurements comparable to those being collected at Indian Point 3.

Item (a) was satisfied, in part by Leahey and Halitsky (1971) whose study developed the techniques and procedures for the analysis of diffusion parameters at Indian Point and provided initial results.

Items (b) and (c) are covered in this report. Seasonal, diurnal and annual analyses are provided and the comparison with the 1955-57 data has been extended through December 1971.

The instrumentation required to satisfy item (d) was obtained but due to de-emphasis of the Verplank site (TR) it was never installed.

2. Instrumentation Characteristics

Meteorological station locations, tower base elevations and information pertaining to instrumentation, parameter of measurement, elevation of sensor above tower base and period of record are presented in Table 1.

2.1 Sensors

a) Climet Wind Speed: threshold, 0.6 MPH; accuracy, ± % or ± 0.15 MPH; distance constant, 5 feet. (Distance constant is defined as the feet of air required to pass through the transmitter to give 63% of a sharp change.)

b) Climet Wind Direction: threshold, < 1 MPH; damping ratio, 0.4; accuracy, ± 3^{o}.

c) Aerovane Wind Speed: threshold, 2.5 MPH; distance constanct, 15 feet.

d) Aerovane Wind Direction: distance constant, 34 feet.

e) Net Exchange Thermal Radiometer: temperature compensation to an accuracy of ± 1% over a range of –20 to ±160^{o} F.

f) R. M. Young-Gill Bivane: threshold, 0.3-0.5 MPH for both azimuth and eletion.

g) Honeywell-Brown Resistance Bulb: when placed in specially designed wells, the time constant is about 3 minutes in a wind of 20 fps, the bulbs are in shielded gold leafed cylinders and aspirated at

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about 20 fps (Davidson and Halitsky (1955)).

h) Rosemount Nickel Resistance Bulb (Temperature Difference): response time 63% < 90 sec. With 15 fps air flow; accuracy, ± 0.5% at icepoint, linear to 0.05% of full scale; overall accuracy, < ± 0.1° F.

i) Rosemount Nickel Resistance Bulb (Ambient Temperature): same as (h) except overall accuracy, ± 0.1° F.

j) Foxboro Dewcel: if ambient temperature ∃ 32° F, dew point accuracy, ± 0.5° F; if ambient temperature ∃0° F < 32° F, dew point accuracy, ± 1.0° F; if ambient temperature < 0° F, dew point accuracy, ± 1.5° F.

Items (h), (i), and (j) are mounted in Climet aspirated shields.

Table 1. Instrumentation Data

Meteorological Station Location	Base Elev. (ft) Above River	Operating Period	Parameter	Instruments	Elev. (ft) Above Base
Indian Point 3 (IP3)	120	1 Jan 70-Present	Wind Speed	Climet	100
			Wind Direction	Climet	100
		1 Jan 70-20 July 71 & 24 July 72-Present	Wind Speed	Aerovane	100
			Wind Direction	Aerovane	100
		14 Aug 70-Present	Net Radiation	Thermal Radiometer	30
		May 70-Dec 70 (Intermittent)	Turbulence	Bivane	100
		1 Jan 70-25 Oct 71	Temperature Difference	Honeywell-Brown Resis-tance Bulb	95-7
		14 Aug 72-Present	Temperature Difference	Rosemount Nickle Resis-tance Bulb	10-100 10-30
			Ambient Temperature	Rosemount Nickle Resis-tance Bulb	30
			Dew Point	Foxboro Dew-cel	100, 30, 10
Trap Rock (TR)	90	1 Jan 70-27 July 72	Wind Speed	Climet	100
			Wind Direction	Climet	100
Montrose Point (MP)	60	1 Jan 70-7 June 71	Wind Speed	Climet	100
			Wind Direction	Climet	100
Cape Charles (CC)	0	16 Mar 70-18 Sept 70	Wind Speed	Aerovane	100
			Wind Direction	Aerovane	100

Table 4. Indian Point 3 Frequency Distribution of Wind Speed and Direction at 100 ft Tower Level According to Temperature Gradient Class-January 1, 1970-December 31, 1970

Wind Direction	Temp. Grad.	Wind Speed (m/s) 0.0-0.5	>0.5-1.0	>1.0-1.5	>1.5-2.5	>2.5-3.5	>3.5-4.5	>4.5-5.5	>5.5-8.5	>8.5-11.0	>11.0	Miss	Total
349- 11	I	.0016	.0031	.0046	.0042	.0038	.0019	.0002	.0005	.0000	.0000	.0001	.0200
	N	.0009	.0010	.0018	.0063	.0093	.0086	.0063	.0045	.0006	.0001	.0015	.0408
	L	.0000	.0007	.0010	.0056	.0104	.0071	.0043	.0050	.0003	.0000	.0001	.0346
	M	.0003	.0008	.0007	.0021	.0040	.0047	.0029	.0019	.0002	.0000	.0000	.0176
	T	.0029	.0056	.0081	.0182	.0274	.0223	.0137	.0119	.0011	.0001	.0017	.1131
12- 33	I	.0003	.0022	.0047	.0125	.0102	.0050	.0013	.0002	.0000	.0000	.0002	.0366
	N	.0002	.0016	.0033	.0091	.0131	.0085	.0021	.0022	.0008	.0001	.0018	.0429
	L	.0001	.0006	.0015	.0067	.0073	.0034	.0018	.0025	.0000	.0000	.0001	.0241
	M	.0001	.0010	.0018	.0080	.0085	.0049	.0022	.0006	.0000	.0000	.0000	.0271
	T	.0008	.0054	.0113	.0364	.0391	.0218	.0073	.0055	.0008	.0001	.0022	.1307
34-56	I	.0007	.0031	.0050	.0090	.0041	.0010	.0003	.0000	.0000	.0000	.0006	.0239
	N	.0002	.0006	.0023	.0043	.0032	.0011	.0003	.0003	.0005	.0000	.0011	.0141
	L	.0002	.0009	.0016	.0048	.0043	.0009	.0006	.0007	.0001	.0000	.0002	.0144
	M	.0002	.0009	.0015	.0051	.0048	.0018	.0007	.0001	.0000	.0000	.0000	.0152
	T	.0014	.0055	.0104	.0233	.0165	.0049	.0019	.0011	.0006	.0000	.0019	.0676
57-78	I	.0009	.0017	.0022	.0019	.0007	.0001	.0001	.0000	.0000	.0000	.0000	.0077
	N	.0003	.0006	.0016	.0011	.0007	.0002	.0000	.0001	.0000	.0000	.0005	.0051
	L	.0001	.0005	.0007	.0014	.0003	.0003	.0000	.0005	.0001	.0000	.0002	.0041
	M	.0001	.0001	.0010	.0018	.0013	.0007	.0005	.0000	.0000	.0000	.0000	.0055
	T	.0015	.0029	.0055	.0063	.0030	.0014	.0006	.0006	.0001	.0000	.0007	.0224
79-101	I	.0001	.0013	.0011	.0016	.0007	.0001	.0000	.0002	.0000	.0000	.0000	.0051
	N	.0002	.0005	.0014	.0016	.0011	.0009	.0006	.0000	.0000	.0000	.0002	.0065
	L	.0000	.0006	.0007	.0008	.0005	.0001	.0000	.0003	.0000	.0000	.0000	.0030
	M	.0000	.0005	.0005	.0018	.0007	.0001	.0003	.0001	.0000	.0000	.0000	.0040
	T	.0003	.0027	.0037	.0058	.0030	.0013	.0009	.0007	.0000	.0000	.0002	.0186

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Table 4 (continued)

Wind Direction	Temp. Grad.	Wind Speed (m/s) 0.0-0.5	>0.5-1.0	>1.0-1.5	>1.5-2.5	>2.5-3.5	>3.5-4.5	>4.5-5.5	>5.5-8.5	>8.5-11.0	>11.0	Miss	Total
102-123	I	.0005	.0008	.0011	.0011	.0011	.0000	.0001	.0000	.0000	.0000	.0001	.0049
	N	.0005	.0015	.0009	.0014	.0015	.0007	.0006	.0003	.0000	.0000	.0003	.0077
	L	.0001	.0005	.0006	.0018	.0002	.0008	.0002	.0002	.0000	.0000	.0000	.0045
	M	.0003	.0006	.0007	.0006	.0010	.0006	.0008	.0001	.0000	.0000	.0000	.0047
	T	.0014	.0033	.0033	.0049	.0039	.0021	.0017	.0007	.0000	.0000	.0005	.0217
124-146	I	.0003	.0014	.0023	.0019	.0014	.0006	.0000	.0000	.0001	.0000	.0001	.0081
	N	.0003	.0010	.0014	.0032	.0027	.0018	.0006	.0006	.0003	.0000	.0001	.0121
	L	.0001	.0006	.0016	.0018	.0015	.0026	.0009	.0017	.0002	.0000	.0000	.0111
	M	.0003	.0009	.0005	.0015	.0022	.0007	.0001	.0002	.0000	.0000	.0000	.0064
	T	.0011	.0039	.0057	.0085	.0078	.0057	.0016	.0025	.0007	.0000	.0002	.0377
147-168	I	.0007	.0015	.0027	.0047	.0040	.0014	.0002	.0002	.0003	.0001	.0000	.0159
	N	.0002	.0008	.0011	.0039	.0064	.0037	.0022	.0035	.0003	.0000	.0002	.0224
	L	.0003	.0003	.0022	.0050	.0098	.0081	.0080	.0110	.0001	.0000	.0001	.0450
	M	.0001	.0005	.0011	.0043	.0057	.0027	.0015	.0003	.0000	.0000	.0000	.0163
	T	.0014	.0031	.0072	.0179	.0260	.0159	.0119	.0151	.0008	.0001	.0003	.0997
169-191	I	.0003	.0013	.0021	.0058	.0047	.0014	.0010	.0002	.0002	.0003	.0000	.0174
	N	.0001	.0005	.0013	.0043	.0049	.0026	.0014	.0005	.0000	.0000	.0001	.0157
	L	.0002	.0010	.0022	.0106	.0101	.0046	.0046	.0027	.0000	.0000	.0003	.0364
	M	.0002	.0005	.0014	.0037	.0065	.0034	.0009	.0007	.0000	.0000	.0000	.0173
	T	.0009	.0032	.0069	.0245	.0262	.0120	.0079	.0041	.0002	.0003	.0005	.0867
192-213	I	.0002	.0021	.0032	.0071	.0040	.0022	.0011	.0002	.0000	.0000	.0002	.0203
	N	.0003	.0006	.0016	.0025	.0013	.0013	.0013	.0008	.0000	.0000	.0000	.0096
	L	.0001	.0002	.0022	.0065	.0039	.0023	.0019	.0008	.0000	.0000	.0001	.0181
	M	.0001	.0003	.0007	.0041	.0032	.0021	.0003	.0002	.0000	.0000	.0000	.0111
	T	.0008	.0032	.0077	.0202	0123	.0078	.0047	.0021	.0000	.0000	.0003	.0591

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Table 4 (continued)

Wind Direction	Temp. Grad.	Wind Speed (m/s) 0.0-0.5	>0.5-1.0	>1.0-1.5	>1.5-2.5	>2.5-3.5	>3.5-4.5	>4.5-5.5	>5.5-8.5	>8.5-11.0	>11.0	Miss	Total
214-236	I	.0011	.0022	.0027	.0062	.0032	.0009	.0008	.0005	.0001	.0000	.0002	.0179
	N	.0002	.0010	.0008	.0014	.0009	.0007	.0003	.0002	.0001	.0001	.0001	.0059
	L	.0000	.0014	.0010	.0042	.0010	.0007	.0010	.0014	.0000	.0000	.0002	.0110
	M	.0001	.0002	.0011	.0031	.0015	.0003	.0002	.0002	.0000	.0000	.0000	.0069
	T	.0015	.0048	.0057	.0149	.0066	.0026	.0024	.0023	.0002	.0001	.0006	.0417
237-258	I	.0007	.0018	.0021	.0021	.0017	.0008	.0013	.0006	.0006	.0000	.0002	.0118
	N	.0000	.0003	.0005	.0010	.0010	.0003	.0003	.0008	.0002	.0001	.0000	.0047
	L	.0003	.0011	.0010	.0014	.0006	.0007	.0007	.0009	.0000	.0000	.0001	.0069
	M	.0000	.0005	.0005	.0007	.0006	.0002	.0001	.0007	.0000	.0000	.0000	.0032
	T	.0010	.0038	.0040	.0051	.0039	.0021	.0024	.0030	.0008	.0001	.0003	.0265
259-281	I	.0009	.0017	.0013	.0008	.0015	.0014	.0006	.0008	.0007	.0003	.0001	.0101
	N	.0001	.0009	.0005	.0008	.0009	.0016	.0009	.0021	.0010	.0002	.0000	.0090
	L	.0001	.0003	.0007	.0017	.0016	.0023	.0018	.0027	.0006	.0001	.0007	.0127
	M	.0000	.0005	.0005	.0006	.0007	.0005	.0008	.0007	.0001	.0000	.0000	.0042
	T	.0011	.0034	.0029	.0039	.0047	.0057	.0041	.0063	.0024	.0007	.0008	.0360
282-303	I	.0005	.0013	.0011	.0003	.0005	.0009	.0014	.0022	.0016	.0010	.0000	.0107
	N	.0001	.0003	.0006	.0003	.0010	.0019	.0033	.0114	.0070	.0011	.0000	.0272
	L	.0006	.0006	.0002	.0011	.0014	.0029	.0015	.0101	.0042	.0017	.0011	.0254
	M	.0000	.0003	.0001	.0005	.0010	.0006	.0009	.0023	.0002	.0003	.0000	.0063
	T	.0011	.0025	.0021	.0023	.0039	.0063	.0071	.0260	.0130	.0042	.0011	.0696
304-326	I	.0005	.0014	.0009	.0006	.0009	.0008	.0007	.0018	.0006	.0002	.0000	.0083
	N	.0003	.0006	.0001	.0010	.0022	.0026	.0047	.0162	.0048	.0016	.0001	.0343
	L	.0001	.0002	.0006	.0010	.0014	.0030	.0033	.0088	.0064	.0025	.0005	.0278
	M	.0000	.0005	.0002	.0010	.0006	.0007	.0026	.0056	.0015	.0003	.0000	.0130
	T	.0009	.0026	.0018	.0037	.0050	.0071	.0113	.0325	.0133	.0047	.0006	.0835

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Table 4 (continued)

Wind Speed (m/s)

Wind Direction	Temp. Grad.	0.0-0.5	>0.5-1.0	>1.0-1.5	>1.5-2.5	>2.5-3.5	>3.5-4.5	>4.5-5.5	>5.5-8.5	>8.5-11.0	>11.0	Miss	Total
327-348	I	.0009	.0015	.0015	.0016	.0011	.0009	.0006	.0010	.0000	.0000	.0001	.0093
	N	.0001	.0005	.0008	.0034	.0033	.0034	.0049	.0083	.0015	.0000	.0002	.0265
	L	.0002	.0005	.0005	.0026	.0043	.0027	.0031	.0077	.0016	.0002	.0000	.0234
	M	.0000	.0005	.0008	.0008	.0011	.0022	.0027	.0053	.0009	.0001	.0000	.0144
	T	.0013	.0029	.0035	.0085	.0099	.0093	.0113	.0223	.0040	.0003	.0003	.0736
Calm	I	.0000										.0000	.0000
	N	.0000										.0000	.0000
	L	.0000										.0000	.0000
	M	.0000										.0000	.0000
	T	.0000										.0000	.0000
Miss	I	.0001	.0003	.0007	.0005	.0003	.0002	.0001	.0003	.0000	.0000	.0017	.0043
	N	.0000	.0000	.0000	.0000	.0000	.0001	.0005	.0010	.0009	.0000	.0010	.0035
	L	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0002	.0001	.0024	.0027
	M	.0000	.0000	.0001	.0000	.0006	.0002	.0002	.0000	.0000	.0000		.0011
	T	.0001	.0003	.0008	.0005	.0009	.0006	.0008	.0014	.0011	.0001	.0051	.0118
Total	I	.0104	.0285	.0393	.0620	.0439	.0197	.0098	.0088	.0042	.0021	.0038	.2324
	N	.0043	.0122	.0199	.0458	.0536	.0401	.0302	.0529	.0181	.0034	.0074	.2881
	L	.0027	.0099	.0182	.0573	.0586	.0425	.0338	.0570	.0139	.0047	.0063	.3051
	M	.0021	.0085	.0131	.0397	.0439	.0264	.0178	.0191	.0030	.0009	.0000	.1743
	T	.0195	.0591	.0905	.2048	.2001	.1287	.0917	.1379	.0392	.0110	.0175	1.0000

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Table 5.Indian Point 3 Frequency Distribution of Wind Speed and Direction at 100 ft Tower Level
According to Temperature Gradient Class-January 1, 1971-December 31, 1971

Wind Direction	Temp. Grad.	Wind Speed (m/s) 0.0-0.5	>0.5-1.0	>1.0-1.5	>1.5-2.5	>2.5-3.5	>3.5-4.5	>4.5-5.5	>5.5-8.5	>8.5-11.0	>11.0	Miss	Total
349- 11	I	.0007	.0023	.0041	.0071	.0063	.0041	.0013	.0019	.0003	.0007	.0000	.0288
	N	.0001	.0005	.0009	.0041	.0065	.0050	.0018	.0046	.0008	.0002	.0002	.0248
	L	.0000	.0005	.0010	.0050	.0067	.0047	.0033	.0037	.0011	.0000	.0001	.0262
	M	.0002	.0021	.0031	.0078	.0121	.0074	.0067	.0031	.0000	.0006	.0008	.0439

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Table 5 (continued)

Wind Direction	Temp. Grad.	0.0-0.5	>0.5-1.0	>1.0-1.5	>1.5-2.5	>2.5-3.5	>3.5-4.5	>4.5-5.5	>5.5-8.5	>8.5-11.0	>11.0	Miss	Total
	T	.0010	.0053	.0091	.0240	.0317	.0213	.0132	.0133	.0023	.0015	.0011	.1237
12- 33	I	.0005	.0033	.0039	.0150	.0169	.0053	.0016	.0018	.0000	.0000	.0000	.0483
	N	.0001	.0011	.0010	.0037	.0070	.0047	.0032	.0016	.0001	.0000	.0000	.0225
	L	.0001	.0002	.0007	.0045	.0043	.0021	.0003	.0007	.0000	.0000	.0001	.0130
	M	.0002	.0023	.0041	.0136	.0145	.0096	.0040	.0022	.0006	.0000	.0002	.0513
	T	.0009	.0070	.0097	.0367	.0428	.0216	.0091	.0063	.0007	.0000	.0003	.1352
34- 56	I	.0002	.0030	.0051	.0070	.0045	.0008	.0001	.0000	.0000	.0000	.0000	.0207
	N	.0001	.0010	.0017	.0025	.0023	.0010	.0001	.0006	.0001	.0000	.0010	.0105
	L	.0001	.0005	.0013	.0022	.0009	.0008	.0001	.0000	.0000	.0000	.0001	.0059
	M	.0002	.0018	.0048	.0096	.0080	.0024	.0009	.0003	.0002	.0000	.0000	.0284
	T	.0007	.0063	.0129	.0213	.0157	.0050	.0013	.0009	.0003	.0000	.0011	.0655
57- 78	I	.0001	.0017	.0027	.0023	.0005	.0003	.0001	.0000	.0000	.0000	.0000	.0078
	N	.0001	.0005	.0006	.0010	.0007	.0016	.0002	.0001	.0000	.0000	.0002	.0050
	L	.0002	.0003	.0001	.0009	.0001	.0002	.0006	.0002	.0000	.0000	.0000	.0027
	M	.0002	.0017	.0022	.0035	.0032	.0006	.0000	.0000	.0000	.0000	.0000	.0114
	T	.0007	.0042	.0056	.0078	.0045	.0027	.0009	.0003	.0000	.0000	.0002	.0270
79-101	I	.0002	.0009	.0017	.0016	.0002	.0000	.0000	.0000	.0000	.0000	.0000	.0047
	N	.0001	.0005	.0009	.0013	.0008	.0005	.0001	.0000	.0000	.0000	.0000	.0041
	L	.0000	.0003	.0002	.0000	.0005	.0007	.0001	.0000	.0000	.0000	.0000	.0018
	M	.0003	.0021	.0016	.0024	.0013	.0003	.0000	.0000	.0000	.0000	.0000	.0080
	T	.0007	.0038	.0045	.0053	.0027	.0015	.0002	.0000	.0000	.0000	.0000	.0186
102-123	I	.0002	.0015	.0016	.0024	.0008	.0001	.0000	.0000	.0000	.0000	.0000	.0066
	N	.0000	.0008	.0002	.0010	.0011	.0001	.0001	.0000	.0000	.0000	.0000	.0034
	L	.0001	.0003	.0003	.0010	.0003	.0003	.0003	.0002	.0000	.0000	.0000	.0031
	M	.0000	.0011	.0021	.0019	.0009	.0001	.0000	.0001	.0000	.0000.	.0000	.0063
	T	.0003	.0038	.0042	.0064	.0032	.0007	.0005	.0003	.0000	.0000	.0000	.0194

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124-146	I	.0003	.0021	.0015	.0046	.0030	.0007	.0000	.0001	.0000	.0000	.0000	.0122
	N	.0000	.0008	.0005	.0015	.0030	.0011	.0008	.0008	.0000	.0000	.0000	.0085
	L	.0000	.0002	.0005	.0011	.0009	.0021	.0015	.0022	.0001	.0000	.0000	.0086
	M	.0001	.0011	.0007	.0029	.0023	.0017	.0014	.0015	.0002	.0002	.0000	.0121
	T	.0005	.0042	.0031	.0101	.0091	.0056	.0037	.0046	.0003	.0002	.0000	.0414
147-168	I	.0005	.0010	.0026	.0051	.0047	.0025	.0007	.0005	.0000	.0000	.0000	.0176
	N	.0000	.0006	.0005	.0032	.0018	.0019	.0022	.0017	.0000	.0000	.0000	.0119
	L	.0000	.0003	.0011	.0047	.0053	.0051	.0048	.0024	.0003	.0000	.0000	.0241
	M	.0000	.0015	.0021	.0071	.0088	.0046	.0022	.0027	.0000	.0002	.0000	.0292
	T	.0005	.0034	.0063	.0201	.0206	.0142	.0098	.0073	.0003	.0002	.0000	.0828
169-191	I	.0002	.0016	.0038	.0070	.0055	.0017	.0001	.0006	.0001	.0000	.0000	.0206
	N	.0000	.0003	.0005	.0018	.0017	.0014	.0006	.0011	.0000	.0000	.0000	.0074
	L	.0000	.0005	.0015	.0055	.0045	.0023	.0019	.0013	.0000	.0000	.0000	.0174
	M	.0001	.0019	.0027	.0090	.0079	.0034	.0030	.0022	.0000	.0000	.0000	.0303
	T	.0003	.0043	.0085	.0233	.0196	.0088	.0056	.0051	.0001	.0000	.0000	.0757
192-213	I	.0005	.0017	.0034	.0051	.0042	.0021	.0007	.0003	.0002	.0000	.0000	.0183
	N	.0001	.0002	.0005	.0010	.0007	.0006	.0006	.0006	.0001	.0000	.0000	.0043
	L	.0000	.0003	.0008	.0032	.0017	.0009	.0014	.0009	.0000	.0000	.0000	.0093
	M	.0000	.0015	.0031	.0061	.0061	.0039	.0013	.0006	.0000	.0000	.0002	.0226
	T	.0006	.0038	.0078	.0154	.0127	.0074	.0039	.0024	.0003	.0000	.0002	.0545

Table 5 (continued)

		Wind Speed (m/s)											
Wind Direction	Temp. Grad.	0.0-0.5	>0.5-1.0	>1.0-1.5	>1.5-2.5	>2.5-3.5	>3.5-4.5	>4.5-5.5	>5.5-8.5	>8.5-11.0	>11.0	Miss	Total
214-236	I	.0001	.0014	.0023	.0033	.0024	.0010	.0003	.0011	.0001	.0000	.0000	.0121
	N	.0000	.0006	.0005	.0007	.0002	.0009	.0005	.0017	.0002	.0001	.0000	.0054
	L	.0001	.0006	.0010	.0018	.0006	.0007	.0002	.0014	.0001	.0000	.0000	.0065
	M	.0002	.0013	.0025	.0046	.0025	.0010	.0008	.0007	.0000	.0000	.0003	.0140
	T	.0005	.0038	.0063	.0104	.0057	.0037	.0018	.0049	.0005	.0001	.0003	.0380
237-258	I	.0005	.0021	.0019	.0024	.0018	.0021	.0007	.0013	.0007	.0002	.0000	.0136

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Wind Direction	Temp. Grad.	0.0-0.5	>0.5-1.0	>1.0-1.5	>1.5-2.5	>2.5-3.5	>3.5-4.5	>4.5-5.5	>5.5-8.5	>8.5-11.0	>11.0	Miss	Total
	N	.0001	.0003	.0005	.0002	.0009	.0014	.0013	.0016	.0006	.0001	.0000	.0070
	L	.0000	.0002	.0003	.0018	.0007	.0007	.0011	.0035	.0014	.0006	.0000	.0104
	M	.0002	.0014	.0013	.0015	.0021	.0006	.0006	.0006	.0000	.0000	.0001	.0082
	T	.0008	.0040	.0040	.0059	.0055	.0047	.0037	.0070	.0026	.0009	.0001	.0392
259-281	I	.0003	.0016	.0013	.0017	.0016	.0019	.0016	.0013	.0007	.0001	.0000	.0121
	N	.0001	.0002	.0000	.0000	.0009	.0013	.0024	.0053	.0013	.0001	.0000	.0115
	L	.0000	.0003	.0003	.0005	.0015	.0027	.0021	.0038	.0023	.0003	.0000	.0138
	M	.0005	.0013	.0014	.0016	.0022	.0023	.0016	.0030	.0014	.0000	.0001	.0152
	T	.0009	.0034	.0030	.0038	.0062	.0082	.0077	.0133	.0056	.0006	.0001	.0527
282-303	I	.0008	.0014	.0009	.0010	.0014	.0017	.0024	.0029	.0008	.0001	.0001	.0135
	N	.0001	.0002	.0003	.0002	.0008	.0023	.0030	.0069	.0024	.0013	.0013	.0188
	L	.0001	.0001	.0009	.0009	.0008	.0023	.0035	.0090	.0041	.0010	.0000	.0229
	M	.0006	.0008	.0014	.0014	.0024	.0025	.0030	.0080	.0024	.0009	.0001	.0234
	T	.0016	.0025	.0035	.0035	.0054	.0088	.0119	.0268	.0097	.0033	.0015	.0786
304-326	I	.0001	.0017	.0005	.0010	.0013	.0019	.0023	.0029	.0003	.0001	.0000	.0121
	N	.0000	.0002	.0002	.0003	.0003	.0015	.0032	.0072	.0011	.0003	.0002	.0148
	L	.0001	.0010	.0006	.0007	.0018	.0023	.0015	.0079	.0041	.0005	.0000	.0205
	M	.0005	.0008	.0010	.0006	.0025	.0030	.0040	.0113	.0056	.0008	.0000	.0301
	T	.0007	.0038	.0023	.0026	.0059	.0087	.0110	.0293	.0112	.0017	.0002	.0774

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Table 5 (continued)

Wind Direction	Temp. Grad.	Wind Speed (m/s) 0.0-0.5	>0.5-1.0	>1.0-1.5	>1.5-2.5	>2.5-3.5	>3.5-4.5	>4.5-5.5	>5.5-8.5	>8.5-11.0	>11.0	Miss	Total
327-348	I	.0006	.0021	.0011	.0017	.0018	.0010	.0014	.0015	.0002	.0000	.0000	.0114
	N	.0001	.0002	.0002	.0003	.0017	.0017	.0023	.0040	.0003	.0001	.0001	.0112
	L	.0000	.0000	.0003	.0017	.0035	.0040	.0019	.0053	.0005	.0002	.0000	.0175
	M	.0000	.0010	.0014	.0025	.0031	.0042	.0039	.0095	.0032	.0003	.0006	.0297
	T	.0007	.0033	.0031	.0063	.0102	.0110	.0095	.0202	.0042	.0007	.0007	.0699
Calm	I	.0000										.0000	.0000
	N	.0000										.0000	.0000
	L	.0000										.0000	.0000

	M	.0000										.0000	.0000
	T	.0000										.0000	.0000
Miss	I	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
	N	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0001	.0001
	L	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0002	.0002
	M	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
	T	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0003	.0003
Total	I	.0058	.0293	.0385	.0684	.0568	.0273	.0133	.0161	.0035	.0013	.0001	.2605
	N	.0011	.0081	.0089	.0230	.0305	.0270	.0223	.0377	.0071	.0023	.0032	.1713
	L	.0009	.0058	.0111	.0356	.0342	.0319	.0248	.0424	.0141	.0026	.0006	.2040
	M	.0034	.0237	.0353	.0760	.0798	.0477	.0333	.0457	.0136	.0031	.0025	.3642
	T	.0113	.0669	.0939	.2030	.2014	.1339	.0937	.1420	.0383	.0093	.0064	1.0000

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Table 6. Trap Rock Frequency Distribution of Wind Speed and Direction
January 1, 1970-December 31, 1970.

Wind Direction		Wind Speed (m/s) 0.0-0.5	>0.5-1.0	>1.0-1.5	>1.5-2.5	>2.5-3.5	>3.5-4.5	>4.5-5.5	>5.5-8.5	>8.5-11.0	>11.0	Miss	Total
349- 11	N	.0049	.0113	.0146	.0247	.0203	.0155	.0117	.0170	.0030	.0013	.0029	.1271
12- 33	NNE	.0024	.0101	.0137	.0244	.0215	.0087	.0051	.0034	.0007	.0002	.0022	.0925
34- 56	NE	.0035	.0088	.0105	.0118	.0086	.0024	.0005	.0009	.0005	.0000	.0007	.0481
57- 78	ENE	.0021	.0047	.0058	.0052	.0015	.0003	.0001	.0008	.0000	.0000	.0002	.0208
79-101	E	.0012	.0028	.0036	.0056	.0028	.0012	.0003	.0003	.0000	.0000	.0001	.0178
102-123	ESE	.0020	.0038	.0039	.0058	.0046	.0022	.0009	.0007	.0001	.0000	.0000	.0241
124-146	SE	.0012	.0044	.0058	.0146	.0152	.0130	.0081	.0133	.0027	.0002	.0002	.0786
147-168	SSE	.0019	.0057	.0109	.0210	.0208	.0126	.0075	.0111	.0010	.0007	.0003	.0935
169-191	S	.0012	.0072	.0084	.0178	.0141	.0084	.0050	.0041	.0001	.0000	.0008	.0671
192-213	SSW	.0021	.0057	.0088	.0130	.0091	.0036	.0027	.0023	.0002	.0001	.0007	.0483
214-236	SW	.0029	.0064	.0065	.0075	.0031	.0024	.0013	.0029	.0001	.0005	.0002	.0338
237-258	WSW	.0028	.0041	.0025	.0045	.0038	.0029	.0028	.0024	.0005	.0005	.0000	.0267

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259-281	W	.0023	.0038	.0027	.0034	.0043	.0051	.0042	.0083	.0030	.0012	.0001	.0383
282-303	WNW	.0022	.0038	.0031	.0065	.0073	.0089	.0079	.0204	.0091	.0056	.0010	.0758
304-326	NW	.0023	.0047	.0049	.0081	.0109	.0131	.0087	.0306	.0191	.0076	.0030	.1130
327-348	NNW	.0038	.0087	.0079	.0145	.0139	.0114	.0076	.0139	.0031	.0030	.0020	.0898
Calm		.0000											.0000
Miss		.0000	.0003	.0005	.0023	.0012	.0001	.0000	.0001	.0001	.0000		.0046
Total		.0385	.0963	.1140	.1905	.1630	.1117	.0743	.1326	.0434	.0208	.0146	1.0000

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Table 7. Trap Rock Frequency Distribution of Wind Speed and Direction January 1, 1971-December 31, 1971.

Wind Direction		Wind Speed (m/s) 0.0-0.5	>0.5-1.0	>1.0-1.5	>1.5-2.5	>2.5-3.5	>3.5-4.5	>4.5-5.5	>5.5-8.5	>8.5-11.0	>11.0	Miss	Total
349- 11	N	.0011	.0105	.0123	.0240	.0217	.0166	.0126	.0266	.0041	.0030	.0002	.1328
12- 33	NNE	.0015	.0061	.0159	.0375	.0281	.0145	.0046	.0037	.0002	.0001	.0008	.1130
34- 56	NE	.0012	.0070	.0078	.0231	.0120	.0044	.0013	.0013	.0001	.0000	.0007	.0590
57- 78	ENE	.0007	.0033	.0055	.0069	.0032	.0012	.0004	.0001	.0001	.0000	.0001	.0214
79-101	E	.0008	.0033	.0033	.0040	.0025	.0009	.0004	.0000	.0000	.0000	.0001	.0152
102-123	ESE	.0011	.0027	.0027	.0071	.0046	.0013	.0009	.0002	.0000	.0000	.0001	.0207
124-146	SE	.0006	.0019	.0036	.0091	.0139	.0099	.0079	.0084	.0007	.0001	.0002	.0565
147-168	SSE	.0007	.0040	.0084	.0179	.0188	.0122	.0086	.0081	.0009	.0001	.0000	.0797
169-191	S	.0015	.0033	.0078	.0161	.0124	.0067	.0041	.0043	.0002	.0001	.0001	.0567
192-213	SSW	.0012	.0047	.0079	.0145	.0099	.0061	.0033	.0027	.0000	.0000	.0008	.0511
214-236	SW	.0012	.0047	.0068	.0120	.0058	.0029	.0025	.0037	.0000	.0002	.0008	.0407
237-258	WSW	.0018	.0053	.0044	.0040	.0047	.0033	.0037	.0071	.0027	.0009	.0008	.0387
259-281	W	.0013	.0040	.0029	.0044	.0053	.0047	.0057	.0106	.0042	.0009	.0016	.0457
282-303	WNW	.0013	.0044	.0029	.0042	.0088	.0111	.0118	.0261	.0085	.0034	.0027	.0852
304-326	NW	.0013	.0057	.0037	.0047	.0071	.0129	.0117	.0360	.0161	.0037	.0022	.1052
327-348	NNW	.0014	.0083	.0077	.0090	.0094	.0113	.0092	.0139	.0026	.0007	.0013	.0748
Calm		.0000											.0000
Miss		.0000	.0000	.0005	.0012	.0005	.0005	.0004	.0005	.0001	.0002		.0037

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Total	.0185	.0790	.1043	.1997	.1685	.1204	.0891	.1535	.0407	.0137	.0127	1.0000

Table 8. Montrose Frequency Distribution of Wind Speed and Direction
January 1, 1970-December 31, 1970.

Wind Direction		Wind Speed (m/s) 0.0-0.5	>0.5-1.0	>1.0-1.5	>1.5-2.5	>2.5-3.5	>3.5-4.5	>4.5-5.5	>5.5-8.5	>8.5-11.0	>11.0	Miss	Total
349-11	N	.0038	.0072	.0054	.0084	.0143	.0099	.0050	.0067	.0010	.0003	.0123	.0743
12-33	NNE	.0033	.0064	.0125	.0294	.0200	.0102	.0077	.0078	.0011	.0000	.0241	.1224
34-56	NE	.0024	.0065	.0092	.0203	.0146	.0054	.0016	.0010	.0001	.0000	.0223	.0834
57-78	ENE	.0016	.0037	.0052	.0038	.0033	.0009	.0001	.0013	.0001	.0000	.0078	.0278
79-101	E	.0010	.0027	.0014	.0044	.0009	.0004	.0009	.0007	.0004	.0000	.0065	.0193
102-123	ESE	.0016	.0028	.0023	.0043	.0021	.0007	.0001	.0006	.0000	.0000	.0047	.0191
124-146	SE	.0013	.0035	.0060	.0102	.0094	.0054	.0020	.0010	.0000	.0000	.0098	.0485
147-168	SSE	.0007	.0026	.0082	.0271	.0240	.0184	.0081	.0070	.0001	.0001	.0160	.1123
169-191	S	.0009	.0030	.0075	.0153	.0133	.0079	.0026	.0011	.0003	.0006	.0140	.0665
192-213	SSW	.0017	.0037	.0051	.0108	.0092	.0058	.0026	.0016	.0001	.0001	.0148	.0555
214-236	SW	.0023	.0045	.0072	.0082	.0044	.0033	.0020	.0026	.0006	.0003	.0099	.0452
237-258	WSW	.0021	.0037	.0045	.0052	.0024	.0013	.0016	.0023	.0007	.0003	.0071	.0312
259-281	W	.0023	.0028	.0030	.0034	.0035	.0038	.0031	.0058	.0011	.0011	.0070	.0370
282-303	WNW	.0017	.0026	.0021	.0030	.0047	.0044	.0051	.0148	.0079	.0028	.0068	.0559
304-326	NW	.0018	.0024	.0020	.0045	.0058	.0067	.0084	.0262	.0148	.0050	.0135	.0911
327-348	NNW	.0040	.0034	.0031	.0045	.0071	.0065	.0081	.0200	.0047	.0007	.0119	.0740
Calm		.0000											.0000
Miss		.0054	.0060	.0070	.0085	.0058	.0021	.0010	.0003	.0003	.0000		.0363
Total		.0377	.0675	.0918	.1712	.1448	.0932	.0597	.1006	.0335	.0113	.1885	1.000

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2.2 Recorders

All instruments produce ink trace records on strip chart analog recorders. Items (a) and (b) use Esterline Angus continuous trace recorders at chart speed of 3 in./hr. Items (c) and (d) use a Bendix-Friez continuous trace recorder at chart speed of 3 in./hr. Item (e) uses a continuous trace recorder at chart speed of 1 in./hr. Item (f) uses a Texas Instrument dual channel continuous trace recorder for elevation and azimuth readings at chart speeds of either 3 in./hr. or 3 in./min. Item (g) uses a 6 channel Honeywell-Brown dot print recorder with a three minute recording cycle. Items (h), (i) and (j) use a common 8 point Bristol recorder which dot prints every 30 seconds at a chart speed of 3 in./hr.

3. Data Log

A monthly tabulation of days in which 12 or more hours of missing data occurred during the period of data collection

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ts. Complete breakdowns due to lightning strikes on the Montrose Point and Trap Rock towers were another source of data loss.

A special reference must be made to the Honeywell-Brown temperature difference system. One of the initial purposes for data collection at Indian Point was to determine initial data from January 1970-December 1971 is presented in Table 2. Data collected beyond this period is indicated as well as instrumentation activation and termination dates.

Data recovery information is given in Table 3. Most data loss occurred when instruments were out of service for repair or when data were deemed erroneous due to improper functioning of instruments. Major instrumentation difficulties included: for the Climet wind system, speed head bearing failures and hypersensitivity of the directional module; and for the Honeywell-Brown Temperature Difference system, aspirator failures due to corroded connections and loss of calibration due to unobserved deterioration of recorder componen f any climatic changes had occurred since 1955-57 that could be detrimental to the Indian Point accident model. It was felt that the utilization of the exact instruments used in 1955-57 would facilitate the evaluation. The original Honeywell-Brown system was rehabilitated and placed in operation. Because of age factors it became difficult to maintain temperature difference data recovery at greater than 70-80% after July 1970. The amount of valid data output, however, was considered sufficient to meet its original function. As it became apparent that a reliable, continuously operating temperature difference system was a necessity, and with the complete breakdown of the aging system on 25 October 1971, a new system was acquired and subsequently installed on 14 August 1972.

4. Analytical Procedure

4.1 Wind Direction

To facilitate the development of the statistical analysis, the parameter categories were standardized on the basis of sixteen compass point (22 1/2° sectors) as opposed to eighteen 20° sectors from 002 1/2° as used by Davidson. IP1 No recovery information indicated.

Table 3. Record of Data Recovery

Station	Parameter	Period	Hours of Valid Data	Total Hours in Period	% Recovery
IP3	Wind Direction (Climet)	Annual 1970	8644	8760	98.68
		Summer 1970	5072	5136	98.75
		Winter 1970	3572	3624	98.57
		Annual 1971	8742	8760	99.80
		Summer 1971	4406	4416	99.77
		Winter 1971	4336	4344	99.82
	Wind Speed (Climet)	Annual 1970	8594	8760	98.11
		Summer 1970	5019	5136	97.72
		Winter 1970	3575	3624	98.65
		Annual 1971	8689	8760	99.19
		Summer 1971	4381	4416	99.21
		Winter 1971	4308	4344	99.17
	Wind Direction (Aerovane)	Summer 1970 (Partial: until Sept. 18)	4003	4104	97.54
	Wind Speed (Aerovane)	Summer 1970 (Partial: until Sept.	4001	4104	97.49

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	Temperature Difference	Annual 1970	7222	8760	82.44
		Annual 1971	5560	8760	63.47
TR	Wind Direction	Annual 1970	8600	8760	98.17
		Summer 1970	4983	5136	97.02
		Winter 1970	3617	3624	99.81
		Annual 1971	8524	8760	97.31
		Summer 1971	4242	4416	96.06
		Winter 1971	4282	4344	98.57
	Wind Speed	Annual 1970	8514	8760	97.19
		Summer 1970	5018	5136	97.70
		Winter 1970	3496	3624	96.47
		Annual 1971	8447	8760	96.43
		Summer 1971	4243	4416	96.08
		Winter 1971	4204	4344	96.78

Table 3 (continued)

Station	Parameter	Hours of Valid Period	Total Hours in Data	% Period	Recovery
MP	Wind Direction	Annual 1970	6794	8760	77.56
		Summer	4045	5136	78.76
	Wind Speed	Annual 1970	5721	8760	65.31
		Summer	3140	5136	61.14
CC	Wind Direction	Summer 1970	4093	4104	99.73
	Wind Speed	Summer 1970	4091	4104	99.68

The frequency density for each wind quadrant is determined by:

$$f = \frac{n}{(N-m)^{\theta}}$$

where:

f = frequency density

n = number of data hours in a specified category

N = total number of hours possible for a selected time period

m = number of hours for which <u>no</u> data was available

θ = sector interval (Note: 20.0° for 1955-1957 data and 22.5° for 1970-1971 data.)

The sum of the sector frequency densities is equal to 1. Calm and missing data are assumed to be equally distributed throughout the compass and are divided by a sector width of 360°.

4.2 <u>Wind Speed</u>

Wind speeds are presented in m/s. (For conversion to mph, multiply by 2.237) The number of speed categories were increased at low end to provide more detailing of the speed distribution in this critical range.

For wind speeds, the frequency is defined by

$$F_x = \frac{n_x}{(N-m)}$$

where:

F_x = frequency in a class interval

n_x = number of data hours within a specified interval

N = total number of hours possible for a selected time period

m = number of hours for which no data was available

The percent probability is defined by

$$P_x = \left[\frac{n_x}{(N-m)+1}\right] x100$$

The cumulative percent probability is defined by

$$CP = \sum_{i=1}^{i=x} P_i$$

where:

i = number of class intervals.

When CP is plotted, a point on the curve is read as the percent time that the wind speed # indicated value.

4.3 Stability Categories

4.31 Atmospheric Stability Categories Defined in Terms of

Temperature Difference

Class	IP1 – 1955-57 (300 ft tower)	IP3 – 1970-71 (100 ft tower)
I = Inversion ("stable")	$T_{150} - T_7 \geq 0$	$T_{95} - T_7 \geq 0$
N = Isothermal-Adiabatic ("neutral")	$0 > T_{150} - T_7 \geq -0.98$	$0 > T_{95} - T_7 \geq -0.58F$
L = Lapse ("unstable")	$-0.98F > T_{150} - T_7$	$-0.58F > T_{95} - T_7$

4.32 Classification of Atmospheric Stability According to U. S. A. E. C. Safety Guide 23

Stablity Classification	Pasquill Categories	Temperature Change with Height (8C/100m)	(°F/88 ft) (IP3)
Extremely Unstable	A	< -1.9	< -0.9
Moderately Unstable	B	-1.9 to –1.7	-0.9 to < -0.8
Slightly Unstable	C	-1.7 to –1.5	-0.8 to < -0.7
Neutral	D	-1.5 to –0.5	-0.7 to < -0.2
Slightly Stable	E	-0.5 to 1.5	-0.2 to < +0.7
Moderately Stable	F	1.5 to 4.0	+0.7 to <1.9
Extremely Stable	G	>4.0	$1.9

4.4 Seasonal Distributions

After careful examination of individual monthly frequency distributions it was determined that a definite seasonal effect existed. Two distinct seasons emerge: summer and winter, with no distinctive transition months. November-March are always "winter" months and May-September always "summer" months. April and October distribution patterns may appropriately fit either a winter or summer season. There was no means of predicting, in advance, what season was appropriate for these two months until the actual data was analyzed. The seasonal breakdown, as used, is as follows:

Winter Summer

1970	January-March, November-December	April-October
	(5 months)	(7 months)
1971	January-April, November-December	May-October
	(6 months)	(6 months)

4.5 Diurnal Analysis

The hourly diurnal mean wind direction and speed were determined by resolving each individual data point into its N-S and E-W components, summing the components within each hourly category and calculating a mean wind direction and speed. When plotting the diurnal direction curve it is assumed that the rotation of the mean between two points will circumvent the smallest arc.

Persistence is defined as,

$$P = \frac{\text{Vector Average Speed}\left(\left|\overline{\overline{V}}\right|\right)}{\text{Magnitude of Scalar Average Speed}\left(\left|\overline{V}\right|\right)}$$

4.6 Spatial Variation

The spatial relationship of wind direction and speed between two tower sites was determined by selecting the category range at one station as the independent variable and calculating a frequency distribution and mean value at the second station for each of the independent variable's categories. This method assumes a gaussian distribution around the mean value at the independent station.

5. Discussion of Data

5.1 Joint Frequency Distribution of Wind Direction, Wind Speed and Temperature Difference

5.1.1 Inversion – Neutral – Lapse Temperature Difference Classification

Annual frequency distributions of wind speed and direction according to temperature gradient class at Indian Point 3 for 1970 and 1971 are given in Tables 4 and 5, respectively.

5.1.2 Pasquill Stability Classification

Annual joint frequency distributions of wind direction and speed according to the Pasquill stability categories, as established by the U. S. A. E. C. Safety Guide 23, at Indian Point 3 for 1970 and 1971 are given in Appendix Table 1 and 2, respectively.

5.2 Joint Frequency Distribution of Wind Direction and Temperature Difference

5.2.1 Monthly Distribution of Temperature Difference

A monthly distribution of temperature difference at IP3 for 1970 and 1971 is given in Figure 2. The distribution for missing hours of temperature difference data is also presented so that a measure of reliability can be determined.

The frequency of inversions remain relatively constant throughout the year. Neutral conditions are more frequent during the winter months while lapse conditions reach their peak during the summer months. These features seem most apparent during those time periods, in both years, when data collection efficiency was greater than 80%.

5.2.2 Annual Joint Frequency Distribution

The annual frequency distributions of wind direction according to temperature difference category for IP1 (1956) and IP3 (1970 and 1971) are presented in Figure 3.

For inversion conditions, a marked discrepancy is observed between the 1956 and 1970-71 data. Substantial decreases are observed in the frequency of observed inversions during 1970-71 in all quadrants except those between 050 to 160°. The reduction is by a factor of 0.50 in some sectors. The cause of this deviation may be due to one, or more of the following: a) different tower locations; b) different measurement height intervals ((150-7 ft (1956) and 95-7 ft (1970-71)); c) change in the surrounding environs (the area is now more developed); or d) a calibration error ((only 0.9°F (1956) and 0.5°F (1970-71)) separate inversions from lapse conditions).

The distribution for neutral and lapse conditions between 1956 and 1970-71 are generally similar. The neutral maxima are from the same quadrant (NNE) as the inversion for all years. Under lapse conditions, maximums are observed of approximately equal magnitude for winds from the N and SSE quadrants.

Major discrepancies between the 1970 and 1971 data occur in the neutral category from 300° -030° and in the lapse category from 140° -200°. These can be attributed to periods of missing data. The decrease in frequency of neutral condition northerly winds in 1971 may be due to missing data during October-December. A strong northwesterly wind persists during these months resulting in dominating neutral atmospheric conditions. The decrease in frequency of lapse conditions for southerly winds in 1971 may be due to data loss during May-July. These are the months in which lapse conditions normally prevail.

In general, throughout 1970-71 most specific descrepancies are associated with periods of maximum data losses.

5.3 Joint Frequency Distribution of Wind Direction and Wind Speed

The annual joint frequency distribution of wind direction and wind speed for Trap Rock (TR) 1970 and 1971, and Montrose Point (MP) 1970 are given in Tables 6, 7, and 8, respectively.

Wind direction distributions for various speed categories are presented in Figure 4. Discrepancies between 1956 and 1970-71 are less apparent than when classified by temperature difference. The 1970 and 1971 distributions are almost a duplicate of each other.

Halitsky, et al. (1971) found that there was a significant increase in the frequency of occurrence of low speed winds from the critical sector, 000° -040°, for 10 months of 1970 data compared to 1956. With 1970 and 1971 data, the frequencies are nearly the same. The remaining small discrepancy may be due to alteration of quadrant angles. The improvement can be attributed to the extension of 1970 data to a complete year and the difference in category range, 0.0-1.5 m/s in this report versus 0.4-2.0 m/s in the 1971 report. The generally slight increase of frequency in 1970 through most directions is due to a change in instrumentation. A Climet speed unit (threshold: 0.6 MPH) was used in 1970-71 while an aerovane (threshold: 2.5 MPH) was used in 1956. Thus, data points which were measured as calm by the aerovane in 1956 would now be present within the distribution with speed between 0.27-1.2 m/s (0.6-2.5 MPH) when measured with the Climet.

Figure 4 also indicates a general backing into the NW of the maximum frequency direction as wind speeds increase for all years.

5.4 Frequency Distribution of Wind Direction

5.4.1 Annual

Annual frequency distributions of wind direction for the various meteorological tower sites are given in Figure 5. Obvious differences that exist between IP1 (1956) and IP3 (1970-71) are: a) decrease in magnitude of the maximum direction; and b) shifting of the secondary maximum from 172° backing to 156° in 1970-71.

Comparison between IP3 and TR for the same years show differences which are no doubt caused by local topographical effects. The IP3 NNE maximum has backed to the N at Trap Rock. There is also a marked increase in both NW and SE winds at Trap Rock with the NW direction now becoming the secondary maximum.

The frequency density distribution at Montrose Point closely resembles that at IP3.

5.4.2 Seasonal

A definite seasonal pattern was observed during the analysis of monthly data at each station. The seasonal frequency distributions for IP3, Trap Rock and Montrose Point are given in Figures 6, 7 and 8, respectively.

The winter season wind direction frequency distribution at IP3 is tri-nodal. The primary maximum appears with winds from N-NNE. The secondary maximum appears for winds from WNW-NW. This second maximum is of the same order of frequency of the primary. The binodal maximum for winds from these northern quadrants may be attributed to the effect of blockage by the Dunderberg Mountain (Figure 1). The winds, as is seen, preferentially go around it rather than over its top. The tertiary maximum during the winter season is for winds from SSE at half the frequency of the north-quadrant maxima.

During the summer season the direction frequency distributions are binodal with two well defined peaked regions, one northerly (NNE) and the other southerly (SSE). Over the 1970-71 seasons, the maxima are approximately equal in frequency. The winter season secondary peak (WNW-NW) is completely lacking during the summer.

Seasonal differences can be attributed to the influence or lack of influence of the atmospheric geostrophic wind. A persistent and strong NW winter geostrophic wind is responsible for the W-N quadrant peaks at the valley stations. The summer distributions reflect the diurnal wind pattern which flows along the valley axis: upvalley during the day and downvalley at night. The diurnal winds dominate the valley flow system during calm and nearly zero atmospheric pressure gradient conditions, which occur primarily during the summer months and account for the increased frequency of SSE-SSW winds.

Comparison between IP3 (Figure 6) and TR (Figure 7) shows that during the winter TR has the same tri-nodal distribution as IP3. At TR, however, the maximum frequency density is for winds from NNW quadrant without a significant change in frequency of NNE or SSE winds. During the summer the IP3 NNE maximum has backed to N at TR as occurs with the annual distribution and becomes binodal. The previously mentioned sharp increase of annual SE winds at TR is seen to occur almost entirely during the summer season. The annual NW increase is divided between the two seasons.

Based on limited data, the seasonal distributions at Montrose Point (Figure 8) are similar to those of IP3 except for a semblance of retention of the tri-nodal aspect during the summer season.

5.4.3 Comparison Between IP3 and Cape Charles for a Summer Season

Cape Charles as compared to IP3 is further indication of the blocking action of the Dunderberg.

Both stations compare relatively well during northerly quadrant winds. The IP3 sharp peak at NNE is spread out at the Cape Charles almost uniformly over the N, NNE and NE quadrants. This disp On 16 March 1970, an aerovane was installed aboard the U.S.S. Cape Charles. The intent of this station was to obtain comparable data to that collected by Davidson in September-October 1955 aboard the U.S. S. Jones in order to produce diurnal hodographs of the mean vector wind for virtually zero and weak pressure gradient conditions. Data retrieval over a lengthy time span could not be achieved because of the "mothball" fleet's disposal operations. However, a hodograph was constructed for July-September 1970 for comparative purposes (Halitsky, et al. (1971).

Frequency distributions of wind direction for Cape Charles and IP3 aerovanes are given in Figure 9. For the record period, SSE winds are most frequent at IP3 and SW at Cape Charles. This variation is related to the dominant valley terrain features that influence each station (Figure 1). The proximity of the Cape Charles to the Dunderberg Mountain is the controlling feature. The prevailing SSE valley wind, as measured at IP3, is forced to veer and flow parallel to the face of Dunderberg Mountain giving the Cape Charles its SW flow. The lack of NW and NNW winds at the ersion probability reflects the vortex generated as the air is deflected by the eastern tip of Dunderberg. In particular the NE peak could also represent nocturnal drainage flow from the Annsville Creek, Sprout Brook and Peekskill Hollow Brook valley complexes.

It should be cautioned that the Cape Charles aerovane was exposed at an elevation of approximately 100 ft MSL while the IP3 aerovane measured at an elevation of 220 ft MSL and some variations may be due to this difference in exposure elevations.

5.5 Cumulative Probability Distribution for Wind Speed

5.5.1 Annual

The annual cumulative probability curves for wind speed are presented for IP1, IP3, Trap Rock and Montrose Point in Figures 10-13. A summary of annual and seasonal median wind speeds is given in Table 9.

Table 9. Annual and Seasonal Median Wind Speeds

Station			Period	Median Wind (m/s)
Indian Point 1	1956	-	Annual	3.0
Indian Point 3	1970	-	Annual	3.0
			-Winter	3.4
			-Summer	2.9
	1971	-	Annual	3.1
			-Winter	3.7
			-Summer	2.7
Trap Rock	1970	-	Annual	2.8
			-Winter	3.3
			-Summer	2.5
		1971 -	Annual	3.0
			-Winter	3.6
			-Summer	2.7
Montrose Point	1970	-	Annual	2.7
			-Winter	3.3

			-Summer	2.3
Cape Charles	1970	-	Summer	2.8
Indian Point 3	1970	-	Summer	2.3
(Aerovane)				

Figure 10 is a comparison of data collected at IP3 in 1970 and 1971. It shows the relationship between the two years for all wind speed data and as functions of temperature gradient categories. When all wind speed data is considered at each of the stations, there is less than 1% probability deviation between the two annual curves. The maximum deviation is observed at 0.5 m/s where the probability of speeds # 0.5 m/s is about 2.0% and 1.1% for 1970 and 1971, respectively.

Before discussing the curves in terms of temperature gradient categories, it should be borne in mind that the annual curve contains all hours for which wind speed data was available. The speed data recovery was 98.1% in 1970 and 99.2% in 1971 (Table 3). The curves now to be considered were constructed from those data hours in which both wind speed and temperature gradient were available: 82% in 1970 and 63.5% in 1971.

In the total inversion category, the two years are nearly identical in the speed range 1.0-2.5 m/s with 1970 having the greater probability of low wind speeds at 0.5 m/s, and 1971 the greater probability of low wind speeds > 2.5 m/s with a spread of approximately 4%. From 3 m/s to 11 m/s the probability of lower speeds is increased from 16.5% to 23.0% in 1970, and from 17.5% to 26.0% in 1971.

In 1970 the neutral and lapse probabilities curves were quite similar. Below 2.4 m/s there were more neutral occurrences and above 2.4 m/s more lapse observations. The neutral-lapse conditions were, as can be anticipated, more frequent at wind speeds greater than 5.5 m/s and less frequent at speeds below 5.5 m/s. This is consistent with their normal association as being high speed phenomena while inversions are associated with low wind speeds. In 1971, however, the neutral-lapse curves were significantly lower than in 1970 and did not approach the probability of the 1971 annual inversion curve. These discrepancies may be, in part, real but most probably the explanation lies in the fact that there was about 20% less valid data available in 1971 compared to 1970. These excess losses are distributed as follows (from Table 2) in terms of missing days:

	J	F	M	A	M	J	J	A	S	O	N	D
1970	0	3	3	0	0	3	0	13	17	20	4	0

1971	0	12	2	1	16	11	13	2	7	5	30	31
1971-1970	0	9	-1	1	16	8	13	-11	-10	-15	26	31

The losses during the summer may account for the lack of lapse data and the excessive winter losses can account for the decrease in neutral data since these are the seasons when such gradients normally prevail.

A comparison of the probability curves between IP1 (1956) and IP3 (1970 and 1971) on an annual basis, for total inversions and inversions in the critical sector, is shown in Figure 11. For all available wind speeds, IP3 is nearly identical to IP1 until 4 m/s after which IP1 indicates a higher probability of lower wind speeds. For all inversion data IP1 indicates a significantly higher probability of low winds speeds. When the analysis of inversion data is restricted to the critical quadrants (002.5° -22.5° in 1956, and 011.5°-033.5° in 1970-71), the 1956 curve always indicates a higher probability of low wind speeds than does IP3 (1970 and 1971).

A comparison between IP3 and Trap Rock (figure 12) shows that a greater percent probability of low wind speeds are observed at TR until about 5.5 m/s in 1970 and 3.5 m/s in 1971 after which TR has greater probability of higher wind speeds.

The low wind speed conditions occur mostly as part of the valley diurnal system. Here, where atmospheric influences are negligible the driving force to a crossvalley or diagonal wind is the gravitational drainage of the valley slopes. The IP3 location is more influenced by such a drainage pattern and will therefore exhibit higher wind speeds at the lower range than over leveler terrain as surrounds Trap Rock.

The Montrose Point cumulative probability resembles that of Trap Rock. It has been found previously that the Montrose Point direction distribution most resembles IP3.

The variations merely emphasize the dominant effects of strictly local terrain.

5.5.2 Seasonal

Seasonal cumulative frequency distributions of wind speed for Montrose Point, IP3 and Trap Rock are given in Figures 13, 14 and 15, respectively. The winter curves show a greater probability of high speeds at all stations than during the summer with but two exceptions: at IP3 in 1970 the winter curve shows a greater probability for wind speeds below 0.68 m/s and Trap Rock in 1971 below 0.87 m/s. Overall, however, the probability of low wind speeds is generally greatest during the summer season.

The summer cumulative frequency speed curves for the Cape Charles and IP3 aerovanes are given in Figure 16. The Cape Charles exhibits a greater percentage of higher wind speeds up to 8.8 m/s after which the IP3 percentage is greater. Higher wind speeds exist at Cape Charles because there are fewer local effects and less frictional loss over water than over land. The deviation at the high speed winds is generally associated with strong W-N geostrophic flow and the Cape Charles is shielded from these by the Dunderberg and Buckberg Mountains.

5.6 Diurnal Variation of the Mean Wind Direction

A primary analysis for data collected in a valley is to determine the extent of the diurnal wind pattern. This pattern is generally ignored in diffusion meteorology models used for nuclear generating sites. The models used are based on air flow over level and unobstructed terrain. The diurnal wind direction fluctuations and the ever present motion of air (zero frequency of calms) in the valley is a definite positive affect on the diffusion potential.

Seasonal diurnal variations of the mean wind direction for IP3, TR, MP and CC are presented in Figures 17-20, respectively.

The seasonal differences are clearly evident. The winter means fluctuate over shorter ranges exhibiting only a slight diurnal variation between the night and day hours. The winter mean at IP3 ranges from 352°-302° in 1970, and from 332°-305° in 1971. The TR mean ranges from 341°-308° in 1970, and from 332°-305° in 1971. The winter range at MP for 1970 was 357°-319°.

On a point by point basis, during the night, TR's mean direction tends to be west of that at IP3 while MP is about the same as IP3. During the day, IP3 and TR are similar while MP holds more to the north.

The summer diurnal curves at IP3 shows the direction rotating through a complete circle, being downvalley at night and upvalley during the day. From 2300-0800 the means hold from 20°-40°. Thereafter the wind direction backs; rapidly during the transition hours of 1000-1100 and 2000-2200, and slower during the intervening hours at a rate of approximately 8 degrees per hour.

The TR summer diurnal wind pattern is similar to that at IP3. From 2300-1000 a downvalley wind exists ranging from 006°-034°. Both transition periods are sharp, occurring at 1000-1100 and 2000-2100. In 1970 the wind backed into the upvalley direction during the day and veered into the downvalley direction at night. This pattern was reversed in 1971.

MP (Figure 19) shows a similar diurnal summer pattern for 1970 but unlike TR it veered into an upvalley flow during the transition hours.

The question of backing or veering during transition hours may be somewhat more complex than is nominally indicated. Referring it back to original analog records during the transition period one can find <u>either</u> the backing <u>or</u> the veering on any particular day. The diurnal wind itself, however, is related not only to a primary valley circulation (katabatic-anabatic) but a locally induced land-sea breeze. It is the strength of this latter feature that will ultimately determine the backing or veering nature of the diurnal wind during transition hours.

The preceding diurnal analyses indicate that there is small likelihood of a wind persisting in the same direction for as much as 24 hours. The valley structure seems to have a built-in natural mechanism which would prevent such a wind condition from existing particularly at low wind speeds.

5.7 Diurnal Variation of the Mean Wind Speed

Seasonal diurnal variations of mean wind speeds for the various stations are given in Figure 21. The curves indicate that the winter mean speeds are greater than the summer speeds for all the stations throughout the day with both winter and summer maximum speeds occurring between 1300-1500 hours.

5.8 Diurnal Variation of Wind Persistence

The wind persistence is used as a measure of the constancy of a wind direction with time. If a wind always blows from the same direction, the persistence is equal to 1. If the wind is equally likely to blow from all directions or blows half the time from one sector and half the time from the opposite sector, the persistence will be zero.

The seasonal diurnal variation of wind persistence for the various stations is given in Figures 22 and 23. It is clearly evident that the higher persistence values are associated with the winter season. This result is however due to atmospheric conditions which enhance the diffusion, mainly high wind speeds.

Both seasons show a diurnal shift with the summer being more pronounced. Maximum persistence occurs between 04-0600 EST. The times of occurrence of the maximum values of persistence correspond to those times at which the nocturnal downvalley flow is a maximum and the daytime upvalley flow reaches its peak. At CC it almost equals the magnitude of the nighttime maximum. The summer minimums occur between 1100-1200 and 1900-2100 which are about the times of the diurnal wind direction transition periods.

5.9 Spatial Variability of Wind Velocity Between Tower Sites

The annual and seasonal spatial variabilities of wind direction and speed between selected stations are presented in Tables 10 through 17.

Table 10. Annual Spatial Variability of Wind

Direction Between Indian Point 3 and

Trap Rock.

IP3	TR	TR	IP3	IP3	TR	TR	IP3
1970	1970	1970	1970	1971	1971	1971	1971
000.0	354	000.0	012	000.0	358	000.0	008
022.5	010	022.5	025	022.5	017	022.5	024
045.0	024	045.0	041	045.0	033	045.0	043
067.5	048	067.5	070	067.5	047	067.5	065
090.0	078	090.0	094	090.0	076	090.0	091
112.5	107	112.5	126	112.5	104	112.5	118
135.0	129	135.0	154	135.0	134	135.0	146
157.5	147	157.5	171	157.5	154	157.5	167
180.0	168	180.0	188	180.0	177	180.0	182
202.5	190	202.5	206	202.5	200	202.5	199
225.0	212	225.0	224	225.0	224	225.0	219
247.5	239	247.5	256	247.5	250	247.5	244
270.0	270	270.0	285	270.0	277	270.0	271
292.5	295	292.5	301	292.5	296	292.5	293
315.0	308	315.0	321	315.0	313	315.0	318
337.5	330	337.5	353	337.5	333	337.5	346

Table 11. Seasonal Spatial Variability of Wind

Direction Between Indian Point 3 and

Trap Rock (1970).

IP3	TR	TR	IP3	IP3	TR	TR	IP3
1970	1970	1970	1970	1970	1970	1970	1970
Wntr	Wntr	Wntr	Wntr	Smmr	Smmr	Smmr	Smmr
000.0	358	000.0	008	000.0	350	000.0	016
022.5	013	022.5	019	022.5	008	022.5	030
045.0	029	045.0	034	045.0	022	045.0	047
067.5	059	067.5	066	067.5	043	067.5	072
090.0	085	090.0	080	090.0	074	090.0	100
112.5	116	112.5	120	112.5	103	112.5	129
135.0	138	135.0	146	135.0	126	135.0	155
157.5	158	157.5	166	157.5	145	157.5	173
180.0	176	180.0	184	180.0	166	180.0	190
202.5	194	202.5	201	202.5	188	202.5	209
225.0	216	225.0	219	225.0	210	225.0	228
247.5	244	247.5	252	247.5	233	247.5	261
270.0	275	270.0	281	270.0	261	270.0	290
292.5	299	292.5	298	292.5	284	292.5	306
315.0	310	315.0	315	315.0	306	315.0	330
337.5	335	337.5	346	337.5	325	337.5	359

Table 12. Seasonal Spatial Variability of Wind

Direction Between Indian Point 3 and

Trap Rock (1971).

IP3	TR	TR	IP3	IP3	TR	TR	IP3
1971	1971	1971	1971	1971	1971	1971	1971
Wntr	Wntr	Wntr	Wntr	Smmr	Smmr	Smmr	Smmr
000.0	359	000.0	007	000.0	358	000.0	009
022.5	014	022.5	022	022.5	018	022.5	025
045.0	030	045.0	048	045.0	034	045.0	040
067.5	046	067.5	070	067.5	049	067.5	062
090.0	076	090.0	091	090.0	076	090.0	091
112.5	109	112.5	112	112.5	102	112.5	119
135.0	136	135.0	143	135.0	132	135.0	148
157.5	157	157.5	165	157.5	152	157.5	168
180.0	178	180.0	181	180.0	176	180.0	183
202.5	203	202.5	201	202.5	198	202.5	198
225.0	226	225.0	222	225.0	222	225.0	215
247.5	250	247.5	243	247.5	249	247.5	246
270.0	277	270.0	272	270.0	275	270.0	268
292.5	296	292.5	292	292.5	296	292.5	296
315.0	313	315.0	317	315.0	312	315.0	322
337.5	331	337.5	345	337.5	335	337.5	348

Table 13. Annual Spatial Variability of Wind Speed (m/s) Between Indian Point 3 and Trap Rock.

IP3	TR	TR	IP3	IP3	TR	TR	IP3					
1970	1970	1970	1970	1971	1971	1971	1971					
0.25	0.6	0.25	1.0	0.25	0.7	0.25	0.9					
0.75	0.9	0.75	1.4	0.75	0.9	0.75	1.1					
1.25	1.2	1.25	1.8	1.25	1.3	1.25	1.6					
2.0	1.9	2.0	2.5	2.0	2.0	2.0	2.2					
3.0	2.8	3.0	3.3	3.0	3.0	3.0	3.1					
4.0	3.9	4.0	4.1	4.0	4.0	4.0	4.0	5.0	4.9	5.0	4.9	5.0
5.1	5.0	4.9										
7.0	6.8	7.0	6.2	7.0	6.9	7.0	6.5					
9.75	9.9	9.75	8.4	9.75	9.7	9.75	9.0					

Table 14. Seasonal Spatial Variability of Wind Speed (m/s) Between Indian Point 3 and Trap Rock.

IP3	TR	TR	IP3	IP3	TR	TR	IP3
1970	1970	1970	1970	1970	1970	1970	1970
Wntr	Wntr	Wntr	Wntr	Smmr	Smmr	Smmr	Smmr
0.25	0.5	0.25	0.8	0.25	0.6	0.25	1.1
0.75	0.8	0.75	1.3	0.75	0.9	0.75	1.4
1.25	1.3	1.25	1.8	1.25	1.2	1.25	1.9

2.0	2.0	2.0	2.4	2.0	1.8	2.0	2.6
3.0	3.0	3.0	3.3	3.0	2.8	3.0	3.4
4.0	4.1	4.0	4.1	4.0	3.8	4.0	4.2
5.0	5.1	5.0	5.0	5.0	4.7	5.0	4.8
7.0	7.2	7.0	6.4	7.0	6.4	7.0	6.0
9.75	10.0	9.75	8.5	9.75	9.3	9.75	8.1

IP3	TR	TR	IP3	IP3	TR	TR	IP3
1971	1971	1971	1971	1971	1971	1971	1971
Wntr	Wntr	Wntr	Wntr	Summr	Summr	Summr	Summr
0.25	0.6	0.25	1.0	0.25	0.7	0.25	0.7
0.75	0.8	0.75	1.1	0.75	1.0	0.75	1.1
1.25	1.2	1.25	1.7	1.25	1.3	1.25	1.5
2.0	1.9	2.0	2.4	2.0	2.1	2.0	2.2
3.0	2.9	3.0	3.3	3.0	3.0	3.0	3.0
4.0	3.9	4.0	4.2	4.0	4.1	4.0	3.8
5.0	5.1	5.0	5.1	5.0	5.2	5.0	4.6
7.0	6.9	7.0	6.8	7.0	6.8	7.0	6.0
9.75	9.6	9.75	9.1	9.75	9.9	9.75	8.5

Table 15. Annual Spatial Variability of Wind Direction and Speed (m/s) Between Montrose Point and Trap Rock.

Wind Direction				Wind Speed			
MP	TR	TR	MP	MP	TR	TR	MP
1970	1970	1970	1970	1970	1970	1970	1970
000.0	351	000.0	018	0.25	0.8	0.25	0.7
022.5	009	022.5	033	0.75	1.2	0.75	1.0
045.0	023	045.0	049	1.25	1.7	1.25	1.4
067.5	035	067.5	069	2.0	2.5	2.0	1.9
090.0	054	090.0	096	3.0	3.7	3.0	2.7
112.5	089	112.5	127	4.0	4.9	4.0	3.4
135.0	135	135.0	150	5.0	5.7	5.0	4.1
157.5	151	157.5	166	7.0	7.4	7.0	5.6
180.0	173	180.0	189	9.75	10.0	9.75	8.0
202.5	191	202.5	207				
225.0	214	225.0	229				
247.5	236	247.5	253				
270.0	269	270.0	277				
292.5	289	292.5	302				
315.0	309	315.0	324				
337.5	323	337.5	356				

Table 16. Annual Spatial Variability of Wind Direction and Speed (m/s) Between Indian Point 3 and Montrose Point.

Wind Direction				Wind Speed			
IP3 1970	MP 1970	MP 1970	IP3 1970	IP3 1970	MP 1970	MP 1970	IP3 1970
000.0	011	000.0	358	0.25	0.6	0.25	0.9
022.5	029	022.5	015	0.75	0.8	0.75	1.3
045.0	043	045.0	027	1.25	1.2	1.25	1.8
067.5	067	067.5	040	2.0	1.8	2.0	2.6
090.0	088	090.0	070	3.0	2.6	3.0	3.5
112.5	123	112.5	108	4.0	3.5	4.0	4.5
135.0	142	135.0	153	5.0	4.4	5.0	5.3
157.5	158	157.5	167	7.0	6.2	7.0	6.9
180.0	175	180.0	183	9.75	9.1	9.75	9.1
202.5	198	202.5	198				
225.0	216	225.0	220				
247.5	238	247.5	244				
270.0	272	270.0	275				
292.5	304	292.5	297				
315.0	320	315.0	315				
337.5	338	337.5	330				

Table 17. Summer (1970) Spatial Variability of Wind Direction and Speed (m/s) Between Cape Charles and Indian Point 3.

CC	IP3	IP3	CC	CC	IP3	IP3	CC
1970	1970	1970	1970	1970	1970	1970	1970
Smmr	Smmr	Smmr	Smmr	Smmr	Smmr	Smmr	Smmr
000.0	001	000.0	007	0.25	0.9	0.25	1.1
022.5	017	022.5	022	0.75	1.1	0.75	1.4
045.0	031	045.0	028	1.25	1.5	1.25	1.9
067.5	049	067.5	044	2.0	1.8	2.0	2.5
090.0	088	090.0	069	3.0	2.5	3.0	3.4
112.5	102	112.5	107	4.0	3.4	4.0	4.5
135.0	122	135.0	153	5.0	4.4	5.0	5.3
157.5	146	157.5	187	7.0	5.4	7.0	6.0
180.0	163	180.0	212	9.75	7.2	9.75	7.3
202.5	180	202.5	221				
225.0	196	225.0	234				
247.5	216	247.5	247				
270.0	285	270.0	268				
292.5	301	292.5	286				
315.0	321	315.0	317				
337.5	346	337.5	355				

The station listed first in each tabular set is the independent variable.

When the role of independent and dependent stations are reversed, the relationships between the mean values are not, a priori, similarly inverted. This discrepancy may be attributed, in part, to the effects of local topography as well as to a reaction time lag engendered by the spatial distance between stations.

The significance of any degree of change in magnitude of speed or wind direciton angle between stations are arbitrary. An angular change of $ 6 10^{o} and speed change of $ 6 0.5 m/s may be considered.

The spatial variability analysis indicated:

1) A general backing of wind direction between IP3 and TR, with TR having a lower speed at the low range and a higher speed at the high range. Seasonal analysis showed the greatest variations occurring during the “summer” months.
2) A general backing of wind direction between MP and TR with speeds being greater at TR.
3) Based on the comparable data available (1970), the wind directions at IP3 and MP were similar to each other and in the relationship to TR. The magnitude of the wind speeds were similar at IP3 and TR, with each being generally stronger than at MP.

References

Halitsky, J., J. Laznow and D. Leahey (1970): Wind Observations at Indian Point, Montrose and Bowline Point. 31 August 1968 to

30 June 1969. New York University, Department of Meteorology and Oceanography Technical Report TR-70-3.

Halitsky, J., E. J. Kaplin and J. Laznow (1971): Wind Observations at Indian Point 26 November 1969-1 October 1970. New York University Department of Meteorology and Oceanography Technical Report TR-71-3.

Leahey, D. M. and J. Halitsky (1971): Low Wind Speed Turbulence Statistics and Related Diffusion Estimates for Indian Point, N.Y. New York University Department of Meteorology and Oceanography Technical Report TR-71-10.

FSAR: Final Facility Description and Safety Analysis Report. Consolidated Edison Company of New York, Inc. Nuclear Generating Unit No. 2. Exhibit B-8.

Davidson, B. and J. Halitsky (1955): A Micrometeorological Survey of the Buchanan, New York Area, Summary of Program to 1 December 1955. New York University Department of Meteorology and Oceanography Technical Report 372.1.

Davidson, B. and J. Halitsky (1957): Evaluation of Potential Radiation Hazard Resulting from Assumed Release of Radioactive Wastes to Atmosphere from Proposed Buchanan Nuclear Power Plant. New York University Department of Meteorology and Oceanography Technical Report 372.3.

U. S. Atomic Energy Commission Safety Guide 23 – Onsite Meteorological Programs, February 17, 1972.

Appendix Table 1. Indian Point B(3) Joint Frequency
Distribution of Wind Speed and Direction
for Pasquill Stability Category A –
January 1, 1970-December 31, 1970.

Wind Direction		Wind Speed (mph) 01-03	04-07	08-12	13-18	19-24	>24	Miss	Total
349- 11	N	.0010	.0086	.0097	.0025	.0002	.0000	.0001	.0222
12- 33	NNE	.0008	.0059	.0031	.0010	.0000	.0000	.0001	.0110
34- 56	NE	.0015	.0041	.0008	.0005	.0001	.0000	.0001	.0071
57- 78	ENE	.0005	.0006	.0000	.0003	.0002	.0000	.0000	.0016
79-101	E	.0003	.0005	.0002	.0002	.0000	.0000	.0000	.0013
102-123	ESE	.0005	.0005	.0003	.0002	.0000	.0000	.0000	.0015
124-146	SE	.0013	.0010	.0010	.0016	.0002	.0000	.0000	.0051
147-168	SSE	.0016	.0074	.0109	.0078	.0003	.0000	.0000	.0280
169-191	S	.0016	.0122	.0081	.0015	.0000	.0000	.0000	.0234
192-213	SSW	.0010	.0071	.0033	.0002	.0000	.0000	.0001	.0118
214-236	SW	.0009	.0034	.0016	.0006	.0000	.0000	.0000	.0065
237-258	WSW	.0010	.0010	.0011	.0002	.0000	.0000	.0001	.0035
259-281	W	.0009	.0022	.0030	.0013	.0007	.0001	.0001	.0082
282-303	WNW	.0005	.0014	.0031	.0053	.0024	.0008	.0010	.0144
304-326	NW	.0005	.0016	.0042	.0054	.0035	.0007	.0005	.0163
327-348	NNW	.0007	.0039	.0049	.0041	.0011	.0001	.0000	.0149
Calm			.0000						.0000
Miss			.0000	.0000	.0000	.0000	.0000	.0000	.0022
Total		.0145	.0614	.0554	.0327	.0089	.0017.	0043.	1790

Indian Point B(3) Joint Frequency Distribution of Wind Speed and Direction for Pasquill Stability Category B – January 1, 1970-December 31, 1970.

Wind Direction		Wind Speed (mph) 01-03	04-07	08-12	13-18	19-24	>24	Miss	Total
349- 11	N	.0002	.0007	.0006	.0001	.0000	.0000	.0000	.0016
12- 33	NNE	.0001	.0009	.0006	.0000	.0000	.0000	.0000	.0016
34- 56	NE	.0005	.0008	.0002	.0000	.0000	.0000	.0000	.0015
57- 78	ENE	.0001	.0003	.0000	.0000	.0000	.0000	.0001	.0006
79-101	E	.0002	.0000	.0000	.0000	.0000	.0000	.0000	.0002
102-123	ESE	.0003	.0007	.0000	.0000	.0000	.0000	.0000	.0010
124-146	SE	.0000	.0003	.0001	.0000	.0000	.0000	.0000	.0005
147-168	SSE	.0001	.0002	.0008	.0008	.0001	.0000	.0000	.0021
169-191	S	.0003	.0010	.0005	.0005	.0000	.0000	.0000	.0023
192-213	SSW	.0000	.0005	.0003	.0002	.0000	.0000	.0000	.0010
214-236	SW	.0002	.0001	.0000	.0000	.0000	.0000	.0000	.0003
237-258	WSW	.0007	.0002	.0000	.0001	.0000	.0000	.0000	.0010
259-281	W	.0000	.0000	.0002	.0001	.0000	.0000	.0000	.0003
282-303	WNW	.0001	.0000	.0001	.0003	.0003	.0001	.0000	.0010
304-326	NW	.0001	.0001	.0007	.0007	.0003	.0003	.0000	.0023
327-348	NNW	.0000	.0001	.0005	.0001	.0000	.0001	.0000	.0008
Calm			.0000						.0000
Miss			.0000	.0000	.0000	.0000	.0000	.0000	.0000
Total		.0031	.0061	.0046	.0030	.0008.	0006.	0001	.0182

Appendix Table 1 (continued)

Indian Point B(3) Joint Frequency Distribution of Wind Speed and Direction for Pasquill Stability Category C – January 1, 1970-December 31, 1970

Wind Direction		Wind Speed (mph) 01-03	04-07	08-12	13-18	19-24	>24	Miss	Total
349- 11	N	.0000	.0007	.0008	.0006	.0000	.0000	.0000	.0021
12- 33	NNE	.0001	.0008	.0009	.0003	.0001	.0000	.0000	.0023
34- 56	NE	.0002	.0006	.0003	.0001	.0000	.0000	.0000	.0013
57- 78	ENE	.0002	.0000	.0003	.0000	.0000	.0000	.0000	.0006
79-101	E	.0001	.0001	.0000	.0001	.0000	.0000	.0000	.0003

102-123	ESE	.0000	.0001	.0000	.0000	.0000	.0000	.0000	.0001
124-146	SE	.0002	.0005	.0006	.0001	.0000	.0000	.0000	.0014
147-168	SSE	.0002	.0013	.0018	.0008	.0000	.0000	.0000	.0041
169-191	S	.0002	.0014	.0008	.0001	.0000	.0000	.0000	.0025
192-213	SSW	.0001	.0005	.0003	.0001	.0000	.0000	.0000	.0010
214-236	SW	.0003	.0005	.0002	.0003	.0000	.0000	.0000	.0014
237-258	WSW	.0001	.0000	.0001	.0003	.0000	.0000	.0000	.0006
259-281	W	.0000	.0002	.0003	.0008	.0001	.0000	.0000	.0015
282-303	WNW	.0003	.0005	.0002	.0015	.0010	.0005	.0000	.0040
304-326	NW	.0000	.0000	.0001	.0001	.0010	.0006	.0000	.0018
327-348	NNW	.0003	.0005	.0005	.0007	.0006	.0000	.0000	.0025
Calm		.0000							.0000
Miss		.0000	.0000	.0000	.0000	.0002	.0001		.0006
Total		.0026	.0074	.0074	.0061	.0031	.0011	.0002	.0280

Indian Point B(3) Joint Frequency Distribution of Wind Speed and Direction for Pasquill Stability Category D - January 1, 1970-December 31, 1970.

Wind		Wind Speed (mph)							
Direction		01-03	04-07	08-12	13-18	19-24	>24	Miss	Total
349- 11	N	.0024	.0143	.0173	.0051	.0007	.0001	.0014	.0413
12- 33	NNE	.0042	.0178	.0122	.0029	.0006	.0001	.0014	.0392
34- 56	NE	.0031	.0078	.0026	.0003	.0003	.0000	.0006	.0147
57- 78	ENE	.0022	.0016	.0001	.0001	.0000	.0000	.0006	.0046
79-101	E	.0021	.0021	.0009	.0000	.0000	.0000	.0002	.0053
102-123	ESE	.0026	.0016	.0011	.0002	.0000	.0000	.0003	.0059
124-146	SE	.0025	.0053	.0038	.0003	.0002	.0000	.0001	.0122
147-168	SSE	.0023	.0110	.0098	.0032	.0000	.0000	.0003	.0266
169-191	S	.0025	.0098	.0053	.0010	.0000	.0000	.0003	.0190
192-213	SSW	.0030	.0038	.0021	.0008	.0000	.0000	.0000	.0096
214-236	SW	.0023	.0017	.0009	.0006	.0001	.0000	.0003	.0059
237-258	WSW	.0008	.0016	.0007	.0008	.0000	.0000	.0000	.0039
259-281	W	.0011	.0019	.0019	.0016	.0008	.0001	.0006	.0081
282-303	WNW	.0011	.0010	.0051	.0095	.0062	.0011	.0001	.0242
304-326	NW	.0010	.0019	.0061	.0109	.0074	.0026	.0000	.0300
327-348	NNW	.0009	.0054	.0069	.0079	.0022	.0002	.0002	.0237

Calm	.0000							.0000
Miss	.0000	.0000	.0000	.0001	.0005	.0000		.0010
Total	.0342	.0886	.0768	.0454	.0190	.0043	.0070	.2753

Appendix Table 1 (continued)

Indian Point B(3) Joint Frequency Distribution of Wind Speed and Direction for Pasquill Stability Category E – January 1, 1970-December 31, 1970.

Wind		Wind Speed (mph)							
Direction		01-03	04-07	08-12	13-18	19-24	>24	Miss	Total
349- 11	N	.0045	.0067	.0043	.0014	.0001	.0000	.0002	.0173
12- 33	NNE	.0045	.0166	.0080	.0005	.0003	.0000	.0006	.0304
34- 56	NE	.0030	.0078	.0019	.0001	.0001	.0000	.0008	.0137
57- 78	ENE	.0023	.0018	.0003	.0000	.0000	.0000	.0000	.0045
79-101	E	.0011	.0023	.0011	.0001	.0001	.0000	.0000	.0048
102-123	ESE	.0014	.0029	.0016	.0001	.0000	.0000	.0001	.0061
124-146	SE	.0027	.0035	.0017	.0001	.0003	.0000	.0001	.0086
147-168	SSE	.0022	.0053	.0041	.0016	.0005	.0002	.0000	.0138
169-191	S	.0017	.0062	.0043	.0003	.0002	.0003	.0001	.0133
192-213	SSW	.0027	.0063	.0045	.0005	.0000	.0000	.0000	.0139
214-236	SW	.0024	.0045	.0024	.0006	.0001	.0001	.0000	.0101
237-258	WSW	.0019	.0027	.0016	.0008	.0007	.0002	.0000	.0080
259-281	W	.0015	.0015	.0034	.0014	.0008	.0005	.0000	.0090
282-303	WNW	.0010	.0006	.0038	.0054	.0041	.0014	.0000	.0162
304-326	NW	.0008	.0017	.0054	.0071	.0016	.0005	.0001	.0171
327-348	NNW	.0017	.0034	.0046	.0027	.0001	.0000	.0001	.0127
Calm		.0000							.0000
Miss		.0005	.0007	.0009	.0007	.0010	.0000		.0048
Total		.0359	.0744	.0541	.0233	.0102	.0032	.0032	.2043

Indian Point B(3) Joint Frequency Distribution of Wind Speed and Direction for Pasquill Stability Category F – January 1, 1970-

December 31, 1970

Wind Direction		Wind Speed (mph) 01-03	04-07	08-12	13-18	19-24	>24	Miss	Total
349- 11	N	.0047	.0022	.0008	.0001	.0000	.0000	.0000	.0078
12- 33	NNE	.0030	.0075	.0022	.0000	.0000	.0000	.0000	.0127
34- 56	NE	.0034	.0042	.0009	.0000	.0000	.0000	.0005	.0090
57- 78	ENE	.0015	.0011	.0001	.0000	.0000	.0000	.0000	.0027
79-101	E	.0011	.0005	.0000	.0000	.0000	.0000	.0000	.0016
102-123	ESE	.0010	.0006	.0000	.0000	.0000	.0000	.0000	.0016
124-146	SE	.0009	.0006	.0000	.0000	.0000	.0000	.0000	.0015
147-168	SSE	.0023	.0027	.0005	.0001	.0002	.0000	.0000	.0058
169-191	S	.0016	.0035	.0009	.0000	.0000	.0000	.0000	.0061
192-213	SSW	.0013	.0032	.0008	.0000	.0000	.0000	.0000	.0053
214-236	SW	.0024	.0040	.0006	.0000	.0000	.0000	.0001	.0071
237-258	WSW	.0018	.0009	.0009	.0000	.0000	.0000	.0001	.0038
259-281	W	.0011	.0003	.0003	.0000	.0003	.0000	.0001	.0023
282-303	WNW	.0007	.0002	.0003	.0002	.0001	.0001	.0000	.0017
304-326	NW	.0011	.0002	.0001	.0000	.0002	.0000	.0000	.0017
327-348	NNW	.0013	.0007	.0003	.0002	.0000	.0000	.0000	.0025
Calm		.0000							.0000
Miss		.0007	.0001	.0000	.0000	.0000	.0000		.0013
Total		.0300	.0327	.0088	.0007	.0009	.0001	.0013	.0744

Appendix Table 1 (Continued)

Indian Point B(3) Joint Frequency Distribution of Wind Speed and Direction for Pasquill Stability Category C – January 1, 1970- December 31, 1970.

Wind Direction		Wind Speed (mph) 01-03	04-07	08-12	13-18	19-24	>24	Miss	Total
349- 11	N	.0019	.0010	.0003	.0000	.0000	.0000	.0000	.0033
12- 33	NNE	.0018	.0037	.0008	.0000	.0000	.0000	.0001	.0064
34- 56	NE	.0030	.0021	.0000	.0000	.0000	.0000	.0000	.0050
57- 78	ENE	.0018	.0006	.0000	.0000	.0000	.0000	.0000	.0024

79-101	E	.0008	.0003	.0000	.0000	.0000	.0000	.0000	.0011
102-123	ESE	.0006	.0002	.0000	.0000	.0000	.0000	.0000	.0008
124-146	SE	.0014	.0007	.0000	.0000	.0000	.0000	.0000	.0021
147-168	SSE	.0013	.0014	.0002	.0000	.0000	.0000	.0000	.0029
169-191	S	.0009	.0015	.0005	.0000	.0000	.0000	.0000	.0029
192-213	SSW	.0024	.0024	.0003	.0000	.0000	.0000	.0002	.0054
214-236	SW	.0019	.0013	.0002	.0000	.0000	.0000	.0001	.0035
237-258	WSW	.0015	.0006	.0003	.0000	.0000	.0000	.0001	.0025
259-281	W	.0018	.0005	.0000	.0000	.0000	.0000	.0000	.0023
282-303	WNW	.0015	.0002	.0000	.0000	.0000	.0000	.0000	.0017
304-326	NW	.0011	.0000	.0000	.0000	.0000	.0000	.0000	.0011
327-348	NNW	.0015	.0005	.0001	.0001	.0000	.0000	.0000	.0022
Calm		.0000							.0000
Miss		.0000	.0000	.0000	.0000	.0000	.0000		.0008
Total		.0253	.0168	.0029	.0001	.0000	.0000	.0014	.0464

Appendix Table 2. Indian Point B(3) Point Frequency Distribution of Wind Speed and Direction for Pasquill Stability Category A - January 1, 1971-December 31, 1971

Wind Direction		Wind Speed (mph) 01-03	04-07	08-12	13-18	19-24	>24	Miss	Total
349- 11	N	.0006	.0054	.0057	.0021	.0011	.0000	.0001	.0150
12- 33	NNE	.0003	.0043	.0018	.0005	.0000	.0000	.0000	.0070
34- 56	NE	.0008	.0011	.0002	.0000	.0000	.0000	.0000	.0022
57- 78	ENE	.0001	.0008	.0000	.0000	.0000	.0000	.0000	.0009
79-101	E	.0000	.0000	.0002	.0000	.0000	.0000	.0000	.0002
102-123	ESE	.0001	.0005	.0005	.0001	.0000	.0000	.0000	.0011
124-146	SE	.0002	.0005	.0018	.0016	.0001	.0000	.0000	.0042
147-168	SSE	.0006	.0050	.0065	.0015	.0003	.0000	.0000	.0140
169-191	S	.0005	.0062	.0041	.0007	.0000	.0000	.0000	.0114
192-213	SSW	.0003	.0029	.0014	.0006	.0000	.0000	.0000	.0051
214-236	SW	.0003	.0015	.0007	.0005	.0000	.0000	.0000	.0030

237-258	WSW	.0003	.0018	.0011	.0024	.0009	.0003	.0000	.0070
259-281	W	.0003	.0009	.0039	.0022	.0017	.0002	.0000	.0093
282-303	WNW	.0008	.0008	.0042	.0050	.0027	.0009	.0000	.0145
304-326	NW	.0005	.0016	.0031	.0046	.0035	.0006	.0000	.0138
327-348	NNW	.0003	.0033	.0046	.0039	.0007	.0001	.0000	.0129
Calm		.0000							.0000
Miss		.0000	.0000	.0000	.0000	.0000	.0000		.0002
Total		.0062	.0366	.0399	.0255	.0112	.0022	.0003	.1219

Indian Point B(3) Joint Frequency Distribution of Wind Speed and Direction
for Pasquill Stability Category B – January 1, 1971-December 31, 1971.

Wind		Wind Speed (mph)							
Direction		01-03	04-07	08-12	13-18	19-24	>24	Miss	Total
349- 11	N	.0000	.0013	.0009	.0001	.0000	.0000	.0000	.0023
12- 33	NNE	.0000	.0008	.0000	.0000	.0000	.0000	.0000	.0008
34- 56	NE	.0002	.0001	.0001	.0000	.0000	.0000	.0000	.0005
57- 78	ENE	.0002	.0000	.0000	.0000	.0000	.0000	.0000	.0002
79-101	E	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
102-123	ESE	.0000	.0001	.0001	.0001	.0000	.0000	.0000	.0003
124-146	SE	.0000	.0001	.0003	.0002	.0000	.0000	.0000	.0007
147-168	SSE	.0002	.0008	.0014	.0001	.0001	.0000	.0000	.0026
169-191	S	.0002	.0009	.0000	.0000	.0000	.0000	.0000	.0011
192-213	SSW	.0003	.0008	.0002	.0001	.0000	.0000	.0000	.0015
214-236	SW	.0001	.0001	.0002	.0001	.0000	.0000	.0000	.0006
237-258	WSW	.0001	.0002	.0000	.0002	.0001	.0001.	0000	.0007
259-281	W	.0000	.0002	.0005	.0000	.0002	.0001	.0000	.0010
282-303	WNW	.0000	.0001	.0005	.0010	.0003	.0000	.0000	.0019
304-326	NW	.0005	.0002	.0003	.0006	.0003	.0001	.0000	.0021
327-348	NNW	.0000	.0003	.0002	.0006	.0000	.0000	.0000	.0011
Calm		.0000							.0000
Miss		.0000	.0000	.0000	.0000	.0000	.0000		.0000
Total		.0018	.0062	.0048	.0032	.0011	.0003	.0000	.0175

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Appendix Table 2 (continued)

Indian Point B(3) Joint Frequency Distribution of Wind Speed and Direction for Pasquill Stability Category C – January 1, 1971-December 31, 1971.

Wind Direction		01-03	04-07	08-12	13-18	19-24	>24	Miss	Total
349- 11	N	.0003	.0014	.0006	.0002	.0000	.0000	.0000	.0025
12- 33	NNE	.0001	.0006	.0006	.0001	.0000	.0000	.0001	.0015
34- 56	NE	.0001	.0003	.0001	.0000	.0000	.0000	.0000	.0006
57- 78	ENE	.0000	.0000	.0001	.0000	.0000	.0000	.0000	.0001
79-101	E	.0001	.0002	.0001	.0000	.0000	.0000	.0000	.0005
102-123	ESE	.0003	.0001	.0002	.0000	.0000	.0000	.0000	.0007
124-146	SE	.0001	.0003	.0005	.0000	.0000	.0000	.0000	.0009
147-168	SSE	.0001	.0006	.0003	.0002	.0000	.0000	.0000	.0013
169-191	S	.0005	.0011	.0005	.0002	.0000	.0000	.0000	.0023
192-213	SSW	.0000	.0005	.0001	.0000	.0000	.0000	.0000	.0006
214-236	SW	.0003	.0001	.0001	.0001	.0001	.0000	.0000	.0008
237-258	WSW	.0001	.0001	.0003	.0001	.0002	.0001	.0000	.0010
259-281	W	.0001	.0001	.0003	.0003	.0001	.0000	.0000	.0010
282-303	WNW	.0001	.0001	.0006	.0010	.0000	.0001	.0000	.0019
304-326	NW	.0003	.0002	.0002	.0005	.0003	.0000	.0000	.0016
327-348	NNW	.0000	.0007	.0007	.0001	.0001	.0000	.0000	.0016
Calm		.0000							.0000
Miss		.0000	.0000	.0000	.0000	.0000	.0000		.0000
Total		.0027	.0065	.0054	.0030	.0009	.0002	.0001	.0180

Indian Point B(3) Joint Frequency Distribution of Wind Speed and Direction for Pasquill Stability Category D – January 1, 1971-December 31, 1971.

Wind Wind Speed (mph)

Direction		01-03	04-07	08-12	13-18	19-24	>24	Miss	Total
349- 11	N	.0014	.0079	.0067	.0034	.0005	.0000	.0001	.0200
12- 33	NNE	.0023	.0066	.0050	.0015	.0000	.0000	.0000	.0154
34- 56	NE	.0029	.0026	.0014	.0005	.0000	.0000	.0001	.0074
57- 78	ENE	.0010	.0011	.0018	.0003	.0000	.0000	.0000	.0043
79-101	E	.0015	.0008	.0011	.0000	.0000	.0000	.0000	.0034
102-123	ESE	.0011	.0015	.0002	.0000	.0000	.0000	.0000	.0029
124-146	SE	.0013	.0026	.0026	.0008	.0001	.0000	.0000	.0074
147-168	SSE	.0013	.0040	.0065	.0014	.0001	.0000	.0000	.0133
169-191	S	.0015	.0030	.0014	.0009	.0001	.0000	.0000	.0069
192-213	SSW	.0009	.0017	.0018	.0006	.0000	.0001	.0000	.0051
214-236	SW	.0017	.0013	.0013	.0019	.0001	.0001	.0000	.0064
237-258	WSW	.0008	.0010	.0026	.0018	.0008	.0002	.0000	.0073
259-281	W	.0005	.0005	.0024	.0041	.0023	.0001	.0000	.0098
282-303	WNW	.0008	.0002	.0056	.0051	.0024	.0010	.0003	.0156
304-326	NW	.0009	.0002	.0031	.0055	.0014	.0000	.0001	.0112
327-348	NNW	.0003	.0010	.0039	.0019	.0008	.0001	.0001	.0082
Calm			.0000						.0000
Miss			.0000	.0000	.0000	.0000	.0000	.0000	.0001
Total		.0201	.0361	.0476	.0298	.0086	.0017	.0009	.1449

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71.

Appendix Table 2 (continued)

Indian Point B (3) Joint Frequency Distribution of Wind Speed and Direction for Pasquill Stability Category E – January 1, 1971-December 31, 1971.

Wind Direction		Wind Speed (mph) 01-03	04-07	08-12	13-18	19-24	>24	Miss	Total
349- 11	N	.0027	.0072	.0094	.0037	.0013	.009	.0001	.0253
12- 33	NNE	.0029	.0144	.0129	.0019	.0002	.0000	.0000	.0324
34- 56	NE	.0035	.0063	.0019	.0001	.0001	.0000	.0010	.0130
57- 78	ENE	.0021	.0021	.0014	.0000	.0000	.0000	.0002	.0057
79-101	E	.0016	.0021	.0001	.0000	.0000	.0000	.0000	.0038

102-123	ESE	.0015	.0034	.0006	.0000	.0000	.0000	.0000	.0055
124-146	SE	.0019	.0063	.0024	.0002	.0000	.0000	.0000	.0109
147-168	SSE	.0016	.0061	.0058	.0009	.0000	.0000	.0000	.0144
169-191	S	.0019	.0058	.0038	.0009	.0001	.0000	.0000	.0126
192-213	SSW	.0016	.0034	.0031	.0005	.0002	.0000	.0000	.0088
214-236	SW	.0010	.0013	.0015	.0008	.0002	.0000	.0000	.0048
237-258	WSW	.0017	.0014	.0026	.0014	.0005	.0002	.0000	.0078
259-281	W	.0006	.0015	.0050	.0025	.0009	.0001	.0000	.0106
282-303	WNW	.0006	.0015	.0051	.0051	.0024	.0006	.0010	.0164
304-326	NW	.0005	.0010	.0063	.0054	.0013	.0003	.0001	.0149
327-348	NNW	.0008	.0016	.0035	.0030	.0006	.0001	.0000	.0096
Calm		.0000							.0000
Miss		.0000	.0000	.0000	.0000	.0000	.0000		.0000
Total		.0265	.0653	.0655	.0264	.0078	.0023	.0025	.1963

Indian Point B(3) Joint Frequency Distribution of Wind Speed and Direction for Pasquill Stability Category F – January 1, 1971-December 31, 1971.

Wind Direction		Wind Speed (mph) 01-03	04-07	08-12	13-18	19-24	>24	Miss	Total
349- 11	N	.0019	.0042	.0015	.0001	.0000	.0000	.0000	.0078
12- 33	NNE	.0026	.0113	.0029	.0000	.0000	.0000	.0000	.0168
34- 56	NE	.0024	.0046	.0008	.0000	.0000	.0000	.0000	.0078
57- 78	ENE	.0010	.0009	.0000	.0000	.0000	.0000	.0000	.0019
79-101	E	.0008	.0003	.0001	.0000	.0000	.0000	.0000	.0013
102-123	ESE	.0011	.0005	.0000	.0000	.0000	.0000	.0000	.0016
124-146	SE	.0010	.0023	.0000	.0000	.0000	.0000	.0000	.0033
147-168	SSE	.0013	.0024	.0005	.0001	.0000	.0000	.0000	.0042
169-191	S	.0014	.0040	.0002	.0001	.0000	.0000	.0000	.0057
192-213	SSW	.0016	.0030	.0006	.0001	.0000	.0000	.0000	.0053
214-236	SW	.0008	.0023	.0005	.0008	.0000	.0000	.0000	.0043
237-258	WSW	.0010	.0016	.0007	.0001	.0002	.0000	.0000	.0037
259-281	W	.0007	.0011	.0008	.0001	.0000	.0000	.0000	.0027
282-303	WNW	.0013	.0005	.0007	.0005	.0001	.0000	.0000	.0030

304-326	NW	.0006	.0008	.0003	.0000	.0000	.0001	.0000	.0018
327-348	NNW	.0010	.0014	.0006	.0001	.0000	.0000	.0000	.0031
Calm		.0000							.0000
Miss		.0000	.0000	.0000	.0000	.0000	.0000		.0000
Total		.0206	.0412	.0101	.0021	.0003	.0001	.0000	.0743

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72.

Appendix Table 2 (continued)

Indian Point B(3) Joint Frequency Distribution of Wind Speed and Direction for Pasquill Stability Category G – January 1, 1971-December 31, 1971.

Wind Direction		Wind Speed (mph) 01-03	04-07	08-12	13-18	19-24	>24	Miss	Total
349- 11	N	.0031	.0034	.0005	.0000	.0000	.0000	.0000	.0070
12- 33	NNE	.0027	.0064	.0008	.0000	.0000	.0000	.0000	.0099
34- 56	NE	.0031	.0026	.0000	.0000	.0000	.0000	.0000	.0057
57- 78	ENE	.0019	.0003	.0000	.0000	.0000	.0000	.0000	.0023
79-101	E	.0009	.0006	.0000	.0000	.0000	.0000	.0000	.0015
102-123	ESE	.0009	.0001	.0000	.0000	.0000	.0000	.0000	.0010
124-146	SE	.0013	.0005	.0000	.0001	.0000	.0000	.0000	.0018
147-168	SSE	.0016	.0019	.0002	.0001	.0000	.0000	.0000	.0039
169-191	S	.0024	.0030	.0000	.0000	.0000	.0000	.0000	.0054
192-213	SSW	.0027	.0027	.0000	.0000	.0000	.0000	.0000	.0055
214-236	SW	.0022	.0016	.0003	.0000	.0000	.0000	.0000	.0041
237-258	WSW	.0019	.0013	.0002	.0001	.0000	.0000	.0000	.0035
259-281	W	.0021	.0007	.0002	.0000	.0000	.0000	.0000	.0030
282-303	WNW	.0014	.0002	.0002	.0000	.0000	.0000	.0000	.0018
304-326	NW	.0013	.0007	.0000	.0000	.0000	.0000	.0000	.0019
327-348	NNW	.0022	.0011	.0002	.0000	.0000	.0000	.0000	.0035
Calm		.0000							.0000
Miss		.0000	.0000	.0000	.0000	.0000	.0000		.0000

Total	.0317	.0272	.0027	.0003	.0000	.0000	.0000	.0620

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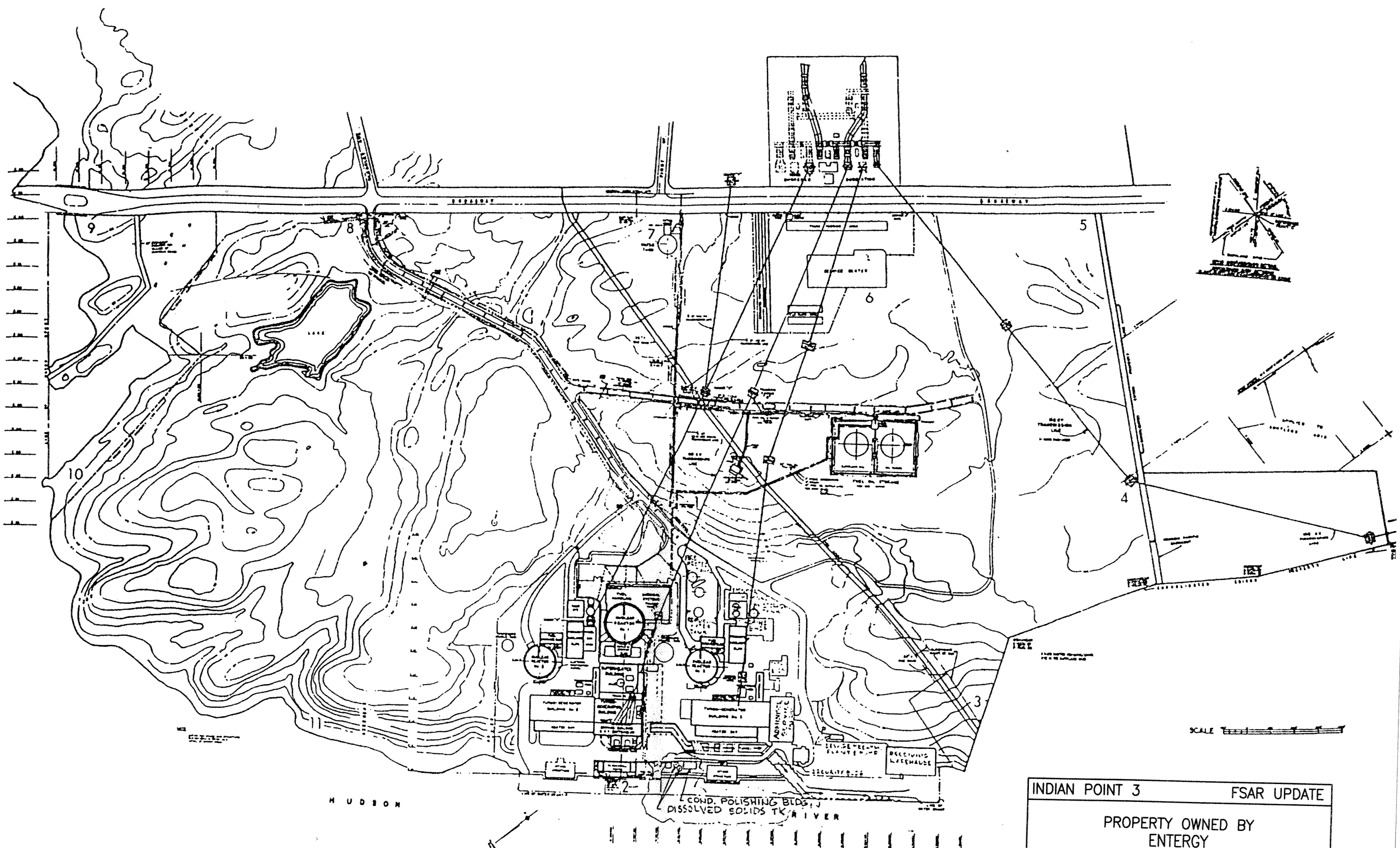
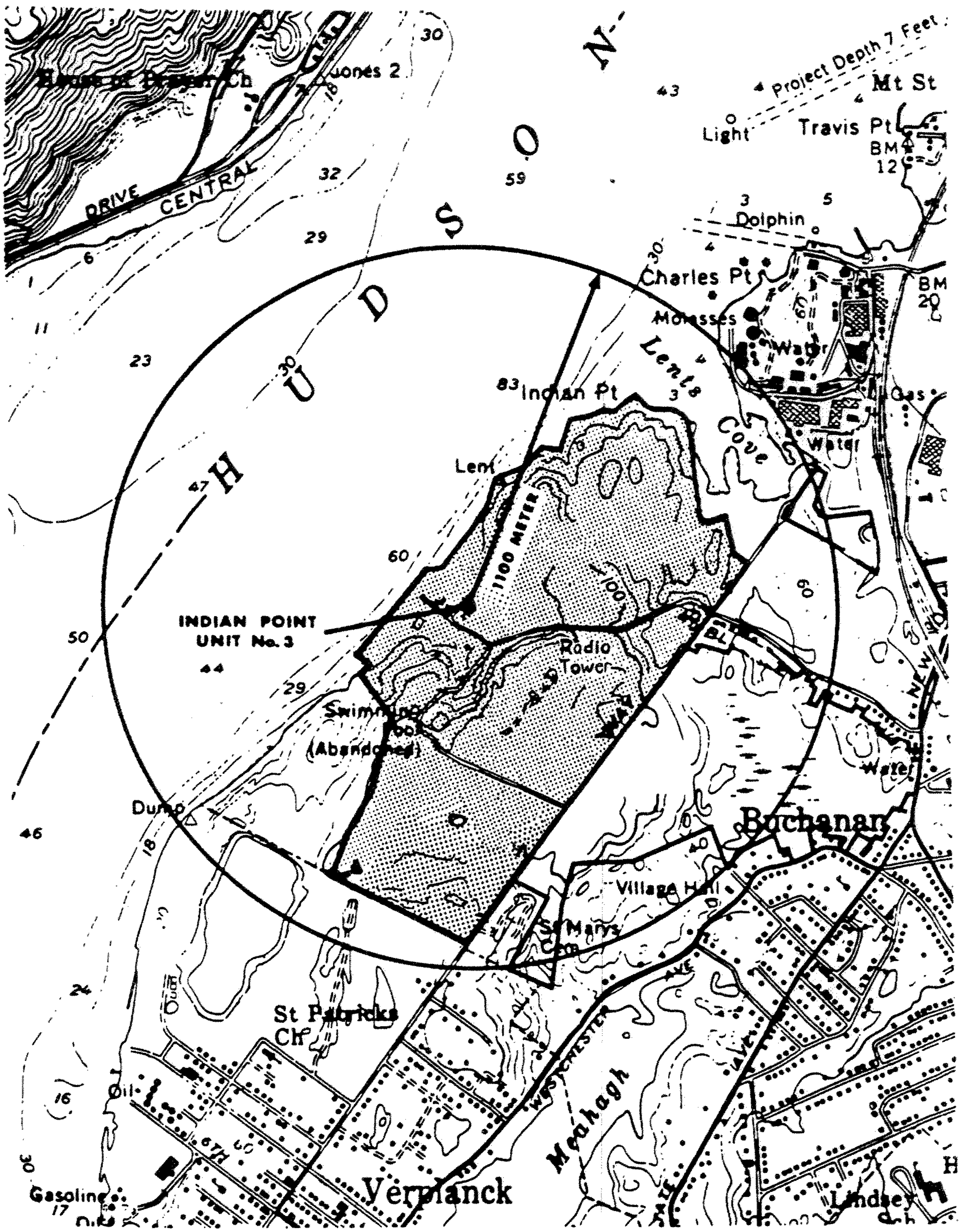
Figure 2.2-3, drawing 9321-F-64513.

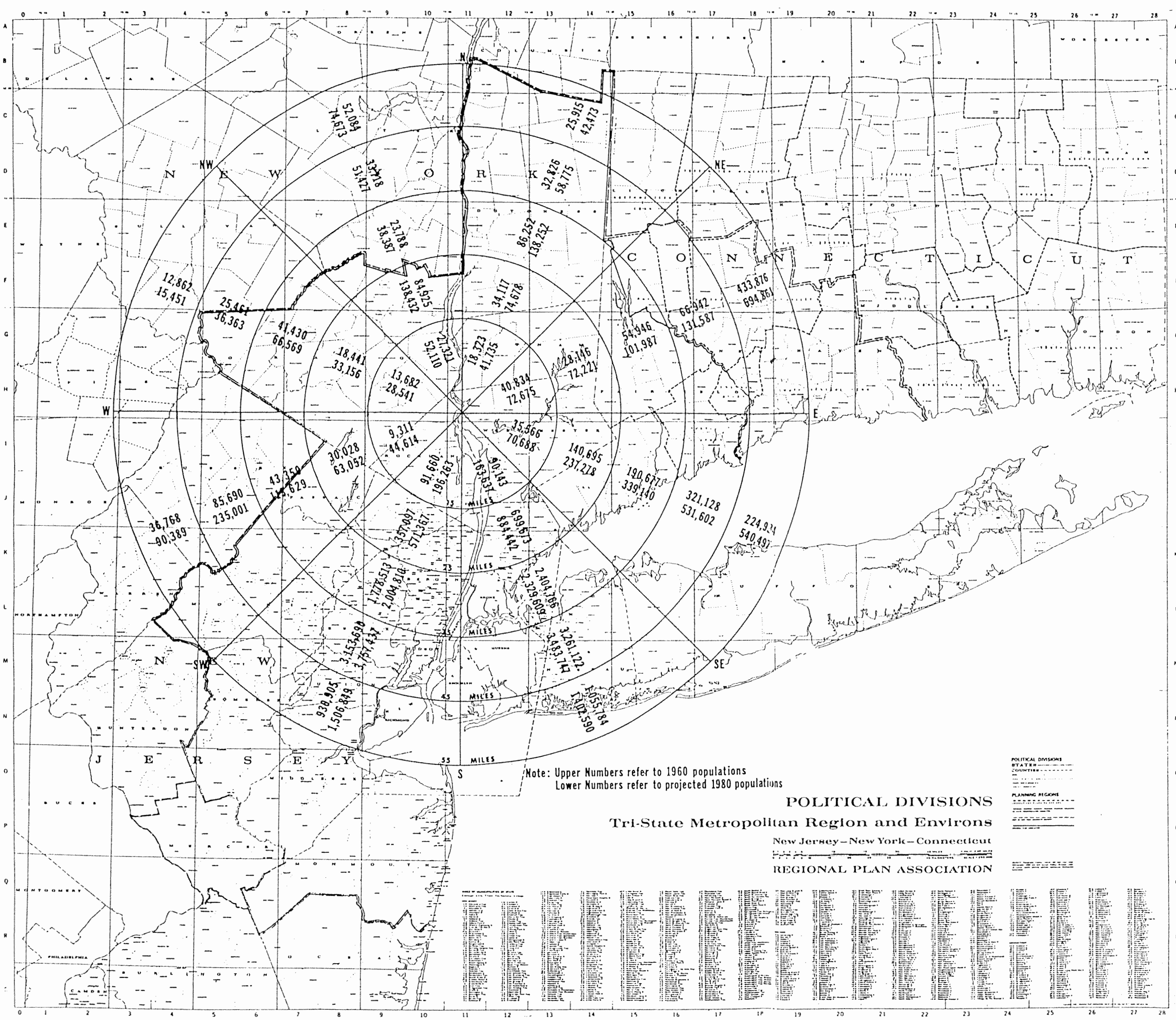
Redacted by NRC staff as sensitive information.



INDIAN POINT 3	FSAR UPDATE
PROPERTY OWNED BY ENTERGY	
REV. 2 OCT 2003	FIGURE NO. 2.2-2



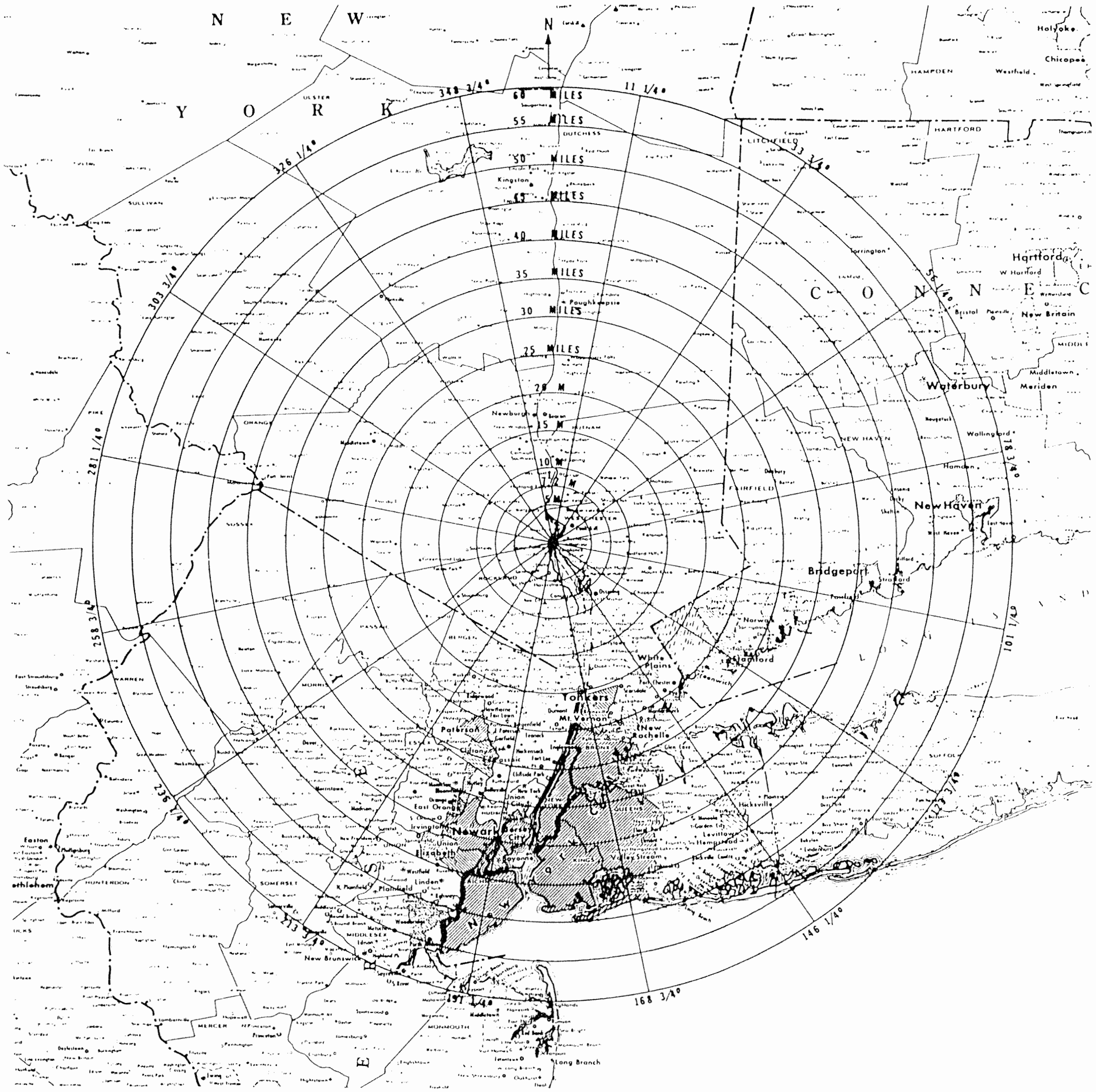
INDIAN POINT 3		FSAR UPDATE
PROPERTY BOUNDRY INDIAN POINT 3 NUCLEAR POWER PLANT		
REV. 1	OCT 2003	FIGURE NO. 2.2-4



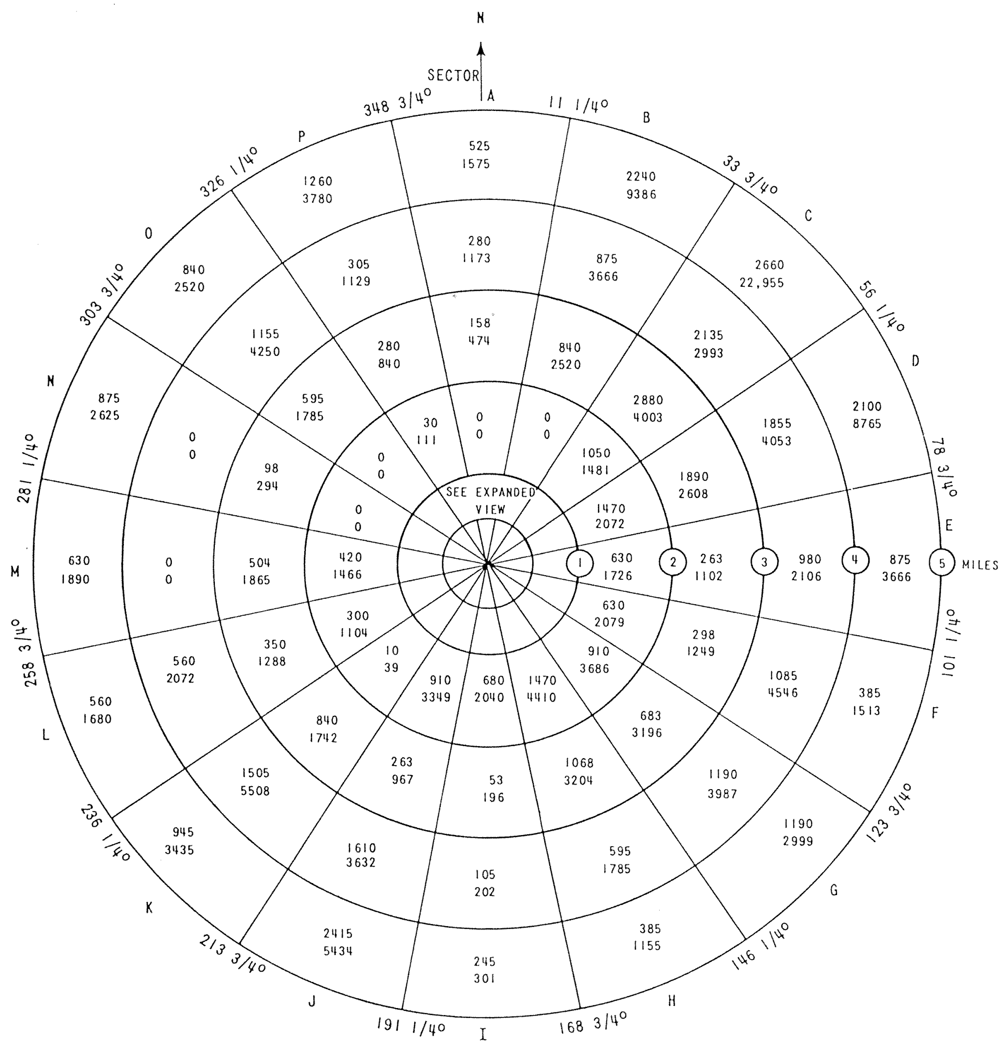
INDIAN POINT 3	FSAR UPDATE

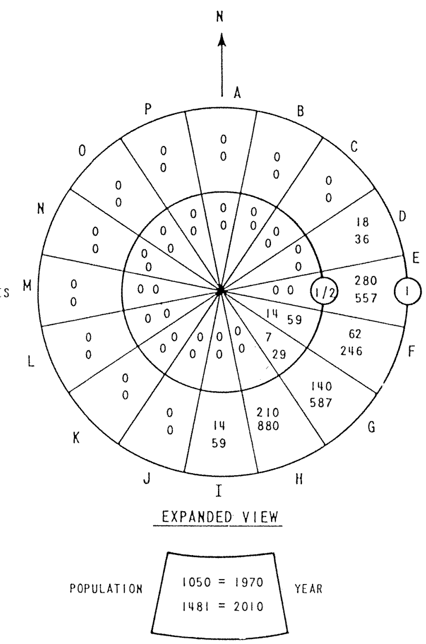
POPULATION DISTRIBUTION
FOR 55 MILE RADIUS
MAY 1970 REPORT

REV. 0	JULY, 1982	FIGURE NO. 2.4-1



INDIAN POINT 3	FSAR UPDATE
POPULATION GRID 60 MILE RADIUS MAY 1970 REPORT	
REV. 0 JULY, 1982	FIGURE NO. 2.4-2

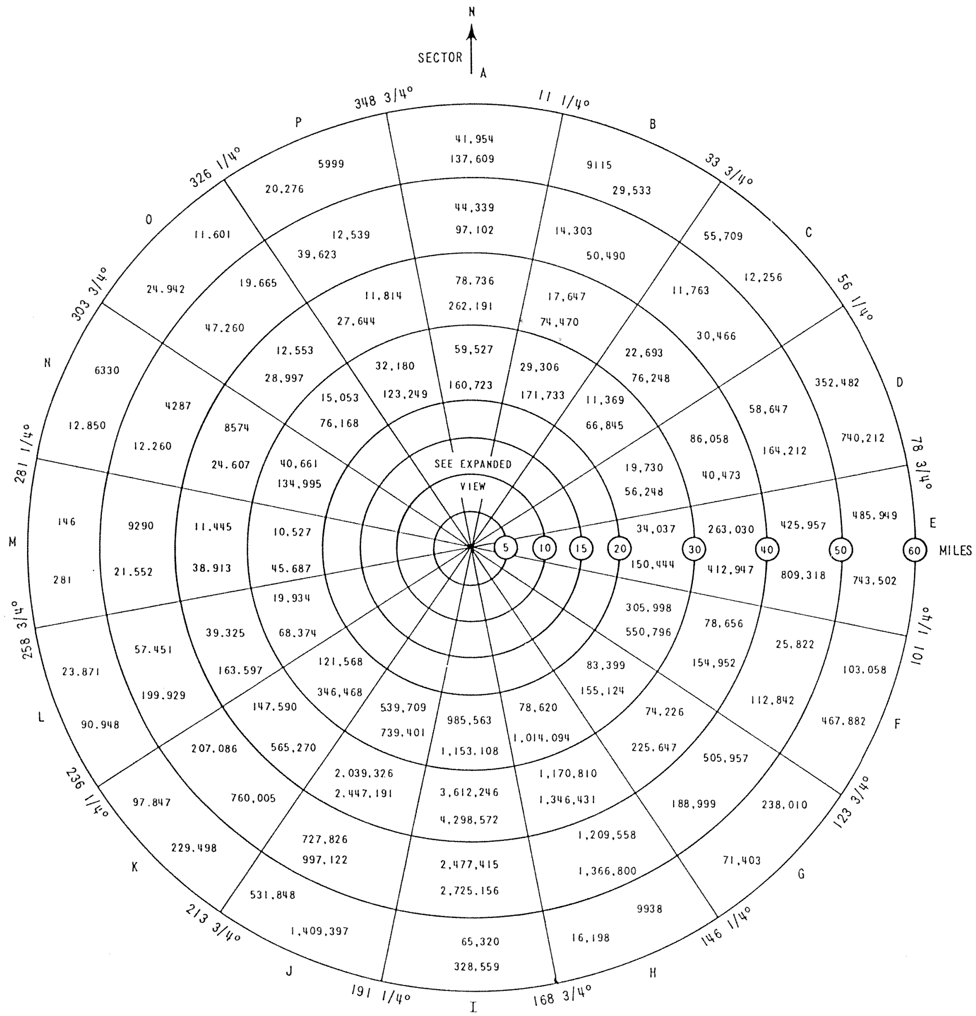


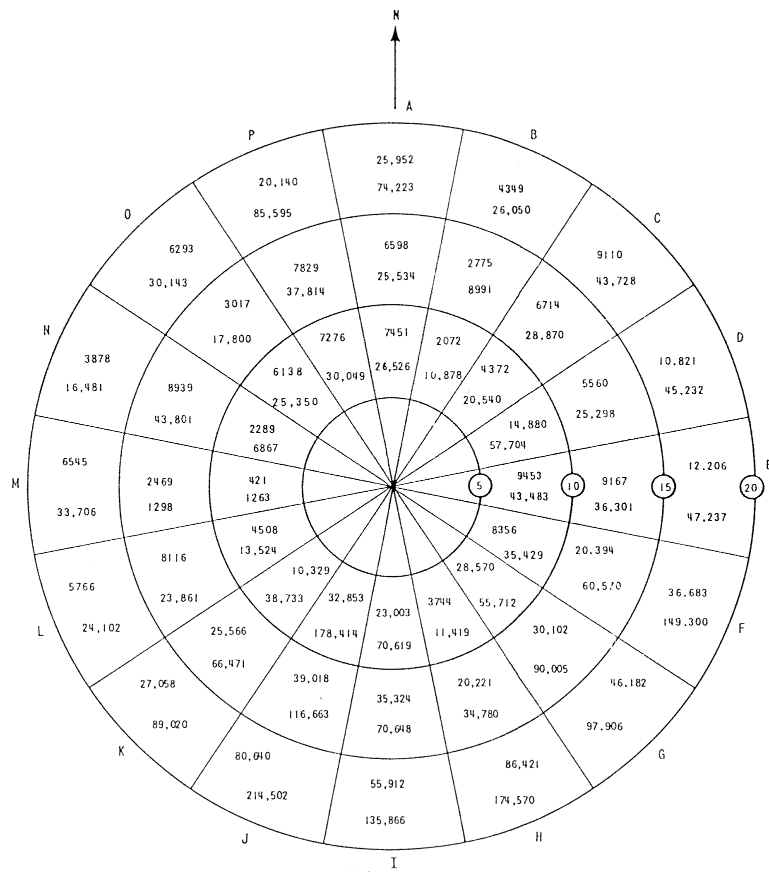


INDIAN POINT 3 — FSAR UPDATE

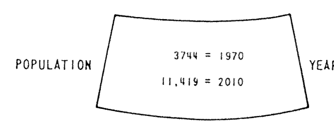
POPULATION DISTRIBUTION
5 MILE RADIUS
BASED ON JUNE 1972 REPORT

REV. 0 — JULY, 1982 — FIGURE NO. 2.4-3

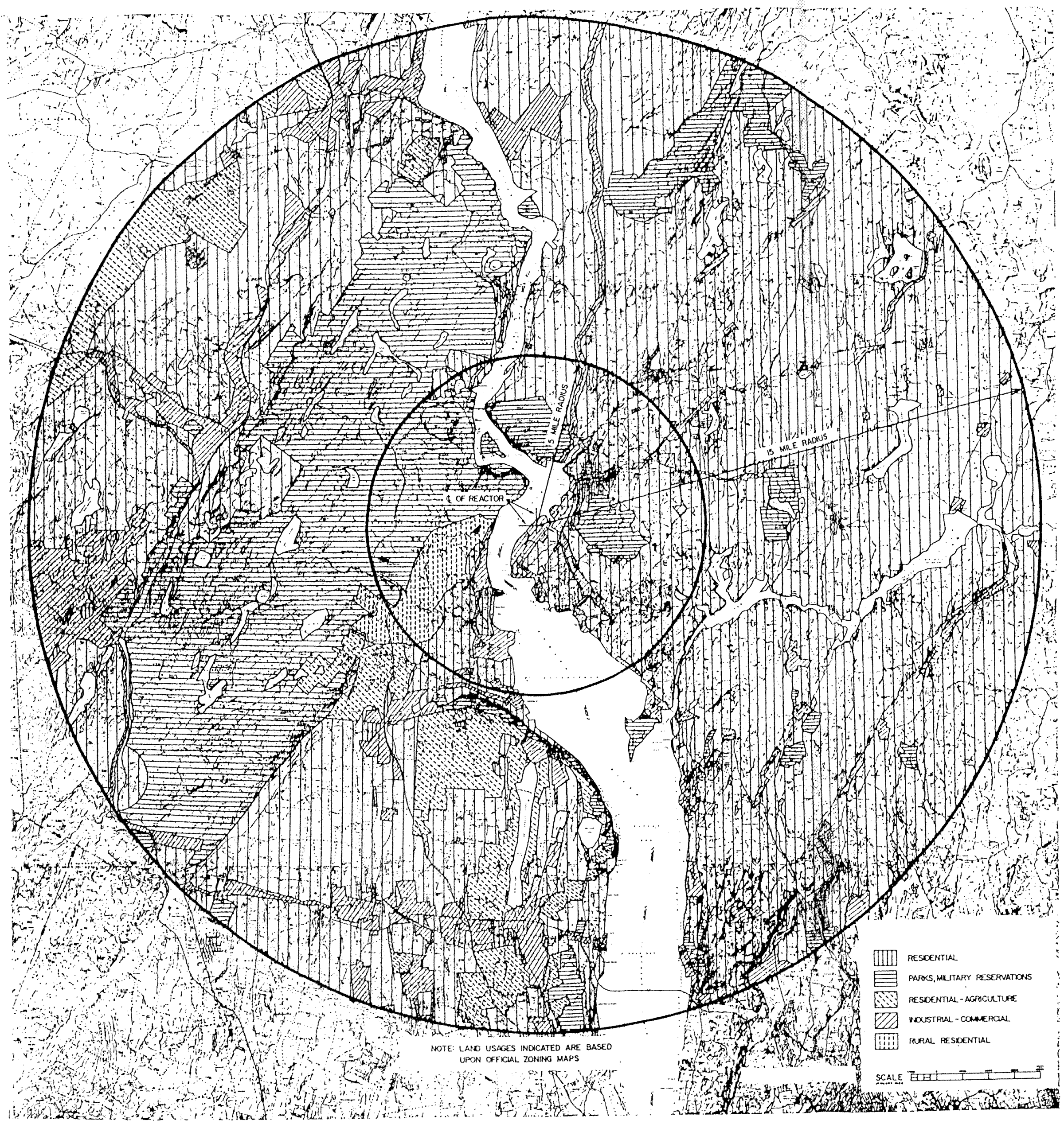




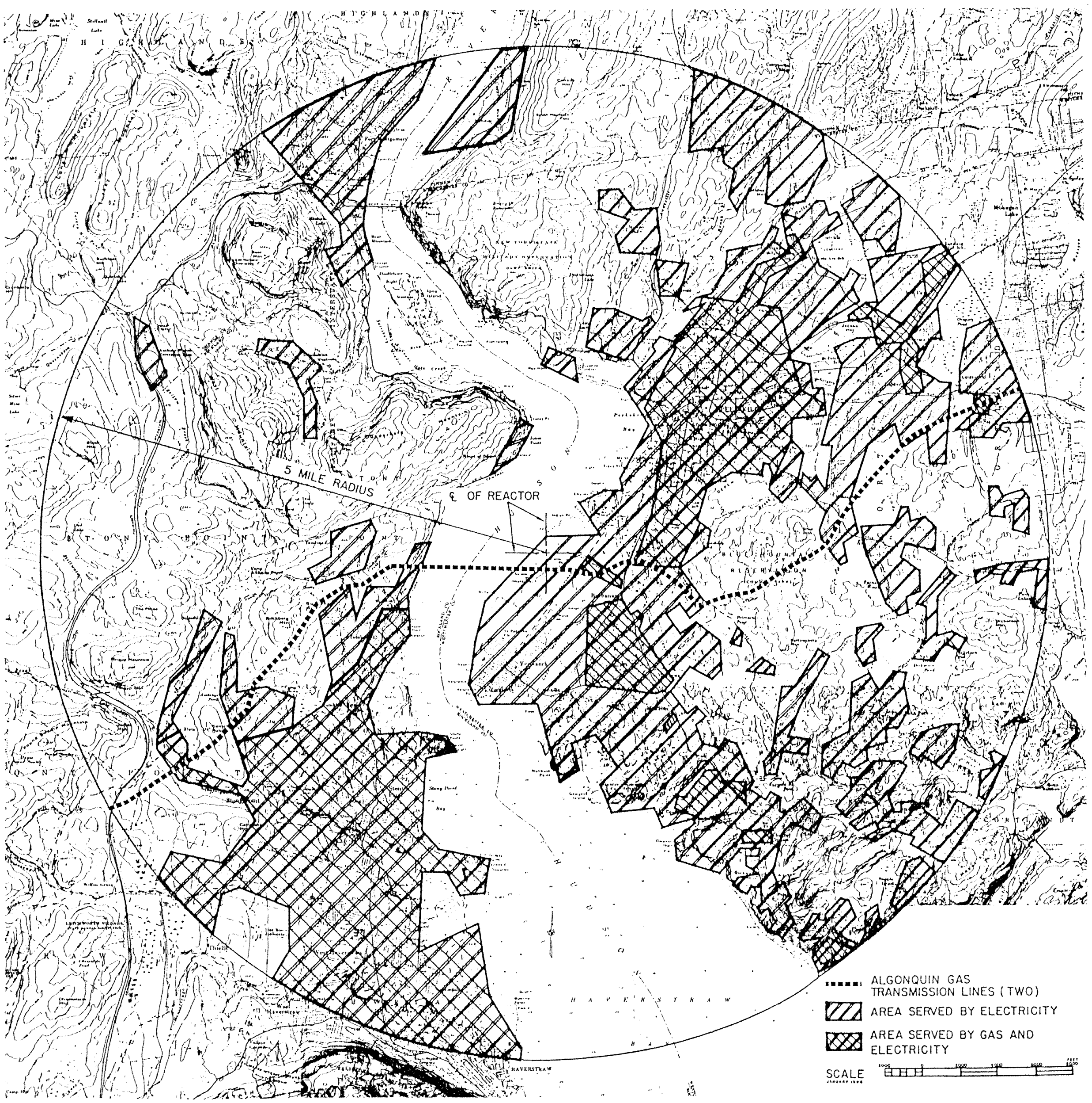
EXPANDED VIEW



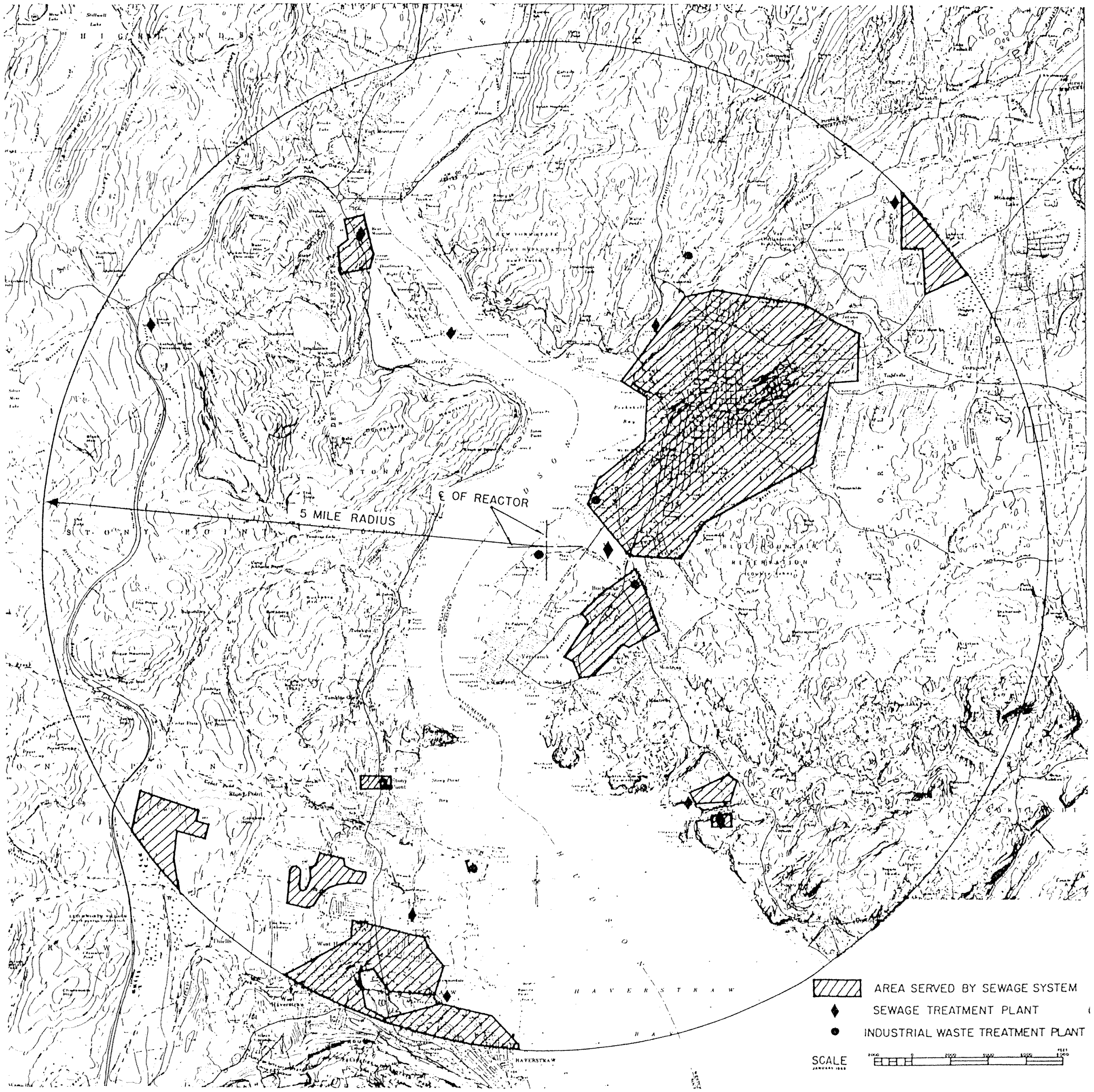
INDIAN POINT 3	FSAR UPDATE
POPULATION DISTRIBUTION 60 MILE RADIUS (1970, 2010) JULY 1972 REPORT	
REV 0 JULY, 1982	FIGURE NO. 2.4-4



INDIAN POINT 3		FSAR UPDATE
LAND USAGE (15 MILE RADIUS)		
REV. 0	JULY, 1982	FIGURE NO. 2.4-6



INDIAN POINT 3		FSAR UPDATE
PUBLIC UTILITIES		
REV. 0	JULY, 1982	FIGURE NO. 2.4-7



INDIAN POINT 3		FSAR UPDATE
SEWAGE SYSTEMS		
REV. 0	JULY, 1982	FIGURE NO. 2.4-8

CHAPTER 3

REACTOR

3.1 DESIGN BASIS

3.1.1 Performance Objectives

The reactor thermal power analyzed is 3216 MWt.

The fuel rod cladding was designed to maintain its integrity for the anticipated fuel assembly life. The effects of gas release, fuel dimensional changes, and corrosion-induced and irradiation-induced changes in the mechanical properties of cladding were considered in the design of the fuel assemblies.

Rod Control Clusters are employed to provide sufficient reactivity control to terminate any credible power transient prior to reaching the design minimum departure from nucleate boiling ratio (DNBR) of the applicable limit. This is accomplished by ensuring sufficient control cluster worth to shut the reactor down by at least 1.3% in the hot condition with the most reactive control cluster 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.

In addition, the control rod worth in conjunction with the boric acid injection from the refueling water storage tank (RWST) is sufficient to prevent an unacceptable return to power level as a result of the maximum credible steam line break (one safety valve stuck fully open) even assuming that the most reactive control rod is fully withdrawn.

With the BIT functionally eliminated, the return to power following a credible steamline break accident has been evaluated showing that the event is bounded by the hypothetical steamline break. The departure from nucleate boiling (DNB) design basis is met with no consequential fuel failures predicted, and assuring that the return to power remains within the limits established for the protection of the health and safety of the public, with margin.

Plant specific analyses performed by Westinghouse for Indian point Unit 3, have shown that the Boron Injection Tank (BIT) may be bypassed, eliminated, or the concentration of its contents reduced, while continuing to meet applicable safety criteria.

The functional elimination of the BIT replaces the concentrated boric acid contained therein, with water from the Refueling Water Storage Tank (RWST); this obviates the need to maintain the BIT and its associated piping at elevated temperatures.

The lowering of the minimum required boric acid concentration in the BIT:

1) reduces the potential for degradation of carbon steel components and supports as a result of leakage;
2) eliminates the need to maintain recirculation of boric acid through BIT;
3) eliminates the need to maintain the BIT heaters and heat tracing on the associated SIS piping and recirculation lines; and
4) eliminates the need for periodic checks of BIT concentration thereby reducing radiation exposure of plant personnel.

5) Eliminates the need to maintain closed the BIT inlet and outlet isolation valves.

Experimental measurements from critical experiments or operating reactors, or both, were used to validate the methods employed in the design. During design, nuclear parameters were calculated for every phase of operation of the first core and reload cycles and, where applicable, were compared with design limits to show that an adequate margin of safety existed. In the thermal hydraulic design of the core, the maximum fuel and clad temperatures during normal reactor operation and at overpower conditions were conservatively evaluated and found to be consistent with safe operating limitations.

3.1.2 Principal Design Criteria

The General Design Criteria presented and discussed in this section are those which were in effect at the time when Indian Point 3 was designed and constructed. These general design criteria, which formed the bases for the Indian Point 3 design, were published by the Atomic Energy Commission in the Federal Register of July 11, 1967, and subsequently made a part of 10 CFR 50.

A study of compliance with 10 CFR Parts 20 and 50 in accordance with some of the provisions of the Commission's Confirmatory Order of February 11, 1980 has been completed. The detailed results of the evaluation of compliance of Indian Point 3 with the General Design Criteria presently established by the Nuclear Regulatory Commission (NRC) in 10 CFR 50 Appendix A, were submitted to the NRC on August 11, 1980, and approved by the Commission on January 19, 1982. These results are presented in Section 1.3.

Reactor Core Design

Criterion 6: The reactor 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.

The reactor core, with its related control and protection system, was 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, provide 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 Control and Protection System was 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 limit.

The integrity of fuel cladding is ensured by preventing excessive clad heating and 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:

1) Minimum DNB ratio equal to or greater than the applicable limit

2) Fuel center temperature below 4700° F

3) The internal gas pressure of the lead rod in the reactor is limited to a value below that which would cause (1) the diametral gap to increase due to outward clad creep during steady-state operation, and (2) extensive DNB propagation to occur

4) Clad stresses less than the Zircaloy or ZIRLO™ yield strength

5) Clad strain less than 1%

The ability of fuel designed and operated to these criteria to withstand postulated normal and abnormal service conditions is shown by the 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.

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 results in a high pressurizer pressure trip and thereby prevents fuel damage for this transient. A loss of external electrical load of 50% of full power or less is normally controlled by rod cluster insertion together with a controlled stream 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 limit 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. Protection circuits actuated directly by the coolant conditions identified with core limits are therefore effective in preventing core damage.

Suppression of Reactor Power Oscillations

Criterion 7: 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.

The potential for possible spatial oscillations of power distribution for this core has been reviewed. It was concluded that low frequency xenon oscillations may occur in the axial dimension, and the control rods can suppress these oscillations. The core is expected to be stable to xenon oscillations in the X-Y dimension. Excore 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. (In-core instrumentation is used to periodically calibrate and verify the information provided by the Excore instrumentation.) The analysis, detection and control of these oscillations is discussed in Reference 2 of Section 3.2.

Redundancy of Reactivity Control

Criterion 27: Two independent reactivity control systems, preferably of different principles, shall be provided.

Two independent reactivity control systems are provided, one involving rod cluster control (RCC) assemblies and the other involving chemical shimming.

Reactivity Hot Shutdown Capability

Criterion 28: The reactivity control systems provided shall be capable of making and holding the core subcritical from any hot standby or hot operating condition.

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 Rod Cluster Control (RCC) assemblies are divided into two categories comprising 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 RCC 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.

Reactivity Shutdown Capability

Criterion 29: 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.

The reactor core, together with the reactor control and protection system was designed so that the minimum allowable DNBR is at least the applicable limit and there is no fuel melting during normal operation including anticipated transients.

The shutdown groups are provided to supplement the control groups RCC assemblies to make the reactor at least 1.3% subcritical at the hot zero power condition following trip from any credible operating condition assuming the most reactive RCC assembly is in the fully withdrawn position.

Sufficient shutdown capability is also provided to prevent an unacceptable return to power level, 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 stream bypass, or relief valve, or safety valve stuck open. This is achieved by the combination of control rods and automatic boric acid addition via the Emergency Core Cooling System. With the BIT functionally eliminated, the return to power following a credible steamline break accident has been evaluated showing that the event is bounded by the hypothetical steamline break. The departure from nucleate boiling (DNB) design basis is met with no consequential fuel failures predicted, and assuring that the return to power remains within the limits established for the protection of the health and safety of the public, with margin.

The minimum shutdown margin was calculated to be at least 1.3% @EOL conditions assuming the maximum worth control rod in the fully withdrawn position allowing 10% uncertainty in the control rod calculations.

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.

Reactivity Holddown Capability

Criterion 30: 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 that there will be no undue risk to the health and safety of the public.

Normal reactivity shutdown capability is provided within 2.7 seconds following a trip signal by control rods, 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 prevents return to critical 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 inject 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. Using either one of the two boric acid transfer pumps, in conjunction with any one of the three charging pumps, the RCS can be borated to hot shutdown even with the control rods fully withdrawn. Additional boration would be used to compensate for xenon decay. At a minimum CVCS design boration rate of 132 ppm/hr, the boron concentration required for cold shutdown can be reached well before xenon decays below its pre-shutdown level. The RWST is a suitable backup source for emergency boration. When two charging pumps are used to transfer borated water from the RWST to the reactor coolant, the boron concentration required for cold shutdown can be reached before xenon decays below its full power pre-shutdown level.

On the basis of the above, the injection of boric acid is shown to afford backup reactivity shutdown capability, independent of control rod clusters 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 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.

With the BIT functionally eliminated, the return to power following a credible steamline break accident has been evaluated showing that the event is bounded by the hypothetical steamline break. The departure from nucleate boiling (DNB) design basis is met with no consequential fuel failures predicted, and assuring that the return to power remains within the limits established for the protection of the health and safety of the public, with margin.

Reactivity Control System Malfunction

Criterion 31: 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, buy limiting reactivity transients to avoid exceeding acceptable fuel damage limits.

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.

Maximum Reactivity Worth of Control Rods

Criterion 32: 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.

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 rods 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 yielding reactivity insertion rates no greater than 66 pcm/sec which is well within the capability of the overpower-overtemperature protection circuits to prevent core damage.

The rod drive design permits only two groups to be withdrawn at the same time. Should more than two groups try to move, insufficient power is available to activate the movable holding latches and the affected control rods would disengage and fall.

Specifically, a bank contains either one or two groups. Within a bank the group may be moved only sequentially one step at a time. Two banks may be moved simultaneously, e.g. banks C and D. However, group 1 in bank D may be moved together (one step), then group 2 in each bank simultaneously (one step). Therefore, no more than two groups can be moved together and this forms the basis of the assumption.

No single credible mechanical or electrical control system malfunction can cause a rod cluster to be withdrawn at a speed greater than 80 steps per minute (~50 inches per minute).

3.1.3 Safety Limits

The reactor is capable of meeting the performance objectives throughout core life under both steady state and transient 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, control, thermal and hydraulic, and mechanical aspects of the design.

Nuclear Limits

At full power (license application power) the nuclear heat flux hot channel factor F_Q^N, is not exceeded. For any condition of power level, coolant temperature and pressure which is permitted by the control and protection system during normal operation and anticipated transients the hot channel power distribution is such that the minimum DNB ratio is greater than the applicable limit and the linear heat rate is less than 22.7 kW/ft. For any normal steady state operating condition, the maximum linear heat rate does not exceed 6.64 x F_Q kW ft, where F_Q is the maximum value dictated by the Core Operating Limits Report (COLR).

Potential axial xenon oscillations are controlled with the control rods to preclude adverse core conditions. The protection system ensures that the nuclear core limits are not exceeded.

Fuel Enrichment Limits

Detailed nuclear analysis (refer to Reference 65 of Section 3.2) has been completed to demonstrate that the existing spent fuel storage racks can safely store fuel with initial enrichments up to 5.0 w/o U-235 provided they are done so as specified in Figures 9.5-2A, 9.5-2B, and 9.5-2C.

Control Bank Insertion Limits

The control bank insertion limits for D, C and B control banks were originally revised for Cycle 6 operation. The associated change in control bank insertion limits (refer to Reference 57 of Section 3.2) results in increased flexibility in core design, and a reduction in the calculated core peaking factor Fq at the bank insertion limit. The revised insertion limit curve for the current cycle is provided in the cycle-specific Core Operating Limits Report.

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 as shown in the Technical Specifications is available assuming a 10% uncertainty in the control rod calculation (see Table 3.2-3).
2) This shutdown margin is maintained with the most reactive RCCA in the fully withdrawn position.
3) The shutdown margin is maintained at ambient temperature by the use of soluble poison.

Thermal and Hydraulic Limits

The reactor core was designed to meet the following limiting thermal and hydraulic criteria:

1) The minimum allowable DNBR during normal operation, including anticipated transients for the WRB-1 correlation is documented in Table 14.1-0. The minimum allowable DNBR for the W-3 correlation is 1.3 from 1000 to 2400 psia and 1.45 from 500 to 1000 psia.

2) Fuel temperature will not exceed 4700° F during any anticipated operating condition.

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 causes 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 W-3 or WRB-1 correlation, to the existing heat flux at the same core location is the DNB ratio. The limiting DNB ratio corresponds to a 95% probability at a 95% confidence level that DNB does not occur and is chosen to maintain an appropriate margin to DNB for all operating conditions.

In connection with the functional elimination of the BIT, two hypothetical and one credible steamline break cases were either analyzed or evaluated:

Hypothetical steamline breaks:

1) Steampipe severance, downstream of the flow restrictor, with offsite power available;
2) Steampipe severance, downstream of the flow restrictor, without offsite power available;

Credible streamline break:

1) A failed secondary safety or relief valve, with offsite power available.

The DNB analyses show that the DNB design basis is met for the hypothetical steamline break with offsite power, and that no consequential fuel failures are anticipated. The hypothetical break without offsite power and the credible steamline break were evaluated and determined to be bounded by the results of the hypothetical steamline break with offsite power.

Mechanical Limits

Reactor Internals

The reactor internal components were 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 were 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.

Fuel Assemblies

The fuel assemblies were 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 assembly. The assembly was 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 seven locations along their length within the fuel assemblies by brazed grid assemblies which were designed to maintain control of the lateral spacing between the rods throughout the design life of the assemblies. The magnitude of the support loads provided by the grids are 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.

The fuel rod cladding was designed to withstand operating pressure loads wtihout rupture and to maintain fuel integrity throughout design life.

Cycle 7 was the first cycle where Vantage 5 fuel was utilized. Features of this fuel include: debris filter bottom nozzle, reconstitutable top nozzle, six inch natural uranium axial blankets and the optimal use of integral fuel burnable absorber (IFBA) which incorporates a thin coating of ZrB2 on the outside cylindrical surface area of the fuel pellet.

Cycle 10 was the first cycle where Vantage + fuel was utilized. Features of this fuel included three intermediate flow mixers, enriched annular axial blankets, and low pressure drop midgrids, ZIRLO guide thimbles and midgrids. ZIRLO as a clad material was introduced in Cycle 9. Vantage + fuel also contains various high burnup features.

Beginning with Cycle 14, “15x15 upgrade” fuel was introduced to the core. See section 3.2.5.5 for details.

Rod Cluster Control Assemblies

The criteria used for the design of the cladding on the individual absorber rods in the rod cluster control assemblies (RCCA) are similar to those used for the fuel rod cladding. The cladding was 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 RCCA cladding thickness. Cladding of RCCA's is stainless steel.

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.

Control Rod Drive Assembly

Each control rod drive assembly was designed as a hermetically sealed unit to prevent leakage of reactor coolant. All pressure containing components were 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 RCCA which results in a rapid shutdown of the reactor for conditions that cannot be handled by the reactor control system.

The part length control rod drive mechanisms (Reference 46) were used to position the part length rod control cluster assemblies within the reactor core. Subsequent to initial plant operation (during the Cycle 1 to Cycle 2 refueling outage), the part length RCCAs were removed from the reactor. Their drive mechanisms and position indication system were retired in place. Thimble plug devices were placed in the fuel assemblies where the part length rods had been.

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. Throughout this section, Cycle 1 and Current Cycle parameters have been utilized in as much as they represent examples and/or somewhat typical values for initial and subsequent cycles. 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 for Cycle 1 and the current cycle is presented in Table 3.2-1. Figures 3.2-2 through 3.2-12 provide core configuration information for Cycle 1 and are included for historical purposes. For current cycle information refer to the cycle specific Nuclear Parameters and Operations Package and the cycle specific Core Operating Limits Report.

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 change in temperature of the reactor coolant from cold shutdown to the hot operating, zero power condition; (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; (4) reactivity changes associated with the power coefficient of reactivity.

Chemical Shim Control

Control to render the reactor subcritical at temperatures below the operating range is provided by a chemical neutron absorber (boron). The boron concentration during refueling is the more restrictive of 2050 ppm or that required to be 5% subcritical. The boron concentration and the control rods together are required to provide at least 5 percent shutdown margin for these operations. Boron concentration requirements for refueling and Plant Startup (BOL) are found in Reference 72.

These boron concentrations are well within solubility limits at ambient temperature. Refueling concentration is also maintained in the spent fuel pit since it is directly connected with the refueling canal during refueling operations.

The initial Cycle 1, full power boron concentration without equilibrium xenon and samarium was 1228 ppm. As fission product poisons were built up, the boron concentration was reduced to 899 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 669 ppm. This concentration is less than the full power operating value with equilibrium xenon.

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 was also specified to provide adequate shutdown with the most reactive rod stuck out of the core.

Control rod reactivity requirements at beginning- and end-of-life for Cycle 1 and the current cycle are summarized in Table 3.2-2. The installed worth of the control rods is shown in Table 3.2-3

The difference between required and installed worth is available for excess shutdown upon reactor trip. The control rod requirements are discussed below.

Total Power Reactivity Defect

Control rods must be available to compensate for the reactivity change in incurred with a change in power level due to the Doppler and moderator temperature effects.

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 moderator temperature coefficient range is given in Table 3.2-1, line 42, while the cumulative reactivity change 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.

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.

The fully withdrawn bank position can vary within a few steps from the reference fully withdrawn condition from cycle to cycle.

The greatest depth in the core to which the control banks may be inserted is known as the insertion limit. The maximum insertion limit on which the FSAR transient analyses are based is 23.5 percent D-Bank insertion, which corresponds to a D-Bank position of 176 steps at 100 percent power. The actual limit may be administratively decreased via a COLR change, to facilitate core design without having to reanalyze the FSAR transients.

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 must be inserted a given distance into the core. The reactivity worth of this insertion has been defined as control rod bite.

The reactivity insertion rate must be sufficient to compensate for reactivity variation due to changes in power and temperature caused either by a ramp load change of five percent per minute, or by a step load change of ten percent. An insertion rate of $4 \times 10^{-5}\ \Delta p$ per second is determined by the transient analysis of the core and plant to be adequate for the most adverse combinations of power and moderator coefficients. To obtain this minimum ramp rate, one control bank of rods had to remain inserted at least 13 percent into the core at the Cycle 1 beginning-of-life. The reactivity associated with this bite was 0.03 percent.

Indian Point 3 is analyzed for continuous operation at the bite position, at no bite position (i.e., ARO) and for any rod position in between (85).

Xenon Stability Control

The control rods are capable of suppressing xenon induced power oscillations in the axial direction, should they occur. Out-of-core instrumentation was 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.(2) A full discussion of axial xenon stability control can be found in Reference 3.

Excess Reactivity Insertion Upon Reactor Trip

The control requirements are nominally based on providing 1.3 percent shutdown at hot, zero power conditions with the highest worth rod stuck in its fully withdrawn position. The condition where excess reactivity insertion is most critical is at the end of a cycle when the steam line break accident is considered. The excess control available at the Cycle 1 end-of-cycle, hot zero power condition with the highest worth rod stuck out, allowing a 10% margin for uncertainty in control rod worth is shown in Table 3.2-3.

Calculated Rod Worths

The complement of 53 full length control rods, arranged in the pattern shown in Figure 3.2-1 meets the shutdown requirements. Table 3.2-3 lists the calculated worths of this rod configuration for beginning-of-life, and end-of life, for Cycles 1 and the current cycle.

In order to be sure of maintaining a conservative margin between calculated and required rod worths, an additional amount has been added to account for uncertainties in the control rod worth calculations. The calculated reactivity worths listed were decreased in the design by 10 percent to account for any errors or uncertainties in the calculation. This worth was 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 operating reactors shows the calculation to be well within the allowed uncertainty of 10%.

Reactor Core Power Distribution

The accuracy of power distribution calculations was initially confirmed through approximately one thousand flux maps during some twenty years of reactor operation under conditions very similar to those which were expected for Indian Point 3. Details of this confirmation are given in Reference 5.

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 which 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). 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 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 clad.

Combined statistically, the net effect of the hot channel factors has a value of 1.03 to be applied to fuel rod surface heat flux.

$F_{\Delta H}^{N}$, 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 Departure from Nucleate Boiling Ratio.

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
= (Maximum kW/ft) / (Average kW/ft)

without densification effects

$$F_Q = F_Q^N \times F_Q^E$$
$$= F_{xy}^N \times F_Z^N \times F_U^N \times F_Q^E$$

where:

F_Q^N and F_Q^E are defined above.

F_U^N = factor for conservatism, assumed to be 1.05.

F_{XY}^N = ratio of peak power density to average power density in the horizontal plans of peak local power.

F_Z^N = 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 is the core average axial peaking factor.

To include the allowances 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

Then:

F_Q = Total peaking factor

= (Maximum kW/ft) / (Average kW/ft)

Including densification allowance,

$$F_Q = \max_{on\ z}\left\{\left[F_{XY}^{N}(Z)\right]\left[P(Z)\right]\left[S(Z)\right]\right\}\left(F_U^N\right)\left(F_Q^E\right)$$

Results reported in Reference 83 show that current fuel designs manufactured by Westinghouse are highly stable with respect to fuel densification and that the formation of small gaps in the axial pellet columns are very infrequent events. Reference 83 concludes that a densification power spike factor (S(Z)) of 1.0 is appropriate for current Westinghouse fuel, effectively eliminating this penalty from the safety analysis.

Radial Power Distributions

The power shape in horizontal sections of the core at full power is a function of the fuel and burnable poison 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 are considered also, but they are quite small. The effect of non-uniform 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. Figure 3.2-2 through 3.2-4 show representative Cycle 1 radial power distributions for one eighth of the core for representative operating conditions. The conditions are:

1) Hot Full Power (HFP) – Beginning-of-Life (BOL) – unrodded – no xenon.
2) HFP – End-of-Life (EOL) – unrodded – equilibrium xenon
3) HFP – BOL – Bank D in – equilibrium xenon

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.

Axial Power Distributions

The shape of the power profile in the axial or vertical directions is largely under the control of the operator through either the manual operation of the control rods or automatic motion of the 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 excore 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, ∆I. Calculations of the core average peaking factor for many plants, and measurements from operating plants under many operating situations, are associated with either I 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:

Axial Offset = $\left(\Phi_t - \Phi_b\right)/\left(\Phi_t + \Phi_b\right)$

where: Φ_t and Φ_b are the top and bottom detector readings.

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 83 show that current fuel designs manufactured by Westinghouse are highly stable with respect to fuel densification and that the formation of small gaps in the axial pellet columns are very infrequent events. Reference 83 concludes that a densification power spike factor (S(Z)) of 1.0 is appropriate for current Westinghouse fuel, effectively eliminating this penalty from the safety analysis.

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 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 PWR's is included in Reference 8. 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.[9] The following paragraphs summarize these reports [8,9] 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 in these calculations. 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 on the limits of interest can be set in terms of the plant parameters which are readily observed. 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 by more than the rod group alignment limits specified in the Technical Specifications.
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 changed from about +10 to –3 percent linearly through the life of the cycle. This minimized xenon transient effects on the axial power distribution, since the procedures essentially kept 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 9 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, burnups, 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 9. The calculated values have been increased by a factor of 1.05 for conservatism, and by a factor of 1.03 for the engineering factor F_Q^E.

The envelope drawn over the calculated (max F_Q x Power) points in the COLR represents an upper bound envelope on local power density versus elevation in the core. 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, s 3.2-5 are 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. These procedures are detailed in the Technical Specifications and are followed by relying only upon excore surveillance supplemented by the normal full core map requirement at every effective full power month, and by computer based alarms or manual logging 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 below:

1) Control rods in a single bank move together with no individual rod insertion differing by more than the rod group alignment limits specified in the Technical Specifications.
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 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 offset, 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 mispositions 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 an appropriate allowance for calorimetric error. It should be noted that a reactor overpower accident is not assumed to occur coincident with an independent operator error.

Analyses of possible operating power shapes show that the appropriate hot channel factors F_Q and $F_{\Delta H}^{N}$ for peak local power density and for DNB analysis at full power are the values addressed in the COLR.

F_Q can be increased with decreasing power as shown in the Technical Specifications. Increasing $F_{\Delta H}^{N}$ 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, provided the above conditions 1) through 4) are observed, the COLR limits are met.

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.

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 Systems.

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 poison 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 burnable poison for Cycle 1 was in the form of borated Pyrex glass rods clad in stainless steel. There were 1434 of these borated Pyrex glass rods in the form of clusters distributed throughout the initial core in vacant rod cluster control guide tubes, as illustrated in Figures, 3.2-6 through 3.2-7. Information regarding research, development and nuclear evaluation of the burnable poison rods can be found in Reference 1. These rods initially controlled 10% $\Delta\rho$ of the installed excess reactivity and their insertion into the core resulted in a reduction of the initial hot zero power boron concentration in the coolant to 1330 ppm. The moderator temperature coefficient is negative at operating conditions with burnable poison rods installed. Subsequent cycles utilized B_4C in AL_20_3 in wet annular burnable absorbers (WABA) and ZrB_2 in Integral Fuel Burnable Absorbers (IFBA) and also the already mentioned Pyrex rods.

The effect of burnup on the moderator temperature coefficient is periodically 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. The calculated net effect and the predicted moderator temperature coefficient at equilibrium xenon for Cycle 1 and at full power BOL was – 0.84 x 10-4/°F. With core burnup, the coefficient became more negative as boron was removed because of a shift in the neutron energy spectrum due to the buildup of plutonium and fission products. At Cycle 1 EOL with no boron or rods in the core, the moderator coefficient was -3.5×10^{-4}/°F. Reference 72 provides the Cycle 16 moderator coefficient.

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 half-degree change in moderator temperature. The calculated Cycle 1 BOL and EOL pressure coefficients are specified in Table 3.2-1, Line 43.

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 from BOL and EOL is specified in Table 3.2-1, Line 44.

Doppler and Power Coefficients

The Doppler coefficient is defined as the change in neutron multiplication* per degree change in fuel temperature. The coefficient was obtained in the past by calculating neutron multiplication as a function of effective fuel temperature by the code LEOPARD.(4) The results for Cycle 1 are shown in Figure 3.2-11. More recent calculations of Doppler and power coefficients are performed using the ANC(84) code.

> *NOTE: Neutron multiplication is defined at the ratio of the average number of neutrons produced by fission in each generation to the total number of corresponding neutrons absorbed.

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. Figure 3.2-12 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.

3.2.1.2 Nuclear Evaluation

The basis for confidence in the procedures and design methods comes from the comparison of these methods with many experimental results. These experiments include criticals performed at the Westinghouse Reactor Evaluation Center (WREC) and other facilities, and also measured data from operating power reactors. A summary of the results and discussion of the agreement between calculated and measured values is given in other Safety Analysis Reports such as the FSAR for Indian Point 2, Docket No. 50-247.

3.2.1.3 Enrichment Error

Any error in enrichment, beyond the normal manufacturing tolerances, can cause power shapes that are more peaked than those calculated with the correct enrichments. There is an 8% uncertainty margin between the calculated worst value and the design value of power peaking assumed for the analysis of normal steady state operation and anticipated transients. The incore system of moveable flux detectors, used to verify power shapes at start of life, is capable of revealing any enrichment error or loading error which causes power shapes to be peaked in excess of the design value. These power shape measurements are taken at low power when extremely adverse power shapes can be tolerated.

An analysis of the effect of an inadvertent loading of an assembly with an enrichment increased by 20% over the nominal value, showed that the error was detectable at many of the detector locations in the core. In the case of a centrally placed assembly with this enrichment error, five flux detectors would show a signal more than 5% above the expected value. If the assembly bearing the enrichment error is placed off-center and as far from a flux detector as possible, the tilt caused by a 20% error in enrichment is detectable in more than half of the detector locations in the core, both as a flux increase over expected symmetric values and as a flux decrease on the opposite side of the core.

If the movable detector system fails to detect an enrichment error, then the power shapes are such that there is margin to the design conditions, and normal plant operation may be safely continued. It is incredible than any positive indication of power shape anomalies which are sufficiently large to cause a significant departure from design conditions, would be ignored.

These measurements are an integral part of the physics startup tests when considerable emphasis is placed on obtaining good power shape measurements.

These considerations, together with the fuel handling procedures described in Section 3.3, preclude power operation in the presence of any significant fuel enrichment error.

3.2.2 Thermal and Hydraulic Design and Evaluation

A large amount of material has been retained in this section as historical background. The thermal and hydraulic design parameters at the stretched power uprate conditions are provided in Table 3.2-4.

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 ration (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 Westinghouse version of the VIPRE-01 (VIPRE) code was used to perform the thermal/hydraulic calculations for both the mini-uprate and stretched power uprate programs. The VIPRE code is equivalent to the THINC-IV (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 (SRE) on WCAP-14565-P-A[(78)].

The design method employed for both fuel types to meet the DNB design basis is the Revised Thermal Design Procedure (RTDP)[(79)]. 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 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 and potential transition core, 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 result in available DNBR margin. The net DNBR margin, after consideration of all penalties, is available for operation and design flexibility. The 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 limit for STDP is 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 the VANTAGE+ fuel assemblies 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) with pressure greater than 1000 psia.

DNB With Physical Burnout

Westinghouse [29] 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 operation were observed.

DNB With Return to Nucleate Boiling

Additional DNB tests have been considered by Westinghouse[30] in 19 and 21 rod bundles. In these tests, DNB without physical burnout was experienced more than once on single rods in the bundle for short periods of time. Each time, a reduction in power of approximately 10% was sufficient to re-establish 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

The interaction of hydrodynamic and spatial effects has been considered and it is concluded that a large margin exists between the design conditions and those for which an instability is possible.

Heated channels in parallel can lead to flow instability. If substantial boiling takes place, periodic flow instabilities have been observed and, as long ago as 1938, Ledinegg[24] proposed a stability criterion on the basis of which the concept of inlet orificing has been developed to stabilize flow. More recent work[25-27] has demonstrated that periodic instabilities are possible which violate the Ledinegg criterion.

In normal flow channels with little or no boiling, the type of instability proposed by Ledinegg is not possible since it results primarily from the large changes in water density along the channel due to boiling. Moreover, the periodic instabilities examined by Quandt[25-26] and Meyer[27] are not exhibited in non-boiling channels of the type found in PWR cores. [28]

3.2.2.1 Thermal and Hydraulic Characteristics of the Design

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 surface temperature of the pellet is affected by the cladding temperature and the thermal conductance of the gap between the pellet and the cladding.

The occurrence of nucleate boiling maintains maximum cladding surface temperature below about 657 °F at nominal system pressure. The contact conductance between the fuel pellet and cladding is a function of the contact pressure and the composition of the gas in the gap (11, 12), and may be calculated by the following equation:

$$h = 0.6P + K/\ f(14.4 \times 10^{-6})$$

where

h = conductance in Btu/hr-ft^2-°F
P = contact pressure in psi
k = the thermal conductivity of the gas mixture in the rod
f = the correction factor for the accommodation coefficient

The thermal conductivity of uranium dioxide was evaluated from published results of work at ORNL [13] Chalk River [14], and WAPD. [15] The design curve for thermal conductivity is given in Figure 3.2-13. The section of the curve at temperatures between 0 °F and 3000 °F is based on the data of Godfrey, et al. [13]

The section of the curve between 3000 °F and 5000 °F was based on two factors:

1) Inpile observations of fuel melting dictate a positive temperature coefficient for conductivity above approximately 3000 °F, The temperature dependence in this range should conform to an exponential curve, since this reflects the most credible physical interpretation of the high temperature conductivity increase.

2) The area under the recommended curve is such that the integral is equal to approximately 97 w/cm as given by Robertson, et at [14] and Duncan. [15] This value is based upon the interpretation of fuel melt radius as determined at Hanford[16] and Chalk River. [14]

Thermal conductivity can be represented best by the following equation:

$$k = (11.8 + 0.0238T)^{-1} + 8.775\times10^{-13}T^3$$

with k in w/cm- °C for 95 percent theoretical density and T in °C.

Based upon the above considerations, the maximum central temperature of the hot pellet at steady state nominal and overpower conditions are shown in Table 3.2-4. The temperature is well below the melting temperatures of the irradiated UO_2 (Refer to Criterion 6 in section 3.1.2), which is taken as the unirradiated fuel temperature of 5080°F[17] decreasing by 58°F per 10,000 MWd/metric ton uranium and covering the manufacturing/modeling uncertainties.

Fuel Thermal Performance

Ability to predict fuel performance at high burnup is based on both published data and proprietary data. Effects of irradiation on UO_2 melting point, fuel swelling, fission gas release, clad creep, clad yield strength, etc., have been incorporated into a computer program to enable the prediction of fuel performance. The proof of performance was determined by the Saxton and Zorita programs. Saxton fuel test rods operated to about 42,000 MWD/MTM peak burnup, and Zorita fuel test rods operated to about 32,000 MWD/MTM, have been used in verifying performance models. In 1972, the Saxton lead rod accumulated additional burnup to about 53,000 MWD/MTM at the end of the Core III operation. Zorita test rods were exposed to about 45,000 MWD/MTM at the end of second cycle operation in mid 1972. In addition, special removable rod assemblies have been irradiated in the Zion, Surry Unit 1 and 2, and Trojan plants. These assemblies have provisions for removable rods which permit non-destructive examination of fuel rods during refueling shutdowns. Two Zion 15 x 15 high burnup test assemblies have been irradiated for five fuel cycles and have achieved assembly burnups of about 55,000 MWD/MTM. Profilometry measurements after cycle Nos. 1, 2, 3, and 4 have shown less fuel swelling and outward cladding strain than predicted by Westinghouse models. Examination of removable rod 17 x 17 assemblies irradiated in Surry Units 1 and 2 for up to four cycles have shown that cladding creep down is somewhat less than predicted by Westinghouse models.

Applying best estimate models to Indian Point 3 Region III peak burnup rod, the cladding strain damage limit was calculated to be reached at a power level greater than 21 kW/ft.

In the calculation of the steady-state performance of a nuclear fuel rod, the following interacting factors were considered:

1) Cladding creep and elastic deflection
2) Pellet swelling, thermal expansion, gas release, and thermal properties as a function of temperature
3) Internal pressure as a function of fission gas release, rod geometry, and temperature distribution.

These effects have been combined in a computer code which is considered to be Westinghouse proprietary information. With these interacting factors considered, the code determines the fuel rod performance characteristics for a given rod geometry, power history, and axial power shape. In particular, internal gas pressure, fuel and cladding temperatures, and cladding deflections are calculated. The fuel rod is divided lengthwise into several sections and radially into a number of annular zones. Fuel density changes, cladding stresses, strains and deformations, and fission gas releases are calculated separately for each segment. The effects are integrated to obtain the internal rod pressure.

The gap conductance between the pellet surface and the cladding inner diameter is calculated as a function of the composition and pressure of the gas mixture, and the contact pressure between clad and pellet. After computing the fuel temperature for each pellet's annular zone, the fractional fission gas release is assessed from the diffusion-trapping model described by

Weisman, et al.[31] The total amount of gas released is based on the average fractional release within each axial increment and the gas generation rate which in turn is a function of burnup. Finally, the gas released is summed over all axial increments and the pressure is calculated.

The code shows good agreement in fit for a variety of published and proprietary data on fission gas release, fuel temperatures and clad deflections. Included in this spectrum are variations in power, time, fuel density, and geometry.

The worst strain conditions in the fuel rod occur at high burnup after the clad and fuel have reached mechanical equilibrium (cladding and fuel in intimate contact). At this time in life, the tolerances have negligible effect and therefore the linear heat rating is calculated using nominal dimensions.

The new PAD fuel rod performance code used to analyze VANTAGE + reloads contained an improved gap conductance model that calculates lower fuel temperatures. The new PAD code was approved by the NRC[80]. Fuel rod thermal evaluations are performed at rated power, maximum overpower, and during transients at various burnups for the stretched power uprate program. These analyses assure that design criteria for reactor core design (criterion 6) given in section 3.1.2 are met.

3.2.2.2 Westinghouse Experience With High Power Fuel Rods

A completed high power test program had the objective of defining failure limits for the combined effects of linear heat generation rate and burnup, providing increased assurance that plants have adequate performance and design margins to the fuel failure threshold, and verifying the adequacy of design methods and computer codes. Results from this program are given in Section 8 of Reference 50. Additional information on fuel rod experience is presented in Reference 51. A summary of the comprehensive experimental program to extend the operating experience to higher power and to higher exposures for fuel rods is provided in Figure 3.2-14.

The figure shows that thirty Saxton Plutonium Project non-pressurized fuel rods have operated at a design peak power level of up to 18.5 kW/ft to a peak exposure of approximately 30,000 MWD per MTM [Megawatt days per metric ton of metal (U + PU)]. No failures have occurred with this fuel. In the Saxton overpower test, two selected fuel rods from the Saxton Plutonium Project assemblies were removed after peak exposure of 18,000 MWD/MTM and inserted in a subassembly for short time irradiation at a design rating of 25 kW/ft. Results of this program indicate satisfactory performance of the fuel in every respect. The Saxton Plutonium Project was extended by irradiating approximately 250 rods to peak burnups of about 50,000 MWD/MTM at design linear power levels ranging form 9.5 to 23.6 kW/ft.

In the above tests (performed on non-pressurized rods), the strain fatigue experienced by the cladding was more severe than expected to occur for pressurized rods which would be placed under identical operating conditions.

Internally pressurized fuel rods have been under investigation (19) at Westinghouse for several years. These investigations include out-of-pile and in-pile experimental programs and analytical studies. Fuel rods internally pressurized with various gases have been irradiated in the Saxton reactor. Tests results show that initial pressurization is effective in substantially reducing the rate of cladding-creep on to the UO_2 fuel. The Saxton test results confirm the results of analyses which predict fuel-cladding mechanical interaction early in life for non-pressurized fuel rods and delayed interaction for initially pressurized fuel rods.

To verify the substantial design margin which exists in the fuel rods with regard to excessive internal pressures in a fuel rod, several highly pressurized Zircaloy-clad fuel rods were irradiated for several months in the Saxton reactor, then removed for examination. At an internal pressure of approximately 3500 psia, the fuel operated satisfactorily for the period of the test without any indication of failure. Two fuel rods, deliberately tested at unrealistically high internal pressures, experienced clad cracking but operated satisfactorily for the period of the test. Thus, even with excessive internal pressures that result in clad failures, the test results are favorable.

3.2.2.3 Sizing of Fuel Rod Plenum

The criterion for sizing the fuel rod plenum length was that the internal gas pressure is limited to a value below which could cause: 1) the diametral gap to increase due to outward cladding creep during steady state operation, and 2) extensive DNB propagation to occur. During operational transients, fuel rod clad rupture due to high internal gas pressure is precluded by meeting the above design basis. The end of life internal gas pressure depends on the initial gas pressure, void volumes (plenum, gap, dish, open porosity, etc.), the amount of fission gases released, and the amount of helium released from IFBA fuel. The estimated fraction of fission gases present in the gap and the plenum was about 20% for the maximum burnup rod (limiting case evaluation) at the end of three cycles of reactor operation.[(31)]

For the lead rod, the calculated internal pressure is less than the limit value; the clad stress is less than the yield strength of Zircaloy; the clad strain is less than 1% at end of life at normal operating conditions. The ability of fuel to withstand expected transients at the end of life has been evaluated. Such phenomena as high internal gas pressure have been studied under transient conditions and do not present any particular problem or significantly influence fuel behavior during transients.

The limiting criterion for evaluating mechanical performance for power transients is presently conservatively assumed to be yield stress. Both slow and rapid transients have been investigated. Slow transients, e.g., those caused by Xenon oscillations or local power shifts due to depletion, are generally of low magnitude and are of no concern since the resulting stresses will be small and will relax due to creep characteristics of the UO_2 and Zircaloy.

Rather rapid transients and transients of sufficient magnitude to cause high clad stresses could arise from some accidents such as major steam line break accident. Calculations indicate that the yield strength of the cladding could be exceeded in some fuel rods which would experience large power increase during this accident. The clad yield stress will also be exceeded during a loss of coolant accident because the yield strength of Zircaloy decreases to a very small value at high temperatures expected with this accident. It is not possible to preclude some rods from failure during severe transients. However, this is consistent with the plant design basis.

3.2.2.4 Heat Flux Ratio and DNB Correlation

WRB-1 Correlation

The DNB heat flux ration (DNBR) as applied to typical cells (flow cells with walls heated) and thimble cells (flow cells with heated and unheated walls) is defined as:

$$DNBR = \frac{q''_{DNB,N}}{q''_{loc}}$$

Where:

$$q''_{DNB,N} = \frac{q''_{DNB,EU}}{F}$$

Where:

$q''_{DNB,EU}$ = the uniform heat flux as predicted by the WRB-1 DNB correlation (Ref. 58)

F = the flux shape factor to account for nonuniform axial heat flux distributions (Reference 18) with the "C" term modified as in Reference 32

q''_{loc} = the actual local heat flux.

The WRB-1 (Reference 58) correlation was developed based exclusively on the large bank of mixing vane grid rod bundle CHF data (over 1100 points) that Westinghouse has collected. The WRB-1 correlation, based on local fluid conditions, represents the rod bundle data which has better accuracy over a wide range of variables than the previous correlation used in design. This correlation accounts directly for both typical and thimble cold wall cell effects, uniform and nonuniform heat flux profiles, and variations in rod heated length and in grid spacing.

The applicable range of variables is:

Pressure: $1440 \le P \le 2490$ psia

Local Mass Velocity: $0.9 \le \frac{G_{loc}}{10^6} \le \frac{3.7lb}{ft^2 - hr}$

Local Quality: $-0.2 \le X_{loc} \le 0.3$

Heated Length, inlet to CHF Location: $L_h \le 14$ feet

Grid Spacing: $13 \le g_{sp} \le 32$ inches

Equivalent Hydraulic Diameter: $0.37 < d_e < 0.60$ inches

Equivalent Heated Hydraulic Diameter: $0.46 < d_h < 0.58$ inches

Figure 3.2-15A shows measured critical heat flux plotted against predicted critical heat flux using the WRB-1 correlation.

As documented in References 81 and 82, a 95/95 limit DNBR of 1.17 is appropriate for 15x15 VANTAGE+ fuel assemblies.

<u>The W-3 DNB Correlation</u>

The W-3 DNB correlation [18 and 32] is used 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 [59]. For system pressure greater than 1000 psia, the W-3 correlation is limited to 1.30. A cold wall factor [33] is applied to the W-3 DNB correlation to account for the pressure of the unheated thermal surfaces.

Historical Information of W-3 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 magnitude of the local fuel rod temperature after DNB depends upon the axial location where DNB occurs.

The W-3 DNB correlation[18] was developed to predict the DNB flux and the location of DNB equally well for a uniform and an axially non-uniform heat flux distribution. This correlation replaced the preceding WAPD q" and H DNB correlations published in Nucleonics(20), May 1963, in order to eliminate the discontinuity of the latter at the saturation temperature and to provide a single unambiguous criterion for 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 form single tube data[32], but was subsequently modified to apply to the "L" –grid[33] 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[18] 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 this correlation included:

WAPD-188	1958	CU-TR-NO. 1 (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-R213 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 measured to predicted DNB flux of this correlation is given in Figure 3.2-15. The local flux DNB ratio versus the probability of not reaching DNB is plotted in Figure 3.2-16. This plot indicates that with a DNBR of 1.3 the probability of not reaching DNB is 95% at a 95% confidence level.

Rod bundle data without mixing vanes agreed very well with the predicted DNB flux as shown in Figure 3.2-17 and rod bundle data with mixing vanes (Figure 3.2-18 show, on the average, an 8% higher value of DNB heat flux than predicted by the W-3 DNB correlation.

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 is obtained when the correlation is applied to test data for rod bundles.

<u>Departure from Nucleate Boiling Ratio</u>

The DNB heat flux ratio (DNBR) as applied to the design when all flow cell walls are heated is:

$$\mathrm{DNBR} = \left(q''_{\mathrm{DNB,N}}\right)\left(F'_s\right)(0.986) \Big/ q''_{\mathrm{loc}}$$

where:

$$q''_{\mathrm{DNB,N}} = \left(q''_{\mathrm{DNB,EU}}\right) \Big/ F$$

and:

$q''_{\mathrm{DNB,EU}}$ is the uniform DNB heat flux as predicted by the W-3 DNB correlation (18) when all flow cell walls are heated.

F is the flux shape factor to account for non-uniform axial heat flux distributions [18] with the "C" term modified as in Reference 32.

F'_s is the modified spacer factor which uses an axial grid spacing coefficient, KS = 0.046, and a thermal diffusion coefficient, TDC = 0.019, based on the 26-inch grid spacing data.

q''_{loc} is the actual local heat flux.

The DNBR as applied to this design when a cold wall is present is:

$$\mathrm{DNBR} = \left(q''_{\mathrm{DBN,N,CW}}\right)\left(F'_S\right)\left(0.986\right) \Big/ q_{\mathrm{loc}}$$

where:

$$q''_{\mathrm{DNB,N,CW}} = \left(q''_{\mathrm{DBN,EU,Dh}}\right)(\mathrm{CWF}) \Big/ F$$

$q''_{DBN,EU,Dh}$ is the uniform heat flux as predicted by the W-3 cold wall DNB correlation[33] when not all flow cell walls are heated (thimble cold wall cell).

CWF = Cold Wall Factor

Local Non-Uniform DNB Flux

The local non-uniform $q''_{DNB,N}$ is calculated as follows

$$q''_{DNB,N} = \frac{q''_{DNB,EU}}{F}$$

Where:

$$F = \frac{C}{q''_{local\_at\_\ell_{DNB}} * (1 - e^{-C\ell_{DNB}})} \int_0^{\ell_{DNB}} q''(z) e^{-C(\ell_{DNB} - z)} dz \qquad (5)$$

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.

The empirical constant, C, as presented in Reference 18 has been updated through the use of more recent non-uniform DNB data. However, the revised expression does not significantly influence (less than once percent deviation from that of Reference 18) the value of the F-factor and the DNBR. It does provide a better prediction of the location of DNB. The new expression is:

$$C = 0.15 \exp\left[\frac{(1 - X_{DNB})^{4.31}}{(G/10^6)^{0.478}}\right] \text{ (in)}^{-1} \qquad (6)$$

Where

G = mass velocity, $\frac{lb}{hr - ft^2}$

X_{DNB} = quality of the coolant at the location where DNB flux is calculated.

In determining the F-factor, the value of q'' local at DNB in equation (5) was measured as $Z = \ell_{DNB}$, 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 comparisons of predictions by using W-3 correlations and the non-uniform DNB data obtained by B&W [21], Winfrith [22 & 23] and Fiat are given in Figure 3.2-19 and 3.2-20. 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 the location of DNB.\

Procedure for Using W-3 Correlation

In predicting the local DNB flux in a non-uniform heat flux channel, the following two steps are required:

1) The uniform DNB heat flux, $q''_{DNB,EU}$, is computed with the W-3 correlations using the specified local reactor conditions.

2) This equivalent uniform heat flux is converted into corresponding non-uniform DNB heat flux, $q''_{DNB,N}$, for the non-uniform flux distribution in the reactor. This is accomplished by dividing the uniform DNB flux by the F-factor. (8) 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 $\left(q''_{DNB,N}/q''_{loc}\right)$ along the channel are evaluated and the minimum value is selected as the minimum DNBR incurred in that channel.

The W-3 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 of heat flux.

Application of the W-3 Correlation in Design

During steady state operation at the nominal design conditions, the DNB ratios are determined. Under other operating conditions, particularly overpower transients, more limiting conditions develop than those existing during steady state operation. The DNB correlations are sensitive to several parameters. In addition, thermal flux general under transient conditions is also sensitive to many parameters. Therefore, for each case studied, a conservative combination of the significant parameters is used as an initial condition. These parameters include:

a) Reactor coolant system pressure
b) Reactor coolant system temperature
c) Reactor power (determined from secondary plant calorimetrics)
d) Core power distribution (hot channel factors)

For transient accident conditions where the power level, system pressure and core temperature may increase, the DNBR is limited to a minimum value of 1.30. The Reactor Control and Protection System is designed to prevent any credible combination of conditions from occurring which would result in a lower DNB ratio.

For the W-3 correlation, the 95/95 limit DNBR is 1.30 at system pressure greater than or equal to 1000 psi. For lower pressure application (500-1000 psi), the 95/95 limit DNBR is 1.45 (Reference 59).

3.2.2.5 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 [34] and is shown in Figure 3.2-21.

$$(hD/k)_f = 0.0193(DG/\mu)_f^{0.80}\left(C_p\mu/k\right)_f^{1.23}\left(p_g/p_b\right)^{0.68}\left(p_g/p_\ell\right)^{0.068-}$$

where: $p_b = p_g a + p_\ell(1-a)$

and

Cp = heat capacity at constant pressure, Btu/lb-F

D = equivalent diameter of flow channel, feet

H = heat transfer coefficient. Btu/hr-ft^2-F

G = mass flow rate, lb/hr-ft^2

k = thermal conductivity, Btu/hr-ft-F

a = void fraction

p = density, lbm/ft^3

μ = viscosity, lbm/ft-hr

<u>Subscripts</u>:

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/sq ft 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/sq ft.

3.2.2.6 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 factors consider the local maximum at a point (the “hot spot” – maximum linear power densities), and the enthalpy rise factors involve the maximum integrated value along a channel (the “hot channel”).

Definition of Engineering Hot Channel Factor

Each of the total hot channel factors is the product of a nuclear hot channel factor describing the neutron flux distribution and an engineering hot channel factor to allow for variations from design conditions. The engineering hot channel factors account for the effects of flow conditions and fabrication tolerances and are made up of subfactors accounting for the influence of the variations of fuel pellet diameter, density and enrichment; inlet flow distribution; flow redistribution; and flow mixing.

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 and enrichment, and has a value of 1.03 at the 95 percent probability level with 95 percent confidence. No DNB penalty need be taken for the short relatively low intensity heat flux spikes caused by variations in the above parameters, as well as fuel pellet eccentricity and fuel rod diameter variation.[35]

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 (VIPRE)[77, 78] thermal subchannel analysis under any reactor operating condition. The items presently considered contributing to the enthalpy rise engineering hot channel factor are discussed below:

Pellet diameter, density and enrichment.

Design values employed in the VIPRE analysis related to the above fabrication variations are based on applicable limiting tolerances such that these design values are met for 95 percent of the limiting channels at a 95 percent confidence level. Measured manufacturing data on Westinghouse fuel show that the tolerances used in this evaluation are conservative. In addition, each fuel assembly is checked to assure the channel spacing design criteria are met. The effect of variations in pellet diameter, enrichment and density is considered in establishing RTDP design limit.

Inlet Flow Maldistribution

Data have been considered from several 1/7 scale hydraulic reactor model tests [36, 37, 38] in arriving at the core inlet flow maldistribution criteria to be used in the subchannel analyses. THINC-1 analyses using these data have indicated that a conservative design basis is to consider a five percent reduction in the flow to the hot assembly.[39] The design basis of 5% flow reduction to the hot assembly is also used in the VIPRE analysis for the stretch power uprate.

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 flow redistribution is inherently considered in the VIPRE analysis

Flow Mixing

Mixing vanes have been incorporated into the spacer grid design. These vanes induce flow mixing between the various flow channels in a fuel assembly and also 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 THINC or VIPRE code and used in reload reactor design is based on experimental data.[40]

3.2.2.7 Core Pressure Drop and Hydraulic Loads

Core and vessel pressure losses are calculated by equations of the form:

$$\Delta P_L = (K + F\frac{L}{D_e})\frac{\rho V^2}{2g_c(144)}$$

where:

ΔP_L = unrecoverable pressure drop, lbf/in^2

ρ = fluid density, lbm/ft lbf/in^3

L = length, feet

D_E = equivalent diameter, feet

V = fluid velocity, ft/sec

g_c = 32.174 lbm-ft/lbf-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 1/7 scale model hydraulic test data on a number of vessels [36][37] and form loss relationships. [41] Moody [42] curves have been used to obtain the single phase friction factors.

The fuel assembly holddown springs were 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 holddown springs were designed to tolerate the possibility of an overdeflection associated with fuel assembly liftoff for this case and provide contract between the fuel assembly and the lower core plate following 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.

3.2.3 Mechanical Design and Evaluation

The reactor core and reactor vessel internals are shown in cross-section in Figure 3.2-22 and in elevation in Figure 3.2-23. The core, consisting of the fuel assemblies, control rods, source rods, burnable poison rods, and guide thimble plugging devices, provides and controls the heat source for the reactor operation. The internals, consisting of the upper and lower core support structure, were 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 in-core instrumentation. A listing of the core mechanical design parameters is given in Table 3.2-5.

The fuel assemblies are arranged in a roughly checkered circular and/or zoned pattern. The assemblies are all identical in configuration, but 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 inches within Zircaloy-4 or ZIRLO™ tubular cladding which is plugged and seal welded at the ends to encapsulate the fuel. The fuel rods are internally pressurized with helium during fabrication. The enrichments of the fuel for the first three regions and current cycle in the core are given in Table 3.2-5. 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 3.2-24. Refueling originally took place generally in accordance with an inward loading schedule, but now a modified loading schedule for low - low leakage core design is used.

The control rods, designated as Rod Cluster Control Assemblies (RCCA), consist of groups of individual absorber rods which are held together by a spider 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. Figure 3.2-25 shows a typical rod cluster control assembly.

As shown in Figure 3.2-23 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. During the refueling outrage for Cycle 8/9, the upper core plate was modified such that locations A5 and A11 were each missing one pin and location B13 was missing both pins. A special analysis was performed by Westinghouse showing that this configuration was acceptable for normal and design basis event operation. This analysis conservatively assumed absence of both alignment pins in these three locations, and also for location A6, which has two intact pins, one of which has been straightened. (71)

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 confines the upward flow of coolant in the core area to the fuel bearing region.

Reactor Internals

Design Description

The reactor internals were 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 3.2-23.

The internals were 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 that employed for Connecticut Yankee, San Onofre, Zorita, Saxton and Yankee. Under the loading conditions specified, which included conservative effects of design earthquake loading, the structure satisfied stress values prescribed in Section III, ASME Nuclear Vessel Code.

The reactor internals were fabricated primarily from type 304 stainless steel.

The reactor internals are equipped with bottom-mounted incore instrumentation supports. These supports were 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 in-core instrumentation support structure.

Lower Core Support Structure

The major containment and support member of the reactor internals is the lower core support structure, shown in Figure 3.2-23. This support structure assembly consists of the core barrel, the core baffle, and 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 provided support and orientation for the fuel assemblies.

The lower core plate is a 2-inch thick member through which the necessary flow distributor holes for each fuel assembly were machined. Fuel assembly locating pins (two for each assembly) are also inserted into this plate. Columns were placed between this plate and the bottom support plate of the core barrel in order to provide stiffness and to transmit the core load to the bottom support plate. Intermediate between the support plate and lower core support plate was positioned a perforated plate 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/core barrel but restricts radial or horizontal movement of the bottom of the shield. Rectangular tubing in which 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 plate, passes through the intermediate diffuser plate and then through the lower core plate. The flow holes in the diffuser plate and the lower core plate are arranged to give a very uniform entrance flow distribution to the core. After passing through the core and 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 thence 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 ID. 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, which are cylindrical in shape, are contoured on their bottom surface to the reactor vessel bottom head geometry. Their number and design were 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 ½ inch, and there is an additional strain displacement in the energy absorbing devices of approximately ¾ inch. 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 ¼ inch is not enough to cause the tips of the shutdown group of RCC assemblies to come out of the guide tubes in the fuel assemblies.

Upper Core Support Assembly

The upper core support assembly, shown in Figure 3.2-28, 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 and 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 3.2-29, 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°. 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 location 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 the reactor vessel head. Transverse loads form 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.

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 in-line 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 form the columns of the upper core support system. The thermocouple conduits are sealed stainless steel tubes.

In addition to the upper in-core 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 inches and the trailing ends of the thimbles (at the seal line) are extracted approximately 15 feet 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 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 feet above the seal line is cleared for the retraction operation. Section 7.4 contains more information on the layout of the incore instrumentation system.

The incore instrumentation support structure was 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.

Evaluation of Core Barrel and Thermal Shield

The internals design was based on analysis, test and operational information. Troubles in previous Westinghouse PWR’s were evaluated and information derived was 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 the Indian Point 3 RCC 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 reactor core barrel. Deflection measuring devices employed in the Connecticut Yankee reactor during the hot-functional test, and deflection and strain gages 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 provided. All the main structural welds were examined, nozzle interfaces were examined for any differential movement, upper core plate inside supports were examined, 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 1/7 scale model of the Indian Point 3 reactor. Measurements taken from those tests indicated very little shield movement, on the order of a few mils when scaled up to Indian Point. 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 testing program and the analyses with the experience gained that the design as employed on the Indian Point Plant is adequate. Further confirmation of the internals design was made on Indian Point 2. Deflection gauges were mounted on the thermal shield top and bottom for the hot-functional test. Six such gauges were mounted in the top of the thermal shield equidistant between the fixed supports and eight located at the bottom, equidistant between the six flexures, and two next to 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.

The final report of the Indian Pont 2 vibrational test, Reference 52, was transmitted to the Deputy Director for Reactor Projects in August, 1972. This report supports the use of Indian Point 2 internals as the prototype for Indian Point 3 internals.

Core Components

Design Description

Fuel Assembly

The original fuel design for Indian Point 3 was the Westinghouse Low Parasitic (LOPAR) fuel assembly. For the Cycle 5 reload, a new design fuel, the Westinghouse Optimized Fuel Assembly (OFA) was introduced (and LOPAR was phased out by Cycle 7). The major design difference between the two designs in the use of the five middle Zircaloy grids for the new design versus five middle Inconel grids for the old design. (Reference 54)

For the Cycle 7 reload, another new design, VANTAGE-5, was introduced (and by Cycle 11, the only OFA assembly is the central assembly).

For the Cycle 9 reload, ZIRLO™ clad was introduced and continues to be used.

For the Cycle 10 reload, a third design, VANTAGE+ was introduced.

For the Cycle 11 reload, the VANTAGE+ design was enhanced to include “PERFORMANCE+” features. This enhanced design has all the features of the VANTAGE+ design and also includes the Protective Bottom Grid (Reference 76).

Beginning with the Cycle 14 core, the design known as 15x15 Upgrade was introduced. This design includes an enhanced grid and IFMs to reduce grid-rod fretting. [See Section 3.2.5.5]

The overall configuration of the fuel assemblies is shown in Figures 3.2-30 and 3.2-31. The assemblies are square in cross-section, nominally 8.426 inches on a side and have an overall height of 13 feet 4 inches.

VANTAGE 5 Fuel

The Vantage 5 Fuel assembly has been designed to be compatible with the OFAs, reactor internals interfaces, the fuel handling equipment, and refueling equipment. The VANTAGE 5 design dimensions are essentially equivalent to the IP3 OFA assembly design from an exterior assembly envelope and reactor internals interface standpoint. (Reference 73)

The significant new mechanical features of the VANTGAGE 5 design relative to the OFA design in operation include the following:

- Integral Fuel Burnable Absorber (IFBA)
- Reconstitutable Top Nozzle
- Slightly longer fuel rod and assembly for extended burnup capability
- Axial Blankets
- Redesigned fuel rod bottom end plug to faciliate reconstitution capability

Other different mechanical features are the use of a standardized chamfer pellet design and the Debris Filter Bottom Nozzle (DFBN).

VANTAGE + Fuel

Vantage + uses the following V5 features

- Reconstitutable Top Nozzle (RTN)
- Extended Burnup Fuel Assembly Design
- Extreme Low Leakage Loading Pattern

- Enriched Integral Fuel Burnable Absorbers (IFBAs)
- Debris Filter Bottom Nozzle DFBN
- Axial Blankets

In addition V+ incorporates the following features as described in Reference 74

- ZIRLO™ Fuel Cladding
- Low Pressure Drop (LPD) Mid-grids
- Integral Flow Mixer grids (IFMs)
- ZIRLO™ guide thimbles and instrumentation tubes
- Variable Pitch Fuel Rod Plenum Spring
- Mid-Enriched Annular Fuel Pellets in Axial Blanket
- Fuel Assembly and Fuel Rod Dimensional Modifications

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 inch between rods. Of the total possible 225 rod locations per assembly, 20 are occupied by guide thimbles for the RCCA rods and one for in-core 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, seven grid assemblies, twenty guide thimbles, and one instrumentation thimble. Occasionally, stainless steel, zirconium alloy filler rods, or slightly enriched uranium fuel rods will replace failed fuel rods in reconstituted fuel assemblies. These rods are identical in shape and size to fuel rods. Special analyses are performed prior to any new fuel cycle utilizing reconstituted fuel.

The guide thimbles in conjunction with the grid assemblies and the top and bottom nozzles comprise the basic structural fuel assembly skeleton. The top and bottom ends of the guide thimbles are secured to the top and bottom nozzles respectively. The grid assemblies, in turn, are fastened 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.

Bottom Nozzle

The bottom nozzle is a square box-like structure which controls the coolant flow distribution to the fuel assembly and functions as the bottom structural element of the fuel assembly. The nozzle, which is square in cross-section, was fabricated from 304 stainless steel parts consisting of a perforated plate, four angle legs, and four pads or feet. The angle legs are welded to the plate forming a plenum space for coolant inlet to the fuel assembly. The perforated plate serves as the bottom limit for radiation and thermally induced growth for the fuel rods. The bottom support surface for the fuel assembly is formed under the plenum space by the four pads which are welded to the corner angles. The bottom nozzle now has more but smaller flow holes to mitigate possible fuel damage from debris.

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 through the holes in the plate. The flow holes in the plate were sized and positioned beneath the fuel rods so that the rods cannot pass through the plate.

The RCC guide thimbles, which carry axial loads imposed on the assembly, are fastened to the bottom nozzle. These loads as well as the weight of the assembly are distributed through the 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. For Vantage 5 fuel the bottom nozzle has been made thinner to allow for increased fuel rod growth and higher burnup. This feature was maintained for V+ fuel.

Top Nozzle

The top nozzle 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 adapter plate enclosure, top plate, two clamps, four 2-leaf holdown springs and assorted hardware. All parts with the exception of the springs and their hold down bolts were constructed of type 304 stainless steel. The springs were made from age hardenable Inconel 718 and the bolts from Inconel 600. The assemblies of the OFA design have about 4.5% increase in hydraulic resistance to flow compared to the LOPAR design. This results in an increased lift force to the OFA and requires 3-leaf holdown springs in the top nozzle instead of 2-leaf springs used for LOPAR assemblies.

The adapter 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 control guide thimbles were fastened through individual bored holes in the plate. Thus, the adapter 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 welded to the periphery of the adapter plate, and 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 RCC absorber rods to pass through the nozzle into the guide thimbles in the fuel assembly and for coolant exit from the fuel assembly to the upper internals area. Two pads containing axial through-holes, which are located on diametrically opposite corners of the top plate, provide a means of positioning and aligning the top of the fuel assembly. As with the bottom nozzle, alignment pins in the upper core plate mate with the holes in the top nozzle plate.

Hold-down forces of sufficient magnitude to oppose the hydraulic lifting forces on the fuel assembly are obtained by means of the leaf springs which are mounted on the top plate. The springs are fastened 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 was accomplished with a clamp which fits over the ends of the springs and two bolts (one per spring) which pass through the clamp and springs, and thread into the top plate. At assembly, the spring mounting bolts were torqued sufficiently to reload against the maximum spring load and then lockwelded to the clamp which is counter-bored to receive the bold 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. In addition, the fit between the spring and key slot and between the spring and its mating slot in the clamp are sized to prevent rotation.

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 adapter plate and the end of the guide tube in the upper internals package. Plugging devices, which fill the ends of the fuel assembly thimble tubes at unrodded core locations, and the spiders which support the source rods and burnable poison rods, are all contained within the fuel top nozzle.

The current top nozzle design is such as to allow easy reconstitution.

<u>Guide Thimbles</u>

The control rod guide thimbles in the fuel assembly provide guided channels for the control rods during insertion and withdrawal. They were fabricated from a single piece of Zircaloy-4 tubing, which is drawn to two different diameters. Starting with Cycle 10, the guide thimble material is ZIRLO™. The larger inside diameter at the top provides a relatively large annular area for rapid insertion during a reactor trip and to accommodate a small amount of upward cooling flow during normal operations. The bottom portion of the guide thimble is a reduced diameter to produce a dashpot action when the absorber rods near the end of travel in the guide thimbles during a reactor trip. The transition zone at the dashpot section is conical in shape so that there are no rapid changes in diameter in the tube.

The OFA design guide thimbles are the same as the guide thimbles for LOPAR design except for a 13 mil inner diameter and water diameter reduction. The OFA guide thimble diameter provides adequate diametral clearance for control rods, source rods, burnable absorber rods, and thimble plugs.

Flow holes are provided just above the transition of the two diameters 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 closed at the bottom by means of a welded end plug. The end plug was fastened to the bottom nozzle during fuel assembly fabrication.

The top ends of the guide thimbles are expanded into stainless steel sleeves which are welded to the top grid. The sleeves are fitted through individual holes in the adaptor plate and welded around the circumference of the holes.

The 15x15 Upgrade incorporates the new tube-in-tube guide thimble design which 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. As the dashpot in this 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 this design modification. The thimble screw for the tube-in-tube design is slightly longer than in the previous guide thimble tube design so that it can properly engage with the threads on the new guide thimble end plug and extend through the end plug of the dashpot tube assembly.

Grids

The spring clip grid assemblies consist of individual slotted straps which are assembled and interlocked in an “egg-crate” type arrangement and then furnace brazed to permanently join the straps at their points of intersection. Details such as spring fingers, support dimples, mixing vanes, and tabs were punched and formed in the individual straps prior to assembly.

Different types of grid assemblies are used in the fuel assembly. One type having mixing vanes which project from the edges of the straps into the coolant stream, is used in the high heat region of the fuel assemblies for mixing of the coolant. A grid of this type is shown in Figure 3.2-32. Grids of another type, located at the bottom and top ends of the assembly, are of the non-mixing type. They are similar to the mixing type with the exception that mixing vanes are not used on the internal straps. They are made of Inconel.

The spacing between grids is shown on Figure 3.2-31. The variation in span lengths is the result of optimization of the thermal-hydraulic and structural parameters. The grids are fastened securely to each guide thimble.

The outside straps on all grids contain mixing vanes which, in addition to their mixing function, aid in guiding the fuel assemblies past projecting surfaces during handling or loading and unloading the core. Additional small tabs on the outside straps and the irregular contour of the straps are also for this purpose.

Inconel 718 was chosen for the grid material for the top and bottom grid because of its good corrosion resistance and high strength properties. After the combined brazing and solution annealing temperature cycle, the grid material was age hardened to obtain the material strength necessary to develop the required grid spring forces.

The OFA design features five middle Zircaloy-4 grids which have thicker and wider straps than the LOPAR Inconel grids to compensate for the difference in natural strength properties. Zircaloy grids maintain their integrity during the most severe load conditions of a combined seismic/LOCA event.

The Vantage+ fuel assemblies have five Low Pressure Drop (LPD) mid grids and three intermediate flow mixing (IFM) grids for increased DNB margin. These are all made of ZIRLO™.

The I-Spring design, introduced in Cycle 14 with the 15x15 Upgrade fuel, uses a revised spring, dimple and strap design to reduce the probability of grid-rod fretting.

Fuel Rods

The fuel rods consist of uranium dioxide ceramic pellets contained in a slightly cold worked and partially annealed Zircaloy-4 or ZIRLO™ tubing which plugged and seal welded at the ends to encapsulated 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, and fuel swelling due to accumulated fission products without over-stressing 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. All fuel rods are internally pressurized with helium during the welding process. [47] A hold-down force of approximately six 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.

The fuel pellets are right circular cylinders consisting of slightly enriched uranium-dioxide power which has been compacted by cold pressing and then sintered 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.

A different fuel enrichment, as listed in Table 3.2-5, was used for each of the three regions in the first core loading. (Checker Board Pattern, see Figure 3.2-24.) Subsequent regions were uniquely designed for their enrichments. Current cycle is provided in Table 3.2-5 and 3.2-24A.

Each fuel assembly is identified by means of a serial number engraved on the upper nozzle. The fuel pellets are fabricated by a batch process so that only one enrichment region is processed at any given time. The serial numbers of the assemblies and corresponding enrichment are documented by the manufacturer and verified prior to shipment.

Each assembly is assigned a specific core loading position prior to insertion. A record is then made of the core loading position, serial number, and enrichment.

During initial core loading and subsequent refueling operations, detailed written handling and checkoff procedures were utilized throughout the sequence. The initial core was loaded in accordance with the core loading diagram similar to Figure 3.2-24 which shows the location for each of the three enrichment types of fuel assemblies used in the loading together with the serial number of the assemblies in the region.

Extensive administrative controls (as discussed in Sections 3.2.3 and 3.3.3) render the possibility of loading fuel assemblies with incorrect enrichments or without their burnable poison rods extremely unlikely. Independent checks are made, prior to fuel loading, of each fuel assembly matching the contents of the assembly with its position in the core. Further checks are provided during core loading utilizing detailed written handling and checkoff procedures.

Achieving criticality during core loading is prohibited in any case as the subcritical neutron flux is continuously monitored [Deleted] to detect any unexpected rise in the subcritical neutron flux. Core loading is stopped should the subcritical count rate increase unexpectedly or behave in an unstable manner. Procedures require the fuel handlers to receive permission from Reactor Engineering prior to unlatching a fuel assembly that has been inserted into the core.

Any such loading error, not significant enough to be detected during initial core loading is of no consequence from a criticality standpoint and would be detected by the power distribution map. During subsequent refueling operations the flux profile is flatter and loading errors can be detected by the power distribution map. (See the Technical Specifications)

Rod Cluster Control Assemblies

The control rods or rod cluster control assemblies (RCCA) each 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 3.2-25 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 significantly increase its worth. The alloy is in the form of extruded single length rods which are sealed stainless steel tubes to prevent the rods from coming in direct contact with the coolant. For Cycle 16, there are 14 new control rods that have chrome plating on the stainless steel tube to reduce wear.

The overall control rod length is such that when the assembly has been withdrawn through its full travel, the tip 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 RCC 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 RCC assembly and absorb the impact energy 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 assure trouble free service. The rods were first threaded into the spider fingers and then pinned to maintain joint tightness, after which the pins were welded in place. The end plug below the pin position was 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 were 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 were provided to accommodate relative thermal expansions and to limit the internal pressure to acceptable levels.

The bottom plugs were 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 (14 with chrome plating) silver-indium-cadmium alloy absorber rods are resistant to radiation and thermal damage thereby ensuring their effectiveness under all operating conditions.

<u>Neutron Source Assemblies</u>

Neutron source assemblies are utilized in the core. In Cycle 1, these consisted of two assemblies with four secondary source rods each and two assemblies with one primary source rod each. In subsequent cycles, secondary sources only are utilized. The secondary source rods are fastened to a spider at the top end of the assembly. The initial core primary source rods were attached to a burnable poison assembly. Beginning with Cycle 14, secondary source assemblies using baseplate mounts began to be introduced into the core.

In the core, the neutron source assemblies are inserted into the RCC guide thimbles in fuel assemblies at unrodded locations. The location and orientation of the assemblies in the initial core is shown in Figure 3.2-33.

The primary and secondary source rods of the initial core utilized the same type of cladding material as the absorber rods (cold-worked type 304 stainless steel tubing) in which the sources are inserted. The source rods contain Sb-Be pellets. The primary source rods each contained capsules of Pu 238-Be source material at a neutron strength of approximately 2 x 10^8 neutrons/sec. Design criteria for the source rods are: 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 clad. For reload cores, the primary source rods have been removed and only the secondary source rods remain in core locations determined by the reload design.

Plugging Devices

In order to limit bypass flow through the RCC guide thimbles in fuel assemblies which do not contain either control rods, source assemblies, or burnable poison rods, the fuel assemblies at those locations were fitted with plugging devices. The plugging devices consist of a flat base plate with short rods suspended from the bottom surface and a spring pack assembly and mixing device attached to the top surface. At installation in the core, the plugging devices fitted with the fuel assembly top nozzles and rested on the adapter plate. The short rods project the upper ends of the thimble tubes to reduce the bypass flow area. The spring pack was compressed by the upper core plate when the upper internals package was lowered into place. Similar short rods are also used on the source assemblies to fill the ends of all vacant fuel assembly guide thimbles and on burnable poison rod assemblies in these guide thimble positions where there are no burnable poison rodlets.

All components in the plugging device, except for the springs, were constructed from type 304 stainless steel. The springs (one per plugging device) were wound from an age hardenable nickel base alloy to obtain higher strength.

Burnable Poison Rods

The burnable poison rods are statically suspended and positioned in vacant assembly guide thimble tubes within the fuel assemblies at nonrodded RCC core locations. The poison rods in each fuel assembly are grouped and attached together at the top end of the rods by a flat base plate which fits with the fuel assembly top nozzle and rests on the top adapter plate.

The base plate and the poison rods are held down and restrained against vertical motion through 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 poison rods cannot be lifted out of the core by flow forces.

The old poison rods consist of borosilicate glass tubes contained within type 304 stainless steel tubular cladding which is plugged and seal welded at the ends to encapsulate the glass. The glass is also supported along the length of its inside diameter by a thin wall type 304 stainless steel tubular inner liner. A typical old burnable poison rod is shown in longitudinal and transverse cross-sections in Figure 3.2-34. These rods can still be used for fluence reduction on the reactor vessel, as well as for burnable poison. The new poison rod design is referred to as Wet Annular Burnable Absorber (WABA). The WABA design has an annular aluminum oxide-boron carbide (Al_2OB_4C) absorber pellets contained within two concentric zircaloy tubings with water flowing through the center tubes as well as around the outer tubes. Eight demonstration poison rods of a new design, consisting of annular pellets of Al_2OB_4C contained within two concentric zircaloy tubings were also employed in the Cycle 3 and Cycle 4 cores. Cycles 5 and beyond have used large complements of WABA. Cycle 7 also utilized the Integral Fuel Burnable Absorber (IFBA). This consists of a thin coating of zirconium diboride on the cyclindrical surface of the pellet. Cycles 9 and beyond have used IFBA and WABA.

The rods were designed in accordance with the standard fuel rod design criteria; i.e.., the cladding is free standing at reactor operating pressures and temperatures and sufficient cold void volume is provided within the rods to limit internal pressures to less than the reactor operating pressure assuming total release of all helium generated in the glass or annular pellets as a result of the $B_{10}(n,\gamma)$ reaction. A more detailed discussion of the old burnable poison rod design is found in WCAP-9000. [(45)] A more detailed discussion of the new burnable poison rods if found in WCAP-10021 (Revision 1) [(55)]. The new burnable poison rod is shown in longitudinal and transverse cross-sections in Figure 3.2-34.

Based on available data on properties of borosilicate glass and on nuclear and thermal calculations for the rods, gross swelling or cracking of the glass tubing is not expected during operation. Some minor creep of the glass at the hot spot on the inner surface of the tube is expected to occur but continues only until the glass comes into contact with the inner liner. The inner liner is provided to maintain the central void along the length of the glass and to prevent the glass from slumping or creeping into the void as a result of softening at the hot spot. The wall thickness of the inner liner was sized to provide adequate support in the event of slumping but to collapse locally before rupture of the exterior cladding if large volume changes due to swelling or cracking should possibly occur.

The top end of the inner liner is open to receive the helium which diffuses out of the glass.

To ensure the integrity of the burnable poison rods, the tubular cladding and end plugs were procured to the same specifications and standard of quality as was used for stainless steel fuel rod cladding and end plugs in other Westinghouse plants. In addition, the end plug seal welds were checked for integrity by visual inspection and X-ray. The finished rods were helium leak checked.

Starting in Cycle 11, Hafnium Flux Suppressors have been placed in the eight core locations closest to the reactor vessel to suppress neutron fluence on the reactor vessel, thereby prolonging vessel life. These hafnium flux suppressors replace eight borosilicate-glass (Pyrex) burnable absorbers that were used up to Cycle 10. The hafnium flux suppressors are constructed with the same upper fixture as the Pyrex and WABA burnable absorbers currently available for use, and are handled in the same manner during refueling. The flux suppressors use unclad, hafnium rods and may be used in future operating cycles. (Reference 76)

Protective Bottom Grid

Starting with Cycle 11, the fuel assemblies (known as PERFORMANCE+) have a Protective Bottom Grid, which is an extra grid strap located at the bottom of the fuel pins, between the Bottom Nozzle and the bottom grid. The purpose of this protective bottom grid is to capture debris entering the fuel assembly and trap it at an elevation below the top of the fuel pin end plug. Therefore, any debris caught by the protective bottom grid cannot fret through or otherwise damage the fuel pin cladding in such a manner that the fuel pellets or rod plenum is exposed, thus making the fuel assembly more resistant to debris-related fuel failures.

The PERFORMANCE+ design also has the fuel pins mounted lower in the grid cage than fuel assembly designs before Cycle 11, almost touching the bottom nozzle. During operation, it is expected that pin growth and grid relaxation will allow the pin to rest on the bottom nozzle. The resultant transfer of weight from the grid cage to the bottom nozzle is expected, and may result in less bowing of the irradiated fuel assembly. (Reference 76)

<u>Evaluation of Core Components</u>

Fuel Evaluation

The fission gas release and the associated buildup of internal gas pressure in the fuel rods were calculated by an overall fuel rod design model [48] which incorporates the time-dependent fuel densification. [49] The increase of internal pressure in the fuel rod due to this phenomena was included in the determination of the maximum cladding stresses at the end of core life when fission product gap inventory is a maximum. IFBA fuel has an additional contribution to rod internal pressure in the form of helium which is assumed to release at a rate of 100 percent.

The maximum allowable strain in the cladding, considering the combined effects of internal fission gap pressure, external coolant pressure, fuel pellet swelling and clad creep is limited to less than one percent throughout core life. The associated stresses are below the yield strength of the material under all normal operating conditions.

To assure that manufactured fuel rods have met a high standard of excellence from the standpoint of functional requirements, many inspections and tests were performed both on the raw material and the finished product. These tests and inspections included chemical analysis, elevated temperature, 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 defect, the high resistance of uranium dioxide fuel pellets to attack by hot water protects against fuel deterioration. 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.

A survey of high burnup uranium dioxide [43] fuel element behavior indicates that for an initial uranium dioxide void volume, which is a function of the fuel density, it is possible to conservatively define the fuel swelling as a function of burnup. The fuel swelling model considered the effect of burnup, temperature distribution, and internal voids. It was incorporated in the overall fuel rod design model. [48]

Actual damage limits depend upon neutron exposure and normal variation of material properties and are greater than the design limits. For most of the fuel rod life the actual stresses and strains are considerably 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 stain are as follows:

1) Internal gas pressure
   The maximum rod internal pressure under nominal conditions is substantially less than the calculated pressure at the design limits. The end-of-life internal gas pressure depends upon the initial pressure, void volume, and fuel rod power history and IFBA loading. However, it does not exceed the design limit defined in Section 3.1.2.

2) Cladding temperature
   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 (715°F) is given in Table 3.2-4, along with many other thermal and hydraulic design parameters.

3) Burnup
   Fuel burnup results in fuel swelling which, along with 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 operation or malfunction transients.

Evaluation of Burnable Poison Rods

The burnable poison rods are [Deleted] positioned in the core inside RCC assembly guide thimbles and held down in place by attachment to a baseplate assembly compressed beneath the upper core plate and hence cannot be ejected and cause a reactivity transient. Due to the low heat generation rate, and the conservative design of the poison 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.

Many nuclear plants are using or have used burnable poison rods of the old design. Reference 51 describes the test operational experience of the old poison rod design. Eight demonstration poison rods of the new design were used in Cycles 3 and 4. Post irradiation examination following Cycles 3 and 4 indicated rodlets performed as expected and these were no anomalies.

Further details of examinations performed can be referenced in WCAP-10021 (Revision 1). [55] The design was retained for Cycles 5 and beyond.

Effects of Vibration and Thermal Cycling on Fuel Assemblies

Analyses of the effect of cyclic deflection of the fuel rods, grid spring fingers, RCC control rods, and burnable poison rods due to hydraulically induced vibrations and thermal cycling show that the design of the components is good for the nominal design lifetime of 12 Effective Full-Power Years.

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), and the stress associated with the motion is significantly small (less than 100 psi). Like wise, 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 merely a slight relative movement between the grid contact surfaces and the clad, which is gradual in nature during heat-up and cool-down. Since the number of cycles of the occurrence is small over the life of a fuel assembly, negligible wear of the mating parts is expected.

These conclusions have been verified by fuel operating experience in a number of nuclear plants as described in Reference 51.

The dynamic deflection of the full length control rods and the burnable poison 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 clad tubing or in the flexture 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 poison 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.

Control Rod Drive Mechanism

Full Length Rods

Design Description

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. Typical total insertion time is less than 1.8 seconds and always less than 2.7 seconds.

The complete drive mechanism, shown in Figure 3.2-36, consists of the internal (latch) assembly, the pressure vessel, the operating coil stack, the drive shaft assembly, and the rod position indictor coil stack.

Each assembly is an independent unit which can be dismantled or assembled separately. Each mechanism pressure hosing is threaded onto an adapter 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 control rod drive mechanism. All working components and the shaft are immersed in the main coolant and depend on it for component damping.

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 into 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 144 inches 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 were designed to operate in water 650°F at and 2485 psig. The temperature at the mechanism head adapter 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 multi-conductor cable connects the mechanism operating coils to the 125 volt DC power supply. The power supply is described in Section 7.3.2.

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 inch. The lower set of latches have a maximum 1/16 inch axial movement to shift the weight of the control rod from the upper to the lower latches.

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.

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 operator 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.

Drive Shaft 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 inches of control rod travel. The grooves are spaced 5/8 inch apart to coincide with the mechanism step length and have 45° 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 ¼ inch diameter disconnect rod runs down the inside of the drive shaft. It utilizes a locking button at its lower end to lock the coupling and control rod. At its lower end, there is a disconnect assembly. For remote disconnection of the drive shaft assembly from the control rod, a button at the top of the drive rod actuates the connect/disconnect 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.

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 inches).

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 steel, 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 304 stainless steel is used for all pressure containment. Hard chrome plating provides wear surfaces on the sliding parts and prevents galling between mating 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 to approximately 0.001 inch thickness to prevent corrosion.

Principles of Operation

The drive mechanisms, shown schematically in Figures 3.2-35 and 3.2-36 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 differential transformer action of the position indicator coil stack surrounding the rod travel housing. The differential 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 moveable 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 then swing out of the shaft groove.

3) Lift Coil – ON
   The 5/8 inch 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 inch. 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 the spring 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 moveable gripper latches drop 5/8 inch 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 and spring force separate the lift armature from the lift magnetic pole and the control rod drops down 5/8 inch.

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 inch for each cycle. Each sequence can be repeated at a rate of up to 72 steps per minute and the control rods can therefore be withdrawn or inserted at a rate of up to 45 inches per minute.

Control Rod Tripping:

The holding or static mode is with the stationary gripper coil. If power to the stationary 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 stationary gripper plunger half against the stationary gripper pole, collapses and the stationary gripper plunger half is forced down by the weight acting upon the latches.

Fuel Assembly and RCCA mechanical Evaluation

To confirm the mechanical adequacy of the fuel assembly and full length RCC assembly, functional test programs were conducted on a full scale Indian Point 2 prototype 12 ft can less fuel assembly and control rod. The prototype assembly was tested under simulated conditions of reactor temperature, pressure, and flow for approximately 1000 hours. The prototype mechanism accumulated 2,260,892 steps and 600 scrams. At the end of the test the CRDM 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% of nominal flow was less than 1.5 seconds to the dashpot (10 ft of travel). Additional tests had previously been made on a full scale San Onofre mock up version of the fuel assembly and control rods. [(44)]

Mockup Tests at 1/7 Scale for Indian Point 2

A 1/7 scale model of the Indian Point 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 were 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 were obtained.

Loading and Handling Tests

Tests simulating the loading of the prototype fuel assembly into a core location were also successfully conducted to determine that proper provisions had been made for guidance of the fuel assembly during refueling operation.

Axial and Lateral Bending Tests

In addition, axial and lateral bending tests were 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 was also used in the design of fuel handling equipment to establish the limits for inadvertent axial loads during refueling.

3.2.4 ZIRLO Clad Fuel

Fuel assemblies containing fuel rods fabricated with the advanced zirconium alloy cladding material ZIRLO™ are being used in the Indian Point Unit 3 Cycle 9 core and beyond. These fuel assemblies will have fuel rods fabricated with ZIRLO™ cladding to obtain additional operational benefit from the cladding's improved corrosion resistance. The transition to VANTAGE 5 fuel was made at Indian Point 3 for Cycle 7 as described in the submittal to the NRC dated January 20, 1989 (IPN-89-007). Cycle 10 began the use of V+ fuel which utilizes ZIRLO™ for guide thimbles and intermediate spacers as well as clad.

This report shows, based on both evaluations and analyses, that no unreviewed safety questions exist as a result of inserting ZIRLO™ clad fuel rods into the Indian Point 3 reactor core. This report shows that the subsequent proposed changes to the Indian Point Unit 3 Technical Specifications will not involve significant hazard considerations.

3.2.4.1 Background

Westinghouse has developed a new zirconium based fuel rod clad alloy, known as ZIRLO™, to enhance fuel reliability and achieve extended burnup. This alloy provides significant improvement in fuel rod clad corrosion resistance and dimensional stability under irradiation. ZIRLO™ cladding corrosion resistance has been evaluated in long-term, out-of-pile tests over a wide range of temperatures (up to 600°F in water tests, up to 932°F in steam tests). Additional tests have also been conducted in lithiated water environments. The improved corrosion resistance of ZIRLO™ cladding has also been demonstrated to very high burnups in the BR-3 reactor.

A conditional licensing approval for the use of this advanced alloy cladding in two demonstration fuel assemblies for the North Anna Unit 1 reactor core was given in a USNRC letter dated May 13, 1987. The USNRC granted an exemption (Reference 66) from the provision of 10CFR50.46, 10 CFR50.44 and 10CFR51.52 with respect to the use of the North Anna demonstration fuel assemblies with the advanced cladding material, ZIRLO™. The Information required to support the licensing basis for the implementation of the ZIRLO™ clad fuel rods in Indian Point 3 is given in References 67 and 68. The fuel assemblies will be utilized in Indian Point 3, beginning with Cycle 9.

3.2.4.2 Areas Assessed

The following areas have been assessed during the safety evaluation process: chemical/mechanical properties, neutronic performance, thermal and hydraulic performance, cladding performance under non-LOCA conditions, and cladding performance under LOCA conditions.

Reference 69 addresses the VANTAGE 5 design and its application to a 17 x 17 fuel assembly. The VANTAGE 5 design may be applied to other fuel assembly arrays (14 x 14, 15 x 15) where such applications are evaluated on a plant specific basis and licensed in accordance with NRC requirements. Indian Point Unit 3 has been licensed for 15 x 15 VANTAGE 5 as already noted. Subsequently, the applicable models and methods employed to address the 15 x 15 VANTAGE 5 design have been licensed for Indian Point Unit 3. The principal difference between the Indian Point Unit 3 Region 11 fuel and the licensed 15 x 15 VANTAGE 5 fuel is the use of ZIRLO™ cladding. The use of ZIRLO™ cladding does not alter the previously licensed models and methods of Reference 69 with the exception of the LOCA model and methodology as noted in Chapter 14. The revised LOCA model and methodology were used as the basis to evaluate the effects of the change in cladding material. These evaluations have shown that the present LOCA related design bases and limits remain valid. Where the models and methods of Reference 69 are not affected by ZIRLO™ cladding, Indian Point 3 plant specific evaluations and analyses have also shown that the current design bases and limits remain valid.

3.2.4.3 Previous Irradiation Experience

Fuel rods fabricated with ZIRLO™ cladding have been previously irradiated in a foreign reactor (BR-3 reactor) at linear power levels up to 17 kw/ft, and burnups significantly greater than those planned for the Indian Point 3 fuel assemblies. Corrosion and hydriding data obtained on the ZIRLO™ were compared with the reference Zircaloy-4 cladding of fuel rods irradiated as controls in the same test assemblies. Based on the irradiation results of the test assemblies in the foreign reactor, the Indian Point Unit 3 ZIRLO™ cladding waterside corrosion and hydriding will be significantly less than that expected for the Zircaloy-4 clad fuel rods. The irradiation test results substantiate a lower clad irradiation growth $(\Delta L / L)$ and creepdown for the ZIRLO™ cladding compared to Zircaloy-4 cladding.

Two demonstration fuel assemblies, containing ZIRLO™ clad fuel rods, began irradiation in the North Anna Unit 1 reactor during June 1987. The ZIRLO™ clad fuel rods achieved over 21,000 MWD/MTU burnup in their first cycle (complete during February 1989). Visual inspection during refueling showed no abnormalities. One demonstration assembly with ZIRLO™ TM clad fuel rods underwent a second cycle of irradiation and achieved over 37,000 MWD/MTU burnup (completed January 1991). Visual inspection of the two cycle ZIRLO™ clad fuel rods during refueling showed no abnormalities. Cladding corrosion measurements showed that the reduced corrosion obtained with the ZIRLO™ clad rods was significantly better than that anticipated on the basis of licensing basis evaluations. The present and future irradiation results are and will be considered in the design of the fuel rods with ZIRLO™ cladding to assure that all fuel rod design bases are satisfied for the planned irradiation life of the Indian Point Unit 3 fuel assemblies.

3.2.4.4 Chemical/Mechanical Properties

This chemical composition (see Table 3.2-6) of the ZIRLO™ clad fuel rods in the Indian Point 3 fuel assemblies is similar to Zircaloy-4 except for slight reductions in the content of tin (Sn), iron (FE), and zirconium (Zr) and the elimination of chromium (Cr). ZIRLO™ cladding also contains a nominal amount of niobium (Nb). These small composition changes are responsible for the improved corrosion resistance compared to Zircaloy-4. The physical and mechanical properties are very similar to Zircaloy-4 while in the same metallurgical phase. However, the temperatures at which the metallurgical phase changes occur are different for Zircaloy-4 and ZIRLO™ cladding (Appendix A of Reference 67). These differences are considered in the evaluations discussed below for cladding behavior under non-LOCA and LOCA conditions. Further aspects of the ZIRLO™ cladding performance under LOCA conditions are given in Reference 67). Evaluations have been performed using the NRC approved fuel rod performance code (Reference 70) to verify that the fuel rod design bases and design criteria are met for assemblies containing ZIRLO™ clad fuel rods. The fuel rod design bases, criteria and models, which are affected by the use of ZIRLO™ cladding are described in Reference 67.

3.2.4.5 Neutronic Performance

The design and predicted nuclear characteristics of fuel rods with ZIRLO™ cladding are similar to those of VANTAGE 5 design (Reference 69). The evaluations have shown (Reference 67) that the nuclear design bases are satisfied for fuel rods with ZIRLO™ cladding and that the use of ZIRLO™ cladding will not affect the standard nuclear design analytical models and methods to accurately describe the neutronic behavior of fuel rods with ZIRLO™ cladding. The safety limit characteristics of the VANTAGE 5 fuel design (Reference 69) are not affected.

3.2.4.6 Thermal and Hydraulic Performance

The thermal and hydraulic design bases for fuel rods with ZIRLO™ cladding are identical to those of the VANTAGE 5 design (reference 69). Since the use of the ZIRLO™ clad fuel does not cause changes affecting the parameters which are major contributors in this area (i.e., DNB, core flow, and rod bow), the design bases of the VANTAGE 5 design (Reference 69) remain valid.

3.2.5 VANTAGE + Design Features

The design features of the V+ fuel design include the following:

- Current Vantage 5 (w/o IFMs) Fuel Features
    - Reconstitutable Top Nozzle (RTN)
    - Extended Burnup Fuel Assembly Design
    - Extreme Low Leakage Loading Pattern
    - Enriched Integral Fuel Burnable Absorbers (IFBAs)
    - Debris Filter Bottom Nozzle (DFBN)
    - Axial Blankets
- ZIRLO™ fuel cladding
- Low Pressure Drop (LPD) Mid grids
- Integral Flow Mixing grids (IFMs)
- ZIRLO™ guide thimble and instrument tubes
- Variable Pitch Fuel Rod Plenum Spring
- Mid-enriched Annual Fuel Pellets in Axial Blanket
- Fuel Assembly and Fuel Rod Dimensional Modifications
- Low Cobalt Top and Bottom Nozzles
- Coated Cladding (Pre-oxidized ZIRLO™ cladding)
- Gripable Top End Plug

3.2.5.1 VANTAGE 5 Fuel Features

The use of ZIRLO™ fuel cladding was implemented in the VANTAGE 5 fuel design beginning in Cycle 9 after issuance of the V5 SER.

With respect to the Extended Burnup Fuel Assembly Design, the V+ fuel design includes an increase in the region average discharge burnup from 40,000 + to 50,000 + MWD/MTU. The effects of the increase in extended burnup on the performance of the fuel are included in the safety analysis via the reload safety analysis parameters which are taken into account in the reload design process.

3.2.5.2 Mid-enriched Annular Fuel Pellets in Axial Blankets and Enriched IFBAs

Axial blankets reduce power at the ends of the rod which increases axial peaking toward the middle of the rod. Used alone, axial blankets reduce DNB margin, but the effect may be offset by the presence of IFBAs which flatten the power distribution. The net effect on the axial shape is a function of the number and configuration of IFBAs in the core and the time in core life. The effects on the reload safety analysis parameters due to axial blankets, including annular fuel pellets in the axial blanket, and IFBAs, including enriched IFBAs, are taken into account in the reload design process. The axial and radial power distribution assumptions used in the safety analysis kinetics calculations have been determined to be applicable for evaluating the axial blankets and IFBAs in the IP3 V+ fuel design.

3.2.5.3 LPD Mid Grids, IFM ZIRLO Guide Thimbles and Instrument Tubes, Low Cobalt Top and Bottom Nozzles and Fuel Assembly and Fuel Rod Dimensional Modifications.

The IP3 V+ fuel design incorporates the use of ZIRLO™ LPD mid grids, IFMs, guide thimbles and instrument tubes, and includes minor fuel assembly and fuel rod dimensional modifications to accommodate the Extended Burnup Fuel Assembly design.

With respect to the non-LOCA accident analysis, the use of ZIRLO™ guide thimbles and instrument tubes, low cobalt top and bottom nozzles and the minor fuel assembly and fuel rod dimensional modifications have no direct effect on the analysis results since the characteristics of these features are not specifically modeled in the transient analysis. Any effects of these items on the performance of the fuel are included in the safety analysis via the reload safety analysis parameters (e.g., fuel temperatures, flow rates, pressure drops) which are taken into account in the reload design process.

The use of the LPD mid grids in the V+ fuel design are primarily for the purpose of offsetting the increase in flow resistance associated with the introduction of the IFMs. Therefore, with respect to the parameters which affected the non LOCA accident analysis, the V+ fuel design is hydraulically compatible with the resident V5 fuel design and has no direct effect on the non LOCA safety analysis. Any localized assembly to assembly hydraulic differences which may affect the performance of the fuel are included in the safety analysis via the reload safety analysis parameters which are taken into account in the reload design process and documented in the Reload Safety Evaluation.

With respect to the implementation of IFMs V+ fuel design, it should be noted that although the use of IFMs provides a DNB benefit caused by enhanced flow mixing, the benefit is not fully realized in the safety analysis since the resident V5 fuel design does not include IFMs and therefore is more limiting with respect to DNB. As a result, the Core Thermal Limits and resulting OTΔT and OPΔT reactor trip setpoints applicable for the transition to VANTAGE + Fuel are currently based on the DNB performance of the more limiting VANTAGE 5 (w/o IFMs) fuel. Once the transition to a full core V+ fuel is completed, the DNB benefit associated with the IFMs can be fully realized.

3.2.5.4 Variable Pitch Fuel Rod Plenum Spring

The optimized fuel rod plenum spring feature of the V+ fuel design is primarily to provide increased fuel rod plenum volume which benefits rod internal pressure concerns. Any effects of this spring design on the performance of the fuel is included in the safety analysis via the reload safety analysis parameters which are taken into account in the reload design process.

3.2.5.5 15x15 Upgrade Design

Beginning with Cycle 14, a modified fuel design, designated “15x15 Upgrade”, was introduced to the Indian Point Unit 3 Core (IP3).

Fuel cladding is ZIRLO on all 15 x 15 Upgrade fuel assemblies [Deleted]. ZIRLO is also used on intermediate grid straps, IFMs and thimble tubes. The Westinghouse model designation for this modified design fuel is 15x15 Upgrade Fuel with I-Spring (hereafter called 15x15 Upgrade). The IP3 core was fully loaded with 15x15 Upgrade fuel in Cycle 15, with the exception of one VANTAGE+ assembly in the center core location, which is a location not considered susceptible to grid-rod fretting. Cycle 16 has a complete core of 15 x 15 Upgrade fuel.

The 15x15 Upgrade fuel design includes a number of new features:

1) I-Spring mid-grid design to reduce grid-to-rod fretting
2) Enhanced Intermediate Flow Mixers (IFMs) to reduce grid-to-rod fretting
3) Balanced vane pattern on all grids to provide more balanced flow for vibration reduction.
4) Tube-in-tube guide thimble design, replacing the Vantage+ dashpot design. The new guide thimbles are designed to improve margin to Incomplete Rod Insertion (IRI).

In addition to these features, the 15x15 Upgrade Fuel includes fuel performance and reliability features that have been added to IP3 fuel in previous cycles. These include: debris resistant bottom nozzles, fuel rod oxide coatings, protective bottom grid and bead-blasted top nozzle spring screws. However, the grade of inconel used in the spring screws will be Type 718, which is stronger than the previously-used Type 600.

Beginning with Cycle 15, mixed axial blankets were introduced to the IP3 core. Axial blankets for previous Unit 3 reload regions have consisted of annular pellets. Starting with the Cycle 15 feed fuel, a combination of annular and solid pellets are used. Annular blankets continue to be utilized in IFBA fuel rods, and solid blanket pellets are used in non-IFBA fuel rods. This approach maximizes the assembly uranium loading while continuing to provide the necessary free volumes in IFBA rods to accommodate helium gas release.

<u>References</u>

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

2) Barry, R. F., et al, “Power Maldistribution Investigations”, WCAP-7407-L, Westinghouse Proprietary, (1970).

3) Westinghouse Proprietary, “Power Distribution Control in Westinghouse Pressurized Water Reactors,” WCAP-7208 (1968).

4) Barry, R. F., “The Revised LEOPARD Code – A Spectrum Dependent Non-Spatial Depletion Program,” WCAP-2759, March 1965.

5) Langford, F. L. and R. J. Nath, “Evaluation of Nuclear Hot Channel Factor Uncertainties,” WCAP-7308-L, April 1969 (Proprietary) and WCAP-7810, December 1971 (Non-Proprietary).

6) Hellman, J. M., (ed.), “Fuel Densification Experimental Results and Model for Reactor Application,” WCAP-8218-P-A, March 1975 (Proprietary) and WCAP-8219A, March 1975 (Non-Proprietary).

7) Deleted

8) “Power Distribution Control of Westinghouse Pressurized Water Reactors,” WCAP-7811, December 1971 (Non-Proprietary).

9) Morita, T., et al., “Power Distribution Control and Load Following Procedures,” WCAP-8385, September 1974 (Proprietary) and WCAP-8403, September 1974 (Non-Proprietary).

10) Deleted

11) Dean, R. A., “Thermal Contact Conductance Between UO2 and Zircaloy-2,” CVNA-127, May 1962.

12) Ross, A. M., and R. D. Stoute, “Heat Transfer Coefficient Between UO2 and Zircaloy-2,” AECL-1552, June 1962.

13) Godfrey, T. G., et al., “Thermal Conductivity of Uranium Dioxide and Armco Iron by an Improved Radial Heat Flow Technique,” ORNL-3556, June 1964.

14) Robertson, J. A. L., et al., “Temperature Distribution of UO2 Fuel Elements,” Journal of Nuclear Materials Vol. 7, No. 3, pp 252-262, 1962.

15) Duncan, R. N., “Rabbitt Irradiation of UO2,” CVNA-142, June 1962.

16) Horn, G. R., and J. A. Christensen, “Identification of the Molten Zone in Irradiated UO2,” ANS Winter Meeting Transaction, p. 348, 1963.

17) Christensen, J. A., R. J. Allio and A. Blancheria, “Melting Point of Irradiated Uranium Dioxide,” WCAP-6065, February 1965.

18) Tong, L.S., “Prediction of Departure from Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution,” Journal of Nuclear Energy, Vol. 21, pp. 241-248, 1967.

19) Ferrai, H. M., et al., “Use of Internally Pressurized Fuel Rods in Westinghouse Pressurized water Reactors,” WCAP-9002, Westinghouse Proprietary.

20) Tong, L. S., H. B. Currin and A. G. Throp II, “New Correlations Predict DNB Conditions,” Nucleonics, May 1963, and WCAP-1997 (1963).

21) Judd, D. F., et al, “Non-Uniform Heat Generation Experimental Program,” BAW-3238-7, 1965.

22) Lee, D. H., J. D. Obertelli, “An Experimental Investigation of Forced Convection Burnout in High Pressure Water, Part II, Preliminary Results for Round Tubes With Non-Uniform Axial Heat Flux Distribution,” AEEW-R-309, Winfrith, England, 1963.

23) Lee, D. H., “An Experimental Investigation of Forced Convection Burnout in High Pressure Water, Part IV, Large Diameter Tubes at About 1600 psi,” AEEW-R-479, Winfrith, England, 1966.

24) Ledinegg, M., Die Warme, No. 61, Vol. 48, p. 891-898, 1938.

25) Quandt, E. R., “Analysis of Parallel Channel Transient Response and Flow Oscillations,” WAPD-AD-TH-489, 1959.

26) Quandt, E. R., “Analysis and Measurement of Flow Oscillations,” Chemical Engineering Progress Symposium Series, Vol. 57, No. 32, p 111, 1961.

27) Meyer, J. E. Rose, R. P., Journal of Heat Transfer, Vol. 85, No. 1, p. 1, 1963.

28) Tong, L.S., et al., “HYDNA Digital Computer Program for Hydrodynamic Transient,” CVNA-77, 1961.

29) Weisman, J., A. H. Wenzel, et al., “Experimental Determination of the Departure from Nucleate Boiling in Large Rod Bundles at High Pressure,” American Institute of Chemical Engineers, Preprint 29, 9th National Heat Transfer Conference, 1967, Seattle, Washington.

30) Tong, L.S., H. Chelemer, et al., “Critical Heat Flux (DNB) in Square and Triangular Array Rod Bundles,” JSME, Semi-International Symposium, Paper No. 256, 1967, Tokyo, Japan.

31) Weisman, J., P. E. MacDonald, et al., “Fission Gas Release from UO2 Fuel Rods with Time Varying Power Histories,” Transactions of the American Society, No. 12, p. 900-901, 1969.

32) Tong, L. S., “Boiling Crisis and Critical Heat Flux,” AEC Critical Review Series, TID-25887, 1972.

33) Motley, F. E. and F. F. Cadek, “Application of Modified Spacer Factor to L-Grid Typical and Cold Wall Cell DNB,” WCAP-7988 (Westinghouse Proprietary), and WCAP-8030-A (Non-Proprietary), October 1972.

34) Tong, L. S., R. P. Sandberg and A. A. Bishop, “Forced Convection Heat Transfer at High Pressure after the Critical Heat Flux,” ASME 65-HT-31, 1965.

35) Hill, K. W., F. E. Motley, and F. F. Cadek, “Effect of Local Heat Flux Spikes on DNB in Non-Uniform Heated Rod Bundles, ” WCAP-8174 (Proprietary), August 1973, and WCAP-8202 (Non-Proprietary), August 1973.

36) Hetsroni, G., “Hydraulic Tests of the San Onofre Reactor Model,” WCAP-3269-8, June 1964

37) Hetsroni, G., “Studies of the Connecticut-Yankee Hydraulic Model,” NYO-3250-2, June 1965.

38) Carter, F. D., “Inlet Orificing of Open PWR Cores,” WCAP-7836, January 1972.

39) Shefcheck, J., “Application of the THINC Program to PWR Design,” WCAP-7359-L (Proprietary), August 1969, and WCAP-7838 (Non-Proprietary), January 1972.

40) Cadek, F. F., “Interchannel Thermal Mixing with Mixing Vane Grids,” WCAP-7667-P-A (Proprietary), and WCAP-7755-A (Non-Proprietary), January 1975.

41) Idel’chik, I.E., “Handbook of Hydraulic Resistance,” AEC-TR-6630, 1960.

42) Moody, L. F., “Friction Factors for Pipe Flow,” Transaction of the American Society of Mechanical Engineers, No 66, p. 671-684, 1944.

43) Daniel, R. C., et al., “Effects of High Burnup on Zircaloy-Clad Bulk UO2, Plate Fuel Element Samples,” WAPD-263, September 1965.

44) Large Closed Cycle Water Reactor Research and Development Program Quarterly Progress Reports for the Period January 1963 through June 1965 (WCAP-3738, 3739, 3743, 3750, 3269-2, 3269-5, 3269-6, 3269-12 and 3269-13).

45) “Nuclear Design of Westinghouse PWR’s with Burnable Poison Rods,” WCAP-9000, Westinghouse Proprietary, March 1969.

46) “Use of Part Length Absorber Rods in Westinghouse Pressurized Water Reactors,” WCAP-7072.

47) Deleted

48) “Improved Analytical Models Used in Westinghouse Fuel Rod Computations,” WCAP-8785, October, 1976.

49) “Fuel Densification Experimental Results and Model for Reactor Operation,” WCAP-8219-A, March 1975.

50) “Safety Related Research and Development for Westinghouse Pressurized Water Reactors – Program Summaries, Fall 1974,” WCAP-8485, Fall 1974, Westinghouse Electric Corp.

51) Skaritka, J., J.A. Iorii, “Operational Experience Wita Westinghouse Cores,” WCAP-8183, Revision 10, Westinghouse Electric Corp., May 1981.

52) “Four Loop PWR Internals Assurance And Test Program,” WCAP-7879, Class 2 (Proprietary), July 1972.

53) Deleted

54) “Reload Transition Safety Report for Indian Point 3” (Proprietary), July 1985, Westinghouse Electric Corporation.

55) “Westinghouse Wet Annular Burnable Absorber Evaluation Report,” WCAP-10021, (Revision 1) (Proprietary), October 1982, Westinghouse Electric Corporation.

56) “Supplemental Nuclear Analysis Report for the Existing Spend Fuel Storage Racks at Indian Point Unit No. 3”, Report No. 037-1-386, (Revision No. 0) April 1986, Northeast Technology Corporation.

57) IP3 Docket Number 50-286-Amendment No. 75 – Change to Technical Specifications Requirement for Control Bank Insertion Limit, dated 6/8/87.

58) F. E. Motley, K. W. Hill, F. F. Cadek, and J. Shefcheck, “New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane – Grids,” WCAP- 8762-P.A. July 1984.

59) Letter from A. C. Thadani (NRC) to W. J. Johnson (Westinghouse), Jan. 31, 1989, Subject: Acceptance for Referencing of Licensing Topical Report, WCAP-9226-P/9227-NP, “Reactor Core Response to Excessive Secondary Steam Releases.”

60) “DNB Test Results for New Mixing Vane Grid” WCAP-7958-A, F. F. Cadek, F. E. Motley.

61) F. F. Cadek, F. E. Motley and D. P. Dominicis, “Effect of Axial Spacing on Interchannel Thermal Mixing with the R. Mixing Vane Grid,” WCAP-7941-L, June, 1972, (Westinghouse Proprietary), and WCAP-7959, October, 1972.

62) D. S. Rowe, C. W. Angle, “Crossflow Mixing Between Parallel Flow Channels During Boiling, Part II Measurement of Flow and Enthalpy in Two Parallel Channels.” BNWL-371, part 2, December, 1967.

63) D. S. Rowe, C. W. Angle, “Crossflow Mixing Between Parallel Flow Channels During Boiling, Part III Effect of Spacers on Mixing Between Two Channels,” BNWL-371, part 3, January, 1969.

64) J. M. Gonzlez-Santalo and P. Griffith, “Two-Phase Flow Mixing in Rod Bundle Subchannels,” ASME Paper 72-WA/NE-19.

65) Indian Point 3, Docket Number 50-286, Amendment No. 90.

66) “Safety Evaluation by the Office of Nuclear Regulation Related to Amendment No. 94 Facility Operating License No. NPF-4 Virginia Electric and Power Company Old Dominion Electric Cooperative Noth Anna Power Station, Unit No 1 Docket No. 50-338,” dated May 13, 1987.

67) Davison, S. L., and Nuhfer, D. L. (Ed.), “VANTAGE + Fuel Assembly Report,” WCAP-12610, June 1990.

68) Letter from A. C. Thadani (NRC) to S. R. Tritch (Westinghouse), “Acceptance for Referencing of Topical Report WCAP-12610 ‘VANTAGE + Fuel Assembly Reference Core Report,” TAC. NO. 77258, July 1, 1991.

69) Davidson, S. L. (Ed.), “VANTAGE 5 Fuel Assembly Reference Core Report,” WCAP-10444-P-A (Proprietary), September 1985.

70) Weiner, R. A., et al., “Improved Fuel Rod Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations,” WCAP-10851-P-A (Proprietary), August 1988.

71) SECL-92-099, “Absence of Fuel Assembly Upper Alignment Pins at Core Locations A5, A6, A11 and B13 for Cycle 9, Westinghouse Electric Corporation,” June 1992.

72) Barnett, N. et al., “Nuclear Parameters and Operations Package for Indian Point Unit 3,” Cycle 16, WCAP-17080-P, March 2009.

73) Vantage 5 Reload Transition Safety Report for the Indian Point Unit 3 Nuclear Station, October 1988, Westinghouse.

74) Reload Transition Safety Report for the Indian Point Unit 3 Nuclear Station Vantage+ Fuel Upgrade, Revision 3, January 1997, Westinghouse.

75) Cycle 10 Margin Verification for RPI T/S Relaxation, 97IN-G-0011, January 1997, Westinghouse.

76) Nuclear Safety Evaluation Number NSE-98-3-138-RCS, Rev. 1, "Core Reload for Cycle 11."

77) Stewart, C. W. et al., "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores." Volumes 1-3 (Revision 3, August 1989), Volume 4 (April 1987), NP-2511-CCM-A, Electric Power Research Institute.

78) Sung, Y. et. al., "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal Hydraulic Safety Analysis," WCAP-14565-P-A/WCAP-15306-NP-A, October1999.

79) Frieldand, A.J. and Ray, S. "Revised Thermal Design Procedure", WCAP 11397-P-A (Proprietary), and WCAP 11397-A (Non-Proprietary), April 1989.

80) Slagle, W. H. (editor) et al. "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," WCAP-15063-P-A, Revision 1 with Errata (Proprietary), and WCAP-15064-NP-A, Revision 1 with Errata (Non-Proprietary). July 2000.

81) Letter from H.A. Sepp (Westinghouse) to T. E. Collins (NRC). "Summary of March 17, 1999. Meeting with Westinghouse and New York Power Authority Regarding Results of Westinghouse 15x15 Fuel DNB Testing", March 29, 1999.

82) Deleted

83) Davidson, S.L. et al., "Assessment of Clad Flattering and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel." WCAP-14297-A, March 1995

84) Liu, Y. S. et al., "ANC: A Westinghouse Advanced Nodal Computer Code." WCAP-10965-P-A. September 1986.

85) Westinghouse letter NF-IN-09-4 Rev. 1, "Indian Point Unit 3 Cycle 16 Final RSE and COLR," Westinghouse (D. Sipes) to Entergy (J. Willoughby).

TABLE 3.2-1

NUCLEAR DESIGN DATA
(For Cycle 1 and Cycle 16

		Cycle 1	Cycle 16
STRUCTURAL CHARACTERISTICS			
1.	Fuel Weight (UO_2), MTU	88.60	87.6
2a.	Zircaloy Weight, lbs	44,450	N/A
2b.	ZIRLO™ Weight, lbs	N/A[2]	< 41,000 (per design basis assumption of hydrogen release)
3.	Core Diameter, inches	132.71	Same as Cycle 1
4.	Core Height, inches	144	Same as Cycle 1
	Reflector Thickness and Composition		
5.	Top – Water Plus Steel	~10 inches	Same as Cycle 1
6.	Bottom – Water Plus Steel	~10 inches	Same as Cycle 1
7.	Side – Water Plus Steel	~ 15 inches	Same as Cycle 1
8.	H_2O/U, (cold) Core (volume)	3.99	N/C[1]
9.	Number of Fuel Assemblies	193	Same as Cycle 1
10.	UO_2 Rods per Assembly	up to 204	204
PERFORMANCE CHARACTERISTICS			
11.	Heat Output, MWt (initial rating)	3025	3216
12	Heat Output, MWt (maximum calculated turbine rating)	3216	3216
13.	Fuel Burnup, MWD/MTU	17,346	24,222 (predicted)

[1] Not Calculated for Cycle 16
[2] Not Applicable to Cycle 1

TABLE 3.2-1
(Cont.)

NUCLEAR DESIGN DATA
(For Cycle 1 and Cycle 16)

		Cycle 1	Cycle 16
	Enrichments, w/o[3]		
14.	Region 1	2.28	(16A) 4.599
15.	Region 2	2.80	(17A) 4.598 (17B) 4.951
16.	Region 3	3.30	(18A) 4.603 (18B) 4.949
17.	Equilibrium Enrichment	3.20	4.41
18.	Nuclear Heat Flux Hot Channel Factor, F_Q^N	2.56[4]	2.50
19.	Nuclear Enthalpy Rise Hot Channel Factor, $F^N{}_{\Delta H}$	1.55	1.70
CONTROL CHARACTERISTICS			
	Effective Multiplication (Beginning of Life) With Rods in; No Boron		
20.	Cold, No Power, Clean	1.197	N/C[5]
21.	Hot, No Power, Clean	1.144	N/C[5]
22.	Hot, Full Power, Clean	1.131	N/C[5]
23.	Hot, Full Power, Xe & Sm Equilibrium	1.091	N/C[5]
24.	Material	5% Cd; 15% In; 80% Ag	Same as Cycle 1
25.	Full Length	53	Same as Cycle 1

[3]Cycle 16 Blankets: Region 16 and 17 are 3.2 w/o. Region 18 is 3.6 w/o top, 3.2 w/o bottom. Region 17 & 18 include mixed blankets (some solid pellets, some annular)
[4] Nuclear peaking factor limits were revised in response to ACRS concerns by a generic peaking factor envelope for Cycle 1 operation.
[5] Not Calculated for Cycle 16

TABLE 3.2-1
(Cont.)

NUCLEAR DESIGN DATA
(For Cycle 1 and Cycle 16)

		Cycle 1	Cycle 16
26.	Partial Length	Removed	None
27.	Number of Absorber Rods per RCC Assembly	20	Same as Cycle 1
28.	Total Rod Worth, BOL, %	See Table 3.2-2	See Table 3.2-2
29.	Fuel Loading Shut down; Rods in (k = 0.85)	2000 ppm	N/C[6]
	(k = 0.90)	1810 ppm	N/C[6]
	(k = 0.95)	N/A[7]	1651 ppm[8]
30.	Shutdown (k=0.99) with Rods Inserted, Clean, cold	1000 ppm	N/C[6]
31.	Shutdown (k=0.99) with Rods Inserted, Clean, Hot	548 ppm	N/C[6]
32.	Shutdown (k=0.99) with No Rods Inserted, Clean, Cold	1500 ppm	N/C[6]
33.	With No Rods Inserted, Clean, Hot to Maintain k=1 at HFP, BOL	1476 ppm	1260 ppm
34.	Clean	1228 ppm	N/C[6]
35.	Xenon Eq.	934 ppm	883 ppm (150 MWD)
36.	Xenon and Samarium, Eq. (1000 MWD)	899 ppm	933 ppm
37.	Shutdown, All But One Rod Inserted, Clean, Cold	1099 ppm (k=0.99)	1147 ppm (k = 0.987, BOL)
38.	Shutdown, All But One Rod Inserted, Clean, Hot	669 ppm (k=.99)	662 ppm (k=.987, BOL)

[6] Not Calculated for Cycle 16
[7] Not Applicable to Cycle 1
[8] Calculated Value – Actual value used is 2050 ppm per the COLR.

TABLE 3.2-1
(Cont.)

NUCLEAR DESIGN DATA
(For Cycle 1 and Cycle 16)

		Cycle 1	Cycle 16
BURNABLE POISON RODS			
39.	Number and Material	1434 Borosilicate Glass	128 Unclad Hafnium 1136 WABA 11908 IFBA
40.	Worth Hot, !p	10.0%	N/C[1]
41.	Worth Cold, !p	8.0%	N/C[1]
KINETIC CHARACTERISTICS			
42.	Moderator Temperature Coefficient at Hot Full Power, pcm/°F	-3 to -35	11.0 to -31.9
43.	Moderator Pressure Coefficient, psi^{-1}	0.3×10^{-6} to 4.0×10^{-6}	N/C[1]
44.	Moderator Density Coefficient, $\Delta k \cdot cm^3/gm$	-0.1 to 0.47	N/C[1]
45.	Doppler Coefficient, pcm/°F	-1.0 to -2.0	-1.63 (BOL, HZP)
46.	Delayed Neutron Fraction, %	0.44 to 0.72	0.52 to 0.64
47.	Prompt Neutron Life, seconds	1.4×10^{-5} to 2.6×10^{-5}	1.18×10^{-5} to 1.44×10^{-5}

[1] Not Calculated for Cycle 16

TABLE 3.2-2

REACTIVITY REQUIREMENTS FOR CONTROL RODS
(For Cycle 1 and Cycle 16)

Requirement	Percent Δp Beginning of Life[10]	Percent Δp End of Life
CYCLE 1		
Control Banks		
Total Control	2.31	3.43
Shutdown Banks		
Shutdown	1.00	1.72
Total Requirement	3.31	5.15
CYCLE 16		
Control Groups		
Total Control Bank Requirement (1)	2.02	3.01
Control Rod Worth (HZP)		
All Rods Inserted Less Most Reactive Stuck Rod[11]	5.33	5.53
(2) Less 10%	4.80	4.98
Shutdown Margin		
Calculated Margin [(2)-(1)]	2.78	1.97
Required Shutdown Margin	1.3	1.3

Cycle 16 Note:
A) Rod worths are calculated at ARO, HFP boron concentration.
B) T_{mod} at 547.0°F at HZP and 569.0°F at HFP. (Nominal vessel Tavg)

[10] 150 MWD/MTU.
[11] N-11 is the most reactive stuck rod at BOL (0.95%Δp) and K-10 is the most reactive stuck rod at EOL (0.76%Δp).

TABLE 3.2-3

CALCULATED ROD WORTHS, Δp
(For Cycle 1 and Cycle 16)

CYCLE 1

Core Condition	Rod Configuration	Worth*	Less 10%**	Design Reactivity Requirements	Shutdown Margin
BOL, HFP	53 Rods in	9.76%			
BOL, HZP	52 Rods in; Highest Worth Rod Stuck Out	7.75%	6.97%	2.31%	4.66%
EOL, HFP	53 Rods in	9.55%			
EOL, HZP	52 Rods in; Highest Worth Rod Stuck Out	7.57%	6.81%	3.43%	3.38%

CYCLE 16

Core Condition	Rod Configuration	Worth*	Less 10%**	Reactivity Requirements	Shutdown Margin
BOL, HFP	53 Rods in	7.73%			
BOL, HZP	52 Rods in; Highest Worth Rod Stuck Out	5.33%	4.80%	1.30%	2.78%
EOL, HFP	53 Rods in	8.48%			
EOL, HZP	52 Rods in; Highest Worth Rod Stuck Out	5.53%	4.98%	1.30%	1.97%

Legend

BOL = Beginning-of-Life
EOL = End-of-Life
HFP = Hot Full Power
HZP = Hot Zero Power

* The values for worth are for rods at the insertion limit.
** Calculated rod worth is reduced by 10% to allow for uncertainties.

TABLE 3.2-4

THERMAL AND HYDRAULIC DESIGN PARAMETERS
(For Cycle 1 and Cycle 16)

	Cycle 1	**Cycle 16**
Total Heat Output, MWt	3025	3216
Total Heat Output, Btu/hr	$10{,}324 \times 10^6$	10973×10^6
Heat Generated in Fuel, %	97.4	Same as Cycle 1
Nominal System Pressure, psia	2250	Same as Cycle 1
Hot Channel Factors		
Heat Flux		
Engineering, F_q^E	1.03	Same as Cycle 1
Total, F_q^{T*}	2.32	2.50
Enthalpy Rise – Nuclear $F_{\Delta H}^N$	1.55	1.70
Total Flow Rate, lbm/hr	130.2×10^6	134.8×10^6
Average Velocity Along Fuel Rods, ft/sec	14.9	14.2
Average Mass Velocity, lbm/hr•ft^2	2.43×10^6	2.42×10^6

*The total heat flux hot channel factor is a generic limit. The actual value is documented in the COLR.

TABLE 3.2-4
(Cont.)

THERMAL AND HYDRAULIC DESIGN PARAMETERS
(For Cycle 1 and Cycle 16)

	Cycle 1	Cycle 16
Coolant Temperature, °F		
Nominal Inlet	541.2	541.0
Average Rise in Vessel	60.5	62.0
Average Rise in Core	63.1	66.5
Average in Core	574.1	575.8
Average in Vessel	571.5	572.0
Nominal Outlet of Hot Channel	635.7	N/C*
Heat Transfer		
Active Heat Transfer Surface Area, ft^2	52,200	52,100
Average Heat Flux Btu/hr•ft^2	193,000	205,200
Maximum Heat Flux, Btu/hr•ft^2	448,000	513,000
Maximum Thermal Output, $^{kw}/_{ft}$	12.7	16.6
Maximum Clad Surface Temperature BOL at Nominal Pressure, °F	657	N/C*
Maximum Average Clad Temperature BOL at Rated Power, °F	715	N/C*

*Not Calculated for Cycle 16.

TABLE 3.2-4
(Cont.)

THERMAL AND HYDRAULIC DESIGN PARAMETERS
(For Cycle 1 and Cycle 16)

	Cycle 1	Cycle 16
Fuel Central Temperatures for nominal Fuel rod dimensions, °F		
Maximum at 100% Power	4100	3670*
Maximum at Overpower Power	4350 (112% power)	4250* (120% power)
DNB Ratio		
Minimum DNB Ratio at nominal operating Conditions	1.80	2.50
Pressure Drop, psi		
Across Core	21	26

*The fuel central temperatures were calculated using the new PAD model (Ref. 80)

TABLE 3.2-5

CORE MECHANICAL DESIGN PARAMETERS(1)
(For Cycle 1 and Cycle 16)

	Cycle 1	Cycle 16
Active Portion of the Core		
Equivalent Diameter, inches	132.7	Same as Cycle 1
Active Fuel Height, inches	144.0	Same as Cycle 1
Length-to-Diameter Ratio	1.09	Same as Cycle 1
Total Cross-Section Area, ft^2	96.06	Same as Cycle 1
Fuel Assemblies		
Number	193	Same as Cycle 1
Rod Array	15 x 15	Same as Cycle 1
Rods per Assembly	204*	Same as Cycle 1
Rod Pitch, inches	0.563	Same as Cycle 1
Overall Dimensions, inches	8.426 x 8.426	Same as Cycle 1
Fuel Weight, MTU	88.60	87.6
Number of Grids per Assembly	7	10
Number of guide Thimbles	20	Same as Cycle 1
Diameter of Guide Thimbles (upper part) inches	0.545 O.D. x 0.515 I.D.	0.533 O.D. x 0.499 I.D.
Diameter of Guide Thimbles (lower part) inches	0.484. O.D. x 0.454 I.D.	0.490 O.D. x 0.455 I.D.
Fuel Rods		
Number	39,372	Same as Cycle 1
Outside Diameter, inches	0.422	Same as Cycle 1
Diametral Gap, inches	0.0075	0.0075**
Clad Thickness, inches	0.0243	Same as Cycle 1
Clad Material	Zircaloy-4	ZIRLO™
Overall Length	151.8	156.3
Length of End Cap, overall, inches	0.688	0.450 (top), 0.680 (bottom)
Length of End Cap, inserted in rod, inches	0.250	0.13

NOTE: All dimensions are for cold conditions.
*Twenty-one rods are omitted: Twenty provide passage for control rods and one to contain in-core instrumentation.
**Nominal gap.

TABLE 3.2-5
(Cont.)

CORE MECHANICAL DESIGN PARAMETERS(1)
(For Cycle 1 and Cycle 16)

	Cycle 1	Cycle 16
Fuel Pellets		
Material	UO_2 sintered	Same as Cycle 1
Density (% of Theoretical)		
Region 1, 2, & 3		
(typical of reload regions)	95 (10.4 g/cc)	
Region 13A & 13B		95.5
Feed Enrichments w/o		
Region 1	2.28	(16A) 4.599
Region 2	2.80	(17A) 4.598
		(17B) 4.951
Region 3	3.30	(18A) 4.603
		(18B) 4.949
Diameter, inches		
Region 1, 2, & 3	0.3659	Same as Cycle 1
Length, inches	0.600	0.439
Rod Cluster Control Assemblies		
Neutron Absorber	5% Cd, 15% In, 80% Ag	Same as Cycle 1
Cladding Material	Type 304 SS-Cold Worked	Same as Cycle 1
Clad Thickness, inches	0.019	Same as Cycle 1
Number of Clusters		
Full Length	53	Same as Cycle 1
Number of Control Rods per Cluster	20	Same as Cycle 1
Length of Control Rod, inches	156.436 (overall)	Same as Cycle 1
	149.136 (insertion length)	Same as Cycle 1
Length of Absorber Section, inches	142.000	Same as Cycle 1

*Note: All dimensions are for cold conditions.

TABLE 3.2-5
(Cont.)

CORE MECHANICAL DESIGN PARAMETERS[1]
(For Cycle 1 and Cycle 16)

	Cycle 1	Cycle 16	
Core Structure			
Core Barrel, inches			
I.D.	148.0	Same as Cycle 1	
O.D.	152.5	Same as Cycle 1	
Thermal Shield, inches			
I.D.	158.5	Same as Cycle 1	
O.D.	164.0	Same as Cycle 1	
Burnable Poison Rods and Flux Suppressors			
Material	Borosilicate Glass	IFBA	
		WABA	
		Hafnium	
Number	1434 (first cycle)	11908	(IFBA)
	1056	1136	(WABA)
		128	(Hafnium)
		13172	Total
Outer Tube, inches	0.4390	N/A	(IFBA)
(outer diameter)		0.532	(WABA)
		0.533	(Hafnium)
Inner Tube, inches	0.2365	N/A	(IFBA)
(outer diameter)		0.381	(WABA)
		N/A	(Hafnium)
Clad Material	Type 304 SS	N/A	(IFBA)
		ZIRLO	(WABA)
		N/A	(Hafnium)
Inner Tube Material	Type 304 SS	N/A	(IFBA)
		ZIRLO	(WABA)
	Type 304 SS	N/A	(Hafnium)
Boron loading (natural) gm/cm	0.0576	0.001043	(IFBA)
		0.00603	(WABA)
		N/A	(Hafnium)

*Note: All dimensions are for cold conditions.

TABLE 3.2-6

NOMINAL COMPOSITION OF ZIRLO™ AND ZIRCALOY-4 CLADDING

Element	Zircaloy-4 (wt %)	ZIRLO™ (wt %)
Sn	1.6	1.0
Fe	0.21	0.1
Cr	0.1	0.0
Nb	0.0	1.0
Zr	>97.0	>97.0

3.3 TESTS AND INSPECTIONS

3.3.1 Physics Tests

1. Tests to Confirm Reactor Core Characteristics

   A detailed series of startup physics tests were performed from zero power up to and including 100% power. As part of these tests, a series of core power distribution measurements were made by means of the core movable detector system. These measurements were analyzed and the results compared with the analytical predictions upon which safety analyses were based.

2. Tests Performed During Operation

   To detect and 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, is normalized to accurately reflect actual core conditions. When full power is reached initially, and with the control groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation continues, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burnup and reactivity is compared with that predicted and corrected if necessary. This normalization is completed before a cycle burnup of 60 Effective Full Power Days (EFPD) is reached. Thereafter, actual boron concentration can be compared with the predicted concentration, and the reactivity prediction of the core can be continuously evaluated and adjusted.

   Any reactivity anomaly greater than one percent would be unexpected, and its occurrence would be thoroughly investigated and evaluated.

3.3.2 Thermal and Hydraulic Tests and Inspections

General hydraulic tests on models have been used to confirm the design flow distributions and pressure drops (1,2). Fuel assemblies and control and drive mechanisms are also tested in this manner. Appropriate on-site measurements are made to confirm the design flow rates.

Vessel and vessel internals inspections were also reviewed to confirm such thermal and hydraulic design values as bypass flow.

A summary report of appropriate plant testing shall be submitted to the NRC following (1), an amendment to the license involving a planned increase in power level, (2) installation of fuel that has a different design and (3) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the testing and comparison of these values with acceptance criteria.

Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup programs, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

3.3.3 Core Component Tests and Inspections

To ensure that all materials, components and assemblies conformed to the design requirements, a release point program was established with the manufacturer. This required surveillance of raw materials, special processes, i.e., welding, heat treating, nondestructive testing, etc. and those characteristics of parts which directly affect the assembly and alignment of the reactor internals. The surveillance was accomplished by the issuance of Inspection Reports by the quality control organization after conformance had been verified.

A resident quality control representative performed a surveillance/audit program at the manufacturer's facility and witnessed the required tests and inspections and issued the inspection reports.

Components and materials supplied by Westinghouse to the assembly manufacturer were subjected to a similar program. Quality Control engineers developed inspection plans for all raw materials, components, and assemblies. Each level of manufacturing was evaluated by a qualified inspector for conformance, i.e., witnessing the ultrasonic testing of core plate raw material. Upon completion of specified events, all documentation was 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 were taken to verify conformance to the design requirements and assure proper fitup between the reactor internals and the reactor pressure vessel.

3.3.3.1 Fuel Quality Assurance

[Deleted] Surveillance audits of the fuel fabrications and the Westinghouse Quality Assurance Program are performed by Entergy.

Quality Assurance Program

The Westinghouse Nuclear Fuel Division's quality assurance program plan is included in Reference 3 that is summarized below.

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 drawing and product, process, and material specifications identify the inspections to be performed.

Quality Control

Quality Control (QC) philosophy is 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 QC 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 QC.

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 taken 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) All weld enclosures are X-rayed or Ultrasonically tested. X-rays are taken in accordance with Westinghouse specifications meeting the requirements of ASTM-E-142.

   c) All rods are dimensionally inspected prior to final release. The requirements include such items as length, camber, and visual appearance.

   d) All of the fuel rods are inspected by X-ray or other approved methods to ensure proper plenum dimensions.

   e) All of the 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 QC.

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 Reference 3.

5) Other inspections:

The following inspections are performed as part of the routine inspection operation:

a) Tool and gage inspection and control, including standardization to primary and/or secondary working standards. Tool inspection is performed at prescribed intervals on all serialized tools. Complete records are kept of calibration and conditions of tools.

b) Audits are performed of inspection activities and records to ensure that prescribed methods are followed and that records are correct and properly maintained.

c) Surveillance inspection, where appropriate, and audits of outside contractors are performed to ensure conformance with specified requirements.

6) Process control:

To prevent the possibility of mixing enrichments during fuel manufacture and assembly, strict enrichment segregation and other process controls are exercised.

The UO_2 powder is kept in sealed containers. The contents are fully identified by labels completely describing the contents. 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 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 QC. Physical barriers prevent mixing of pellets of different nominal densities and enrichments in this storage area. Substandard pellets are reprocessed and utilized for recycle material.

A serialized traceability code label is electroetched into the tube, to provide unique identification. This identification is maintained and used throughout assembly fabrication and inspection.

Loading of pellets into the clad, which previously had bottom end plug welded in place, is performed in isolated production lines, and again only one density and enrichment are loaded on a line at a time except that the axial blanket pellets are usually a lower enrichment.

The “annular” axial blanket pellets are sintered with a hole in the center to increase plenum volume. The axial blanket pellets are carefully controlled on separate trays located in separate cabinets and have the added characteristic of the centerline hole to minimize improper placement in the pellets stacks.

The loading of IFBA burnable poison rods containing ZrB_2 coated pellets is performed in a separate production line physically separated from standard UO_2 production line. All other operations are similar with the same close control to prevent the misloading of pellets with different enrichment.

The top end plugs are inserted and then welded to seal the tube. At the time of installation into an assembly, a matrix is generated to identify each rod’s position within a given assembly. The top nozzle is inscribed with a permanent identification number providing traceability to the fuel contained in the assembly.

The preceding discussion and the references stated describe the efforts in the application of quality control and the use of reliability techniques during the development, design, fabrication, and shipment of fuel assemblies.

Upon delivery of the fuel to the site, an inspection is performed for evidence of shipping damage, loose parts and debris. The shipping container internals, seals, container bolts, clamping fixture, and shock overload probe indicators are inspected. After removal from the container, the fuel itself is inspected for evidence of shipping damage to approved Inspection Procedures. The above inspections, as a minimum, are accomplished for all fuel assemblies and are appropriately documented.

3.3.3.2 Burnable Poison Rod Tests and Inspections

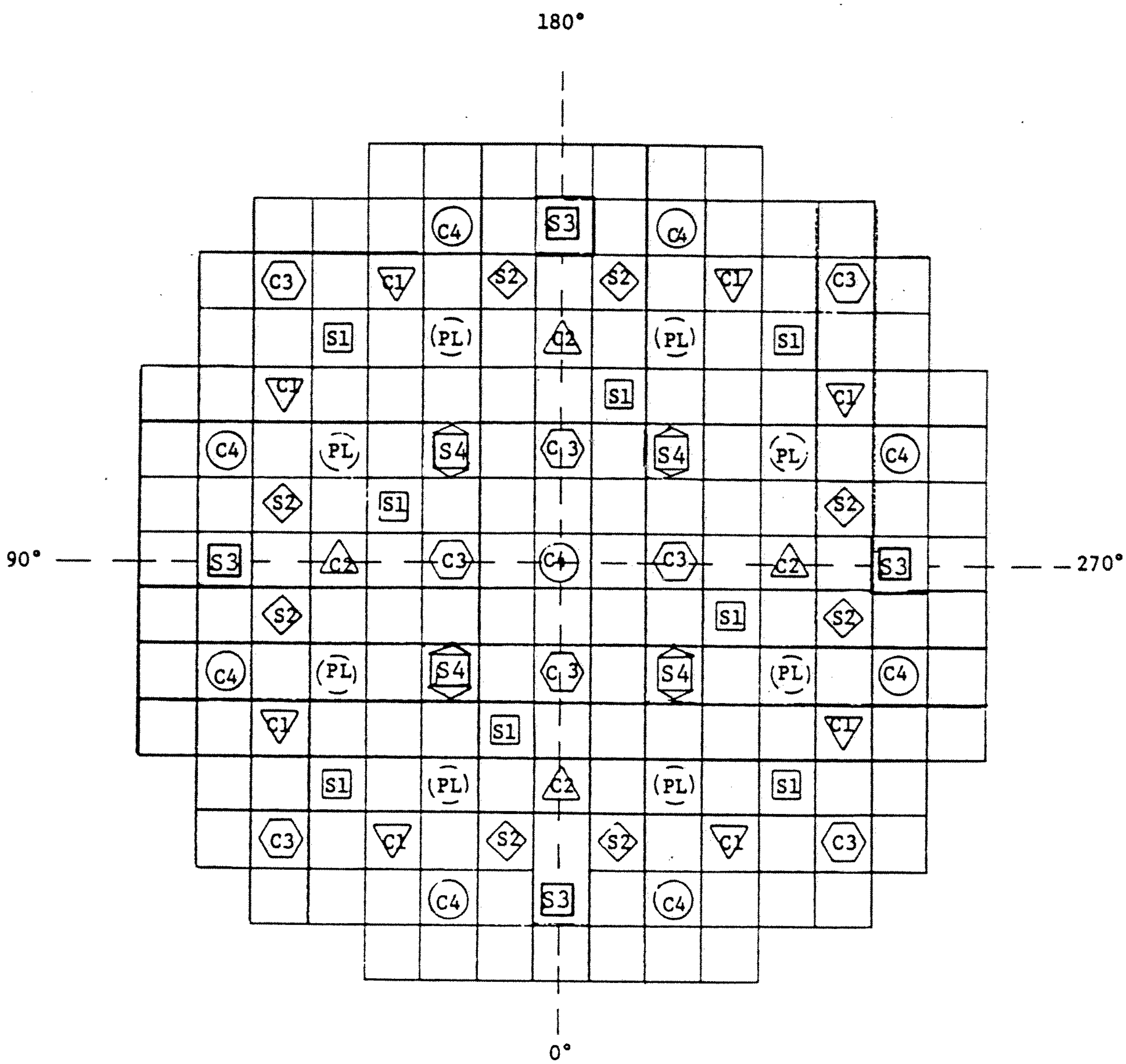
The end plug seal welds are checked for integrity by visual inspection, and X-ray or Ultrasonic testing. The finished rods are helium leak checked.

REFERENCES

1. Hetsroni, G., “Hydraulic Tests of the San Onofre Reactor Model,” WCAP-3269-8, 1964.

2. Hetsroni, G., “Studies of the Connecticut-Yankee Hydraulic Model,” WCAP-2761, 1965.

3. Moore, J., “Nuclear Fuel Division Quality Assurance Program Plan,” WCAP-7800-Revision 5, November 1979.



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 (Removed)
		61

INDIAN POINT 3		FSAR UPDATE
ROD CLUSTER CONTROL BANKS		
REV. 0	JULY, 1982	FIGURE NO. 3.2-1

INT, CYCLE 1

CL								CL
CL	1.255 X							CL
	1.159	1.236						
	1.230	1.123	1.152					
	1.134	1.188	.983	1.080				
	1.237	1.121	1.147	.957	1.206			
	1.205	1.201	1.147	1.083	.853	.976		
	1.072	.995	1.030	.980	.846	.468		
	.746	.688	.601					
CL								

CL - CENTER-LINE

(CYCLE 1)

$F^N_{\Delta H}$ = 1.35 at (x), HFP, NO XENON

INDIAN POINT 3		FSAR UPDATE
ASSEMBLYWISE AVERAGE POWER DISTRIBUTION BEGINNING OF LIFE, UNRODDED CORE (CYCLE 1)		
REV. 0	JULY, 1982	FIGURE NO. 3.2-2

INT, Cycle 1

CL

CL								CL
	1.08							
	1.11	1.08						
	1.08	1.12	1.09					
	1.12	1.09	1.10	1.11				
	1.09	1.12	1.10	1.11	X 1.20			
	1.13	1.07	1.13	1.07	1.02	1.03		
	1.00	1.03	.98	1.01	.88	.57		
	.76	.81	.73	.63				

CL - CENTER-LINE

(CYCLE 1)

$F^{N}_{\Delta H} = 1.30$ at (x), HFP

INDIAN POINT 3		FSAR UPDATE
ASSEMBLYWISE AVERAGE POWER DISTRIBUTION END OF LIFE, UNRODDED CORE (CYCLE 1)		
REV. 0	JULY, 1982	FIGURE NO. 3.2-3

INT, CYCLE 1

D .652							
1.077	1.283						
1.376	1.277	1.360				D	
1.318	1.394	1.155	1.268				
X 1.410	1.258	1.273	1.070	1.355			
1.270	1.207	1.043	1.072	.925	1.092		
1.046	.843	D .474	.796	.855	.517		
.693	.697	.462	.450				

CL - CENTER-LINE

D - BAND D LOCATIONS

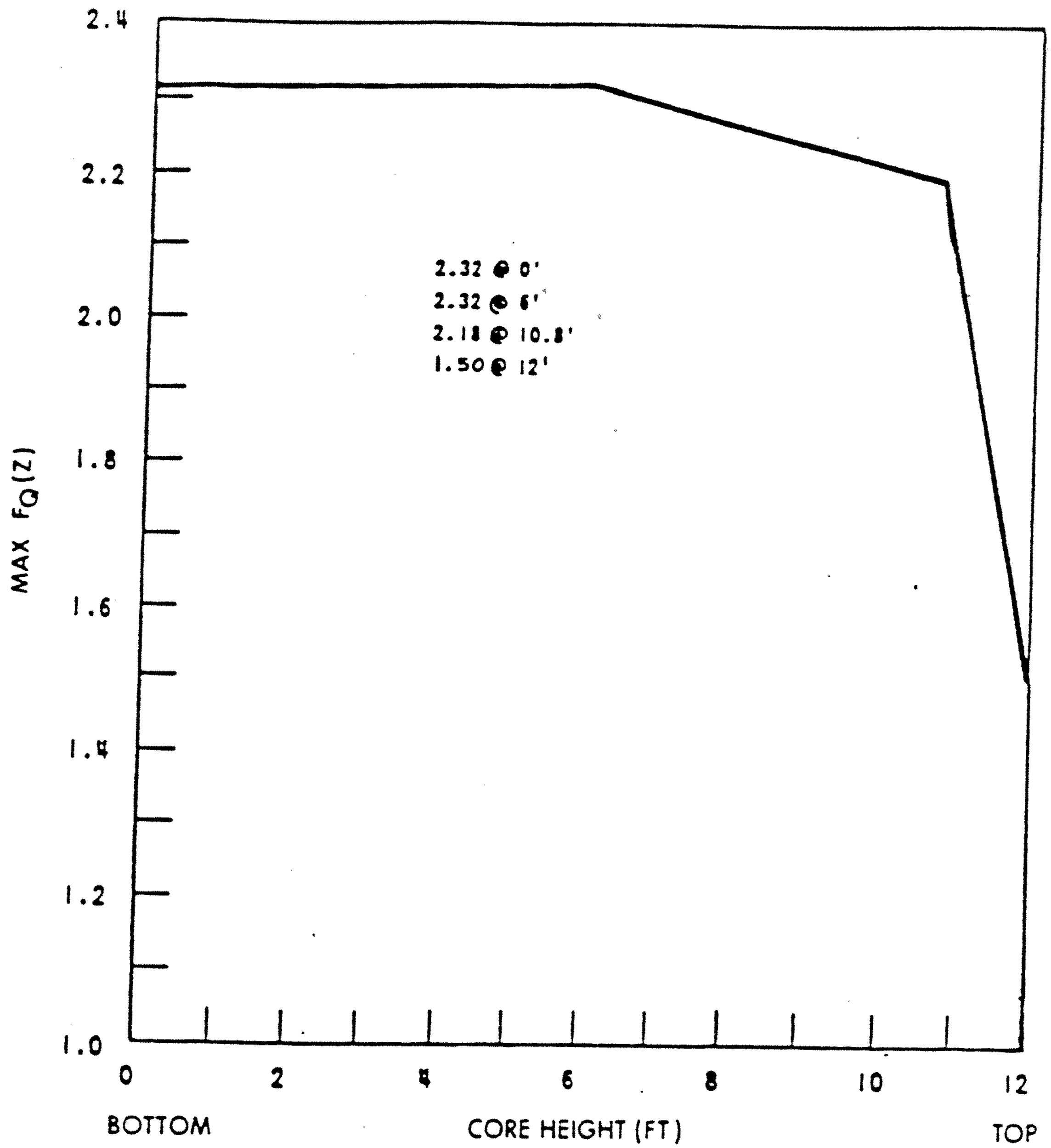
(CYCLE 1)

$F^N_{\Delta H}$ = 1.52 at (x), Equilibrium Xenon

INDIAN POINT 3 FSAR UPDATE

ASSEMBLYWISE AVERAGE POWER DISTRIBUTION BEGINNING OF LIFE, BANK D INSERTED (CYCLE 1)

REV. 0 JULY, 1982 FIGURE NO. 3.2-4



INDIAN POINT 3 FSAR UPDATE

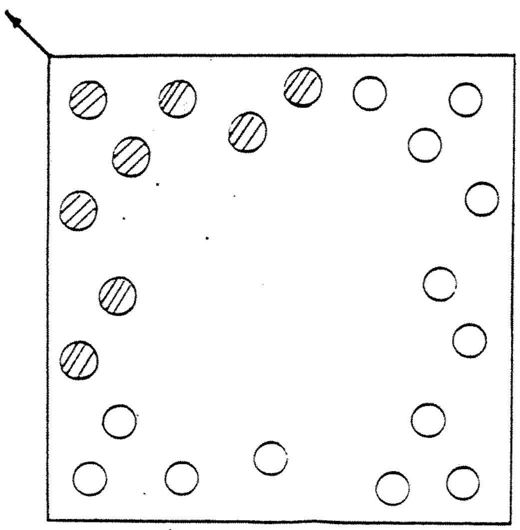
MAX FQ(Z) VS. AXIAL HEIGHT DURING NORMAL OPERATION (CYCLE 1)

REV. 1, JULY 1990 FIGURE NO. 3.2-5

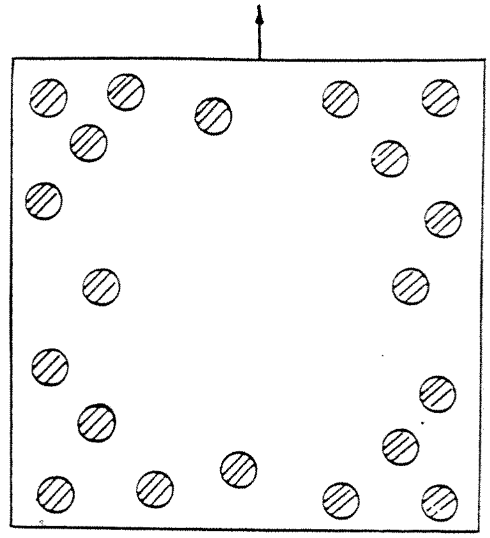
					9		9		9					
		8		12		20		20		12		8		
	8		20		12		12		12		20		8	
		20		20		16		16		20		20		
	12		20		16		16		16		20		12	
9		12		16		20		20		16		12		9
	20		16		20		16		20		16		20	
9		12		16		16		16		16		12		9
	19 S		16		20		16		20		16		19 S	
9		12		16		20		20		16		12		9
	12		20		16		16		16		20		12	
		20		20		16		16		20		20		
	8		20		12		12		12		20		8	
		8		12		20		20		12		8		
					9		9		9					

[1434 total 2 source rods (s)]

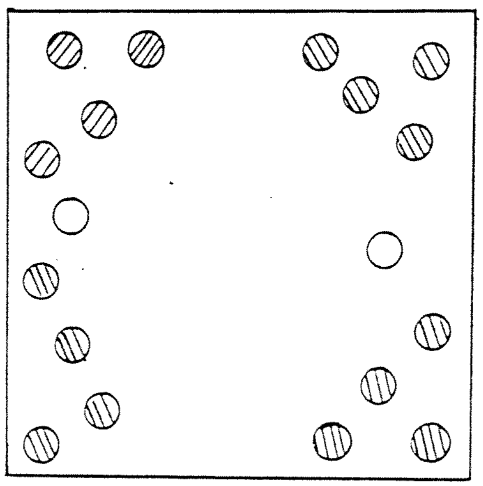
INDIAN POINT 3		FSAR UPDATE
DISTRIBUTION OF BURNABLE POISON RODS (CYCLE 1)		
REV. 0	JULY, 1982	FIGURE NO. 3.2-6



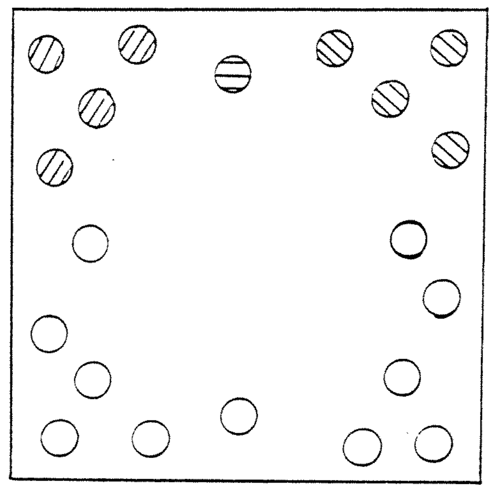
8 RODS



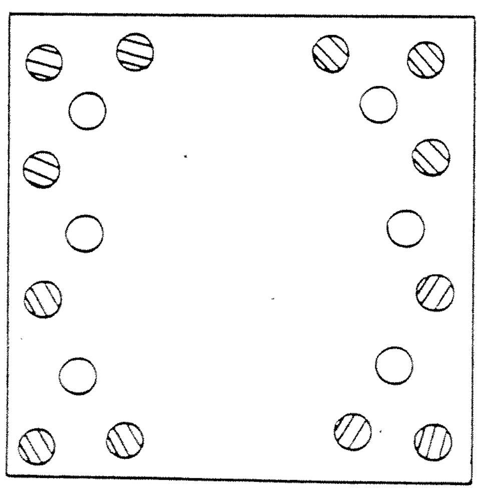
20 RODS



16 RODS

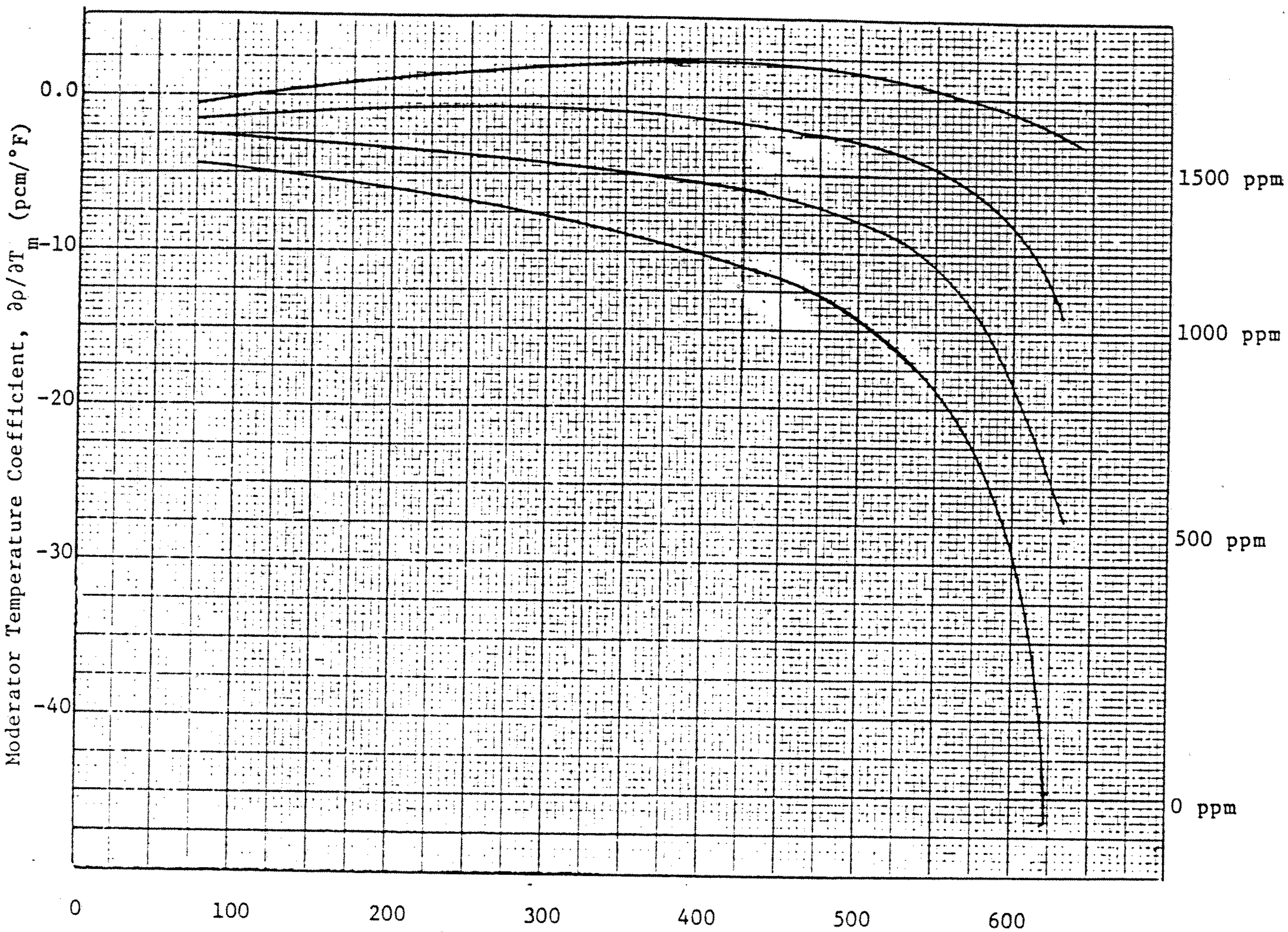


9 RODS



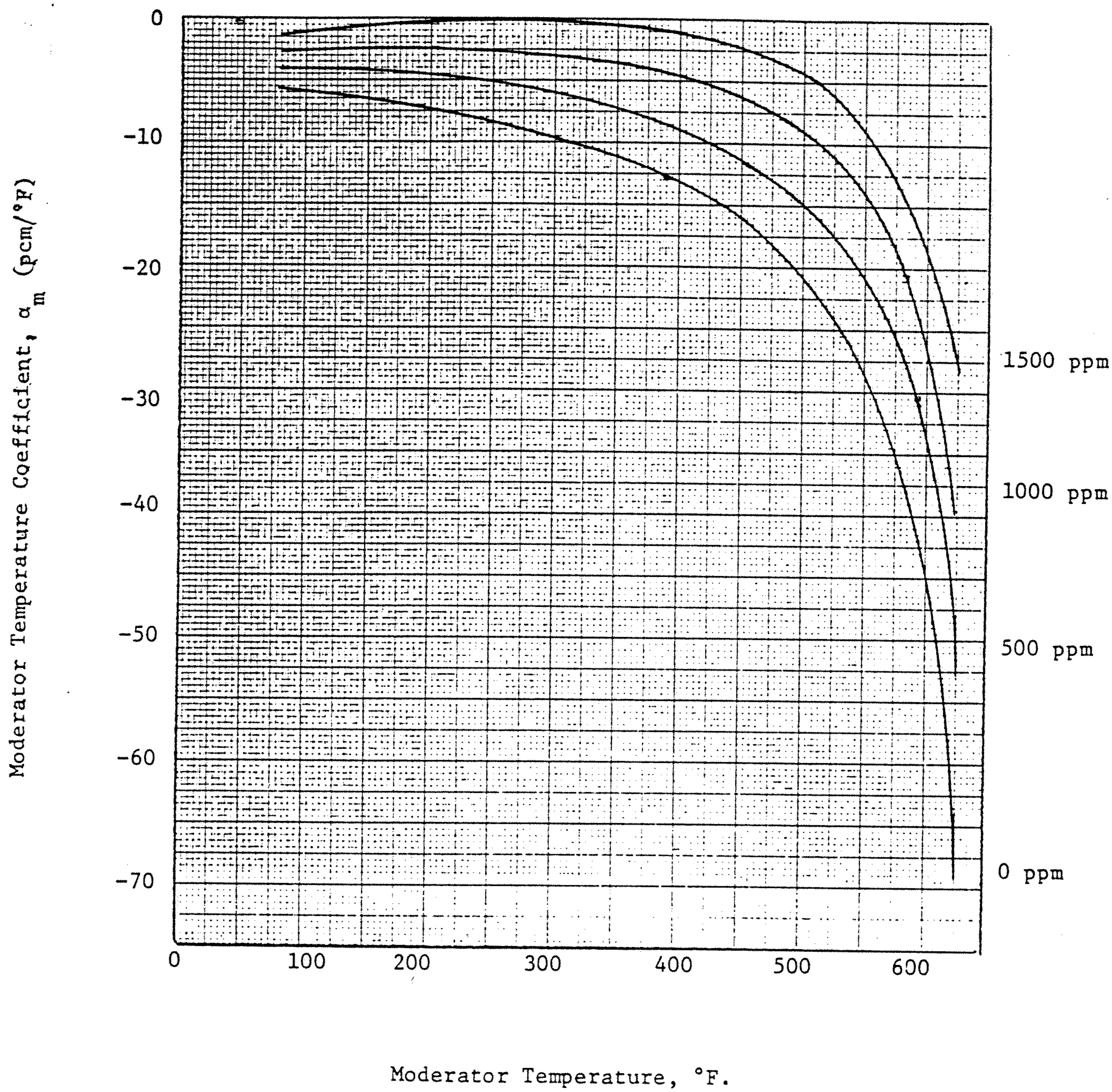
12 RODS

INDIAN POINT 3	FSAR UPDATE
ARRANGEMENT OF BURNABLE POISON RODS (CYCLE 1)	
REV. 0 JULY, 1982	FIGURE NO. 3.2-7

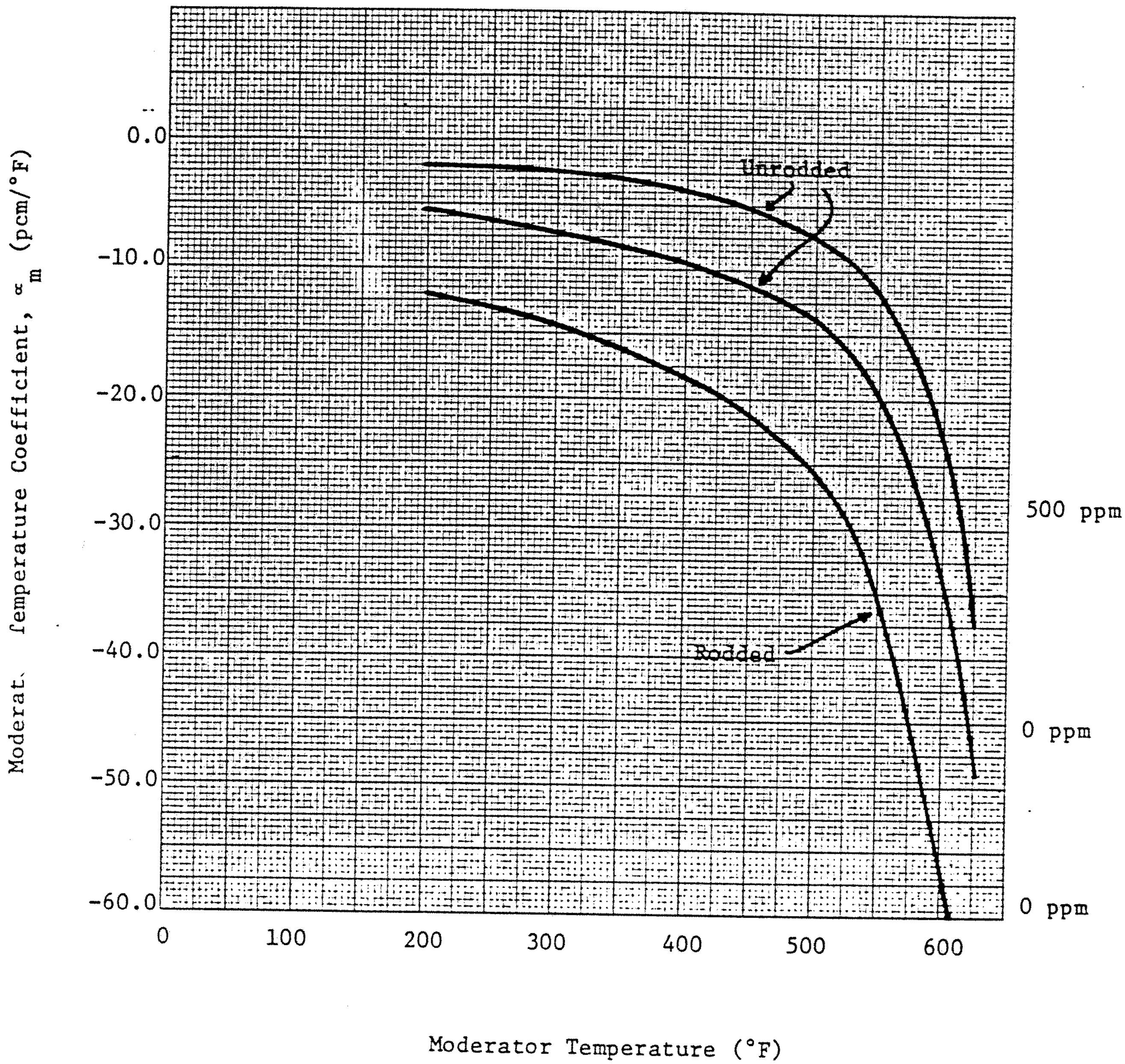


Moderator Temperature (°F)

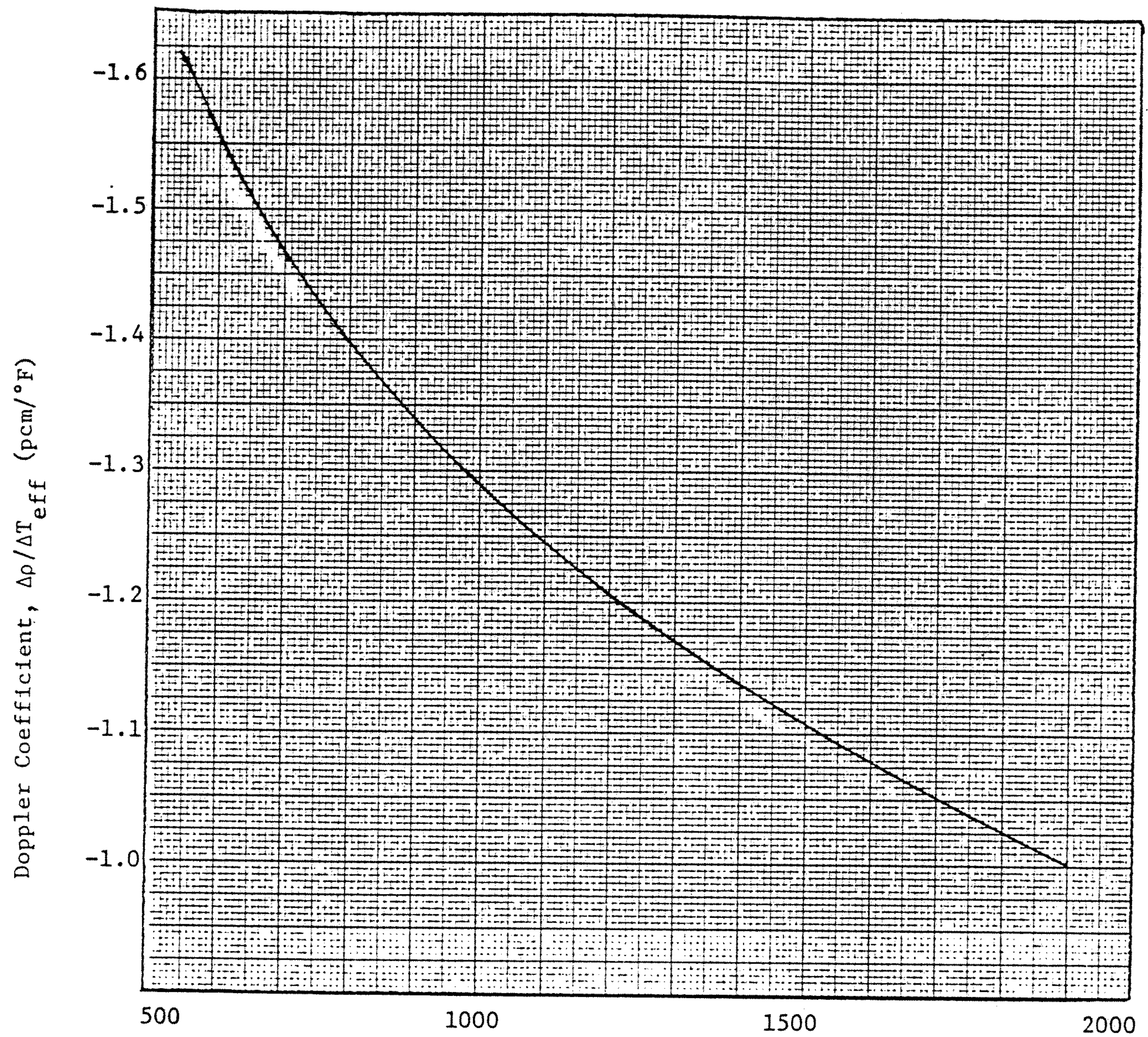
INDIAN POINT 3	FSAR UPDATE
MODERATOR TEMPERATURE COEFFICIENT VS. MODERATOR TEMPERATURE,BOL, CYCLE 1	
REV. 0 JULY, 1982	FIGURE NO. 3.2-8



INDIAN POINT 3		FSAR UPDATE
MODERATOR TEMPERATURE COEFFICIENT VS. MODERATOR TEMPERATURE,BOL, CYCLE 1 CONTROL RODS PRESENT		
REV. 0	JULY, 1982	FIGURE NO. 3.2-9



INDIAN POINT 3		FSAR UPDATE
MODERATOR TEMPERATURE COEFFICIENT VS. MODERATOR TEMPERATURE,EOL, CYCLE 1		
REV. 0	JULY, 1982	FIGURE NO. 3.2-10

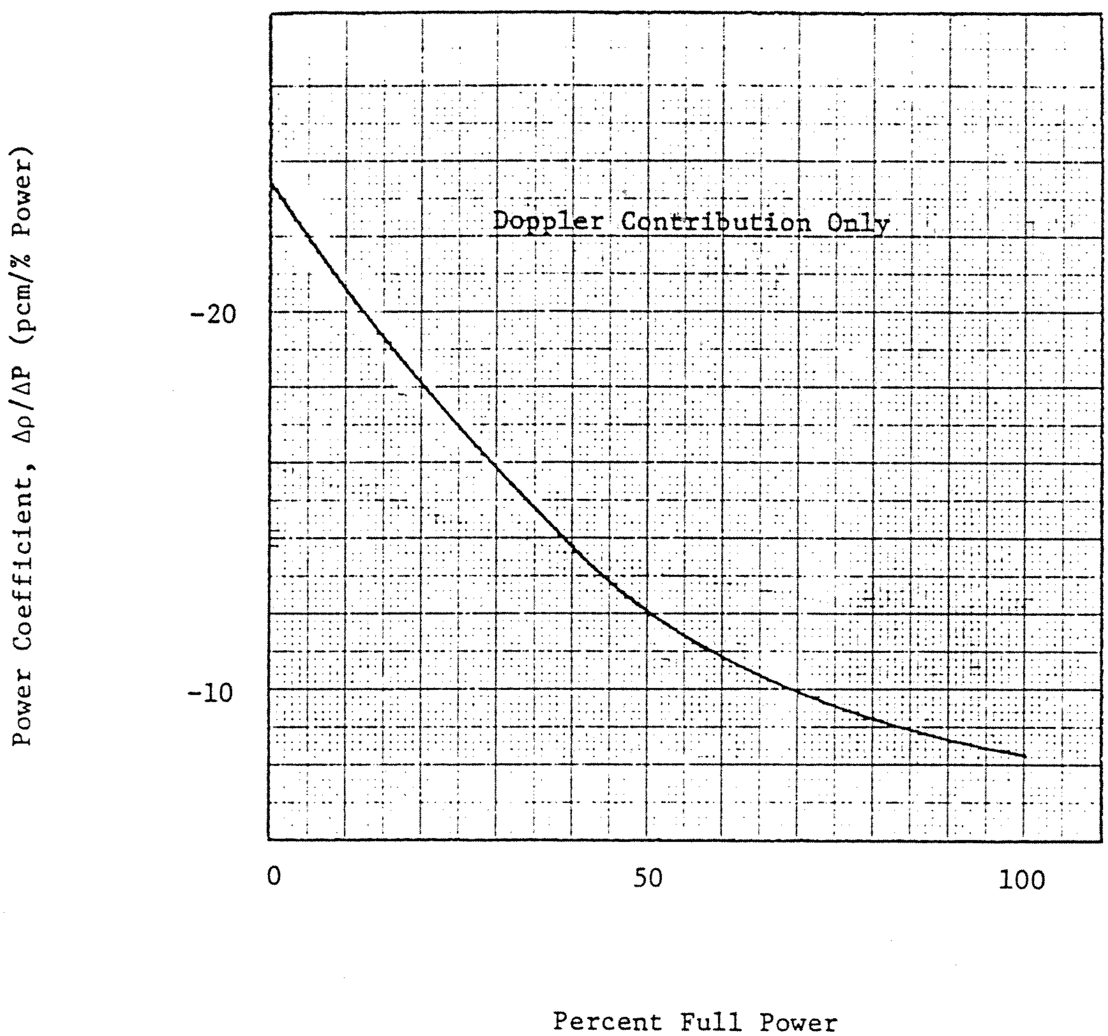


BOL, cycle 1

T_m = 574°F

Avg. Enrichment = 2.8

INDIAN POINT 3		FSAR UPDATE
DOPPLER COEFFICIENT VS. RESONANCE EFFECTIVE TEMPERATURE		
REV. 0	JULY, 1982	FIGURE NO. 3.2-11

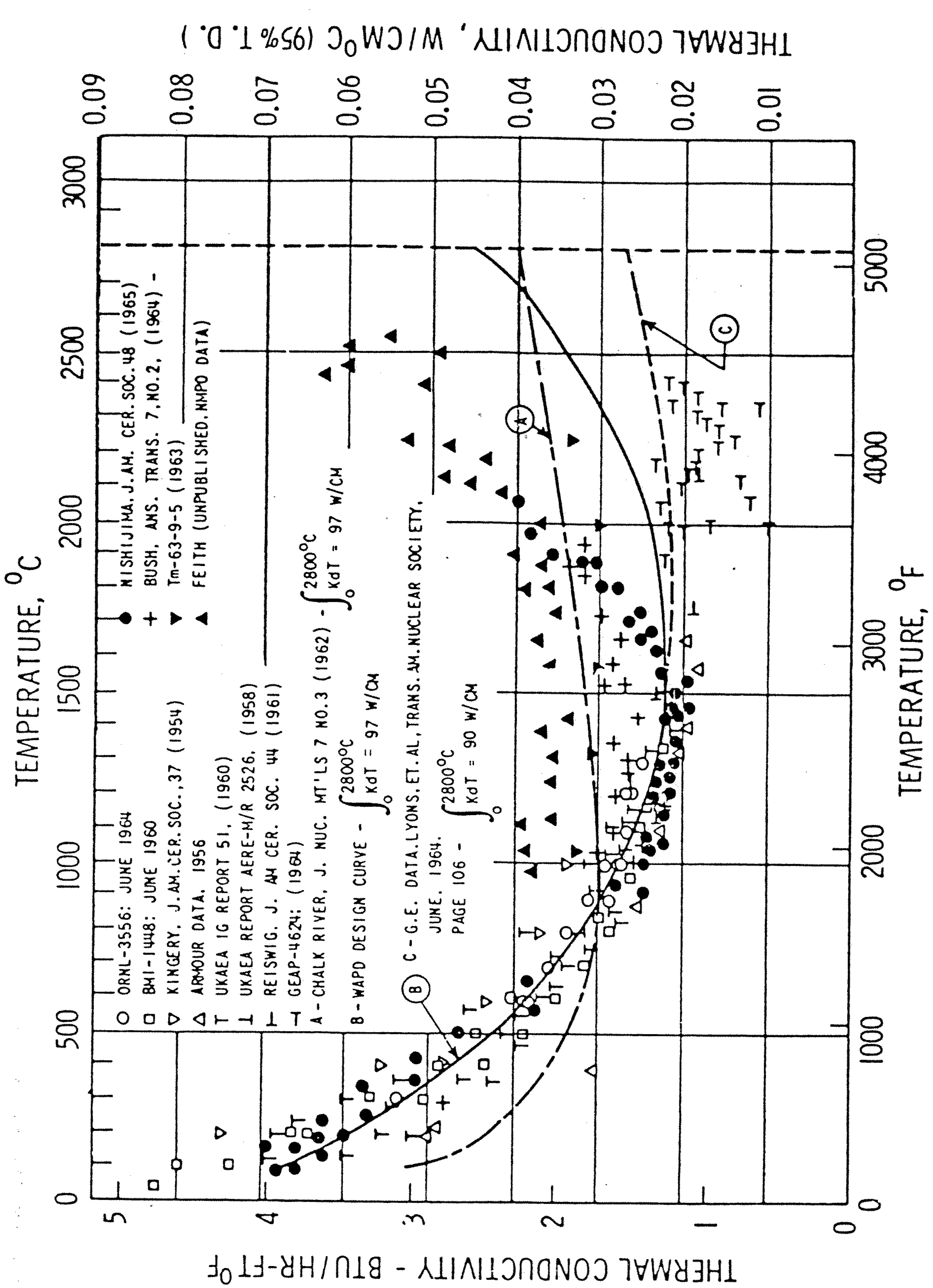


BOL, cycle 1

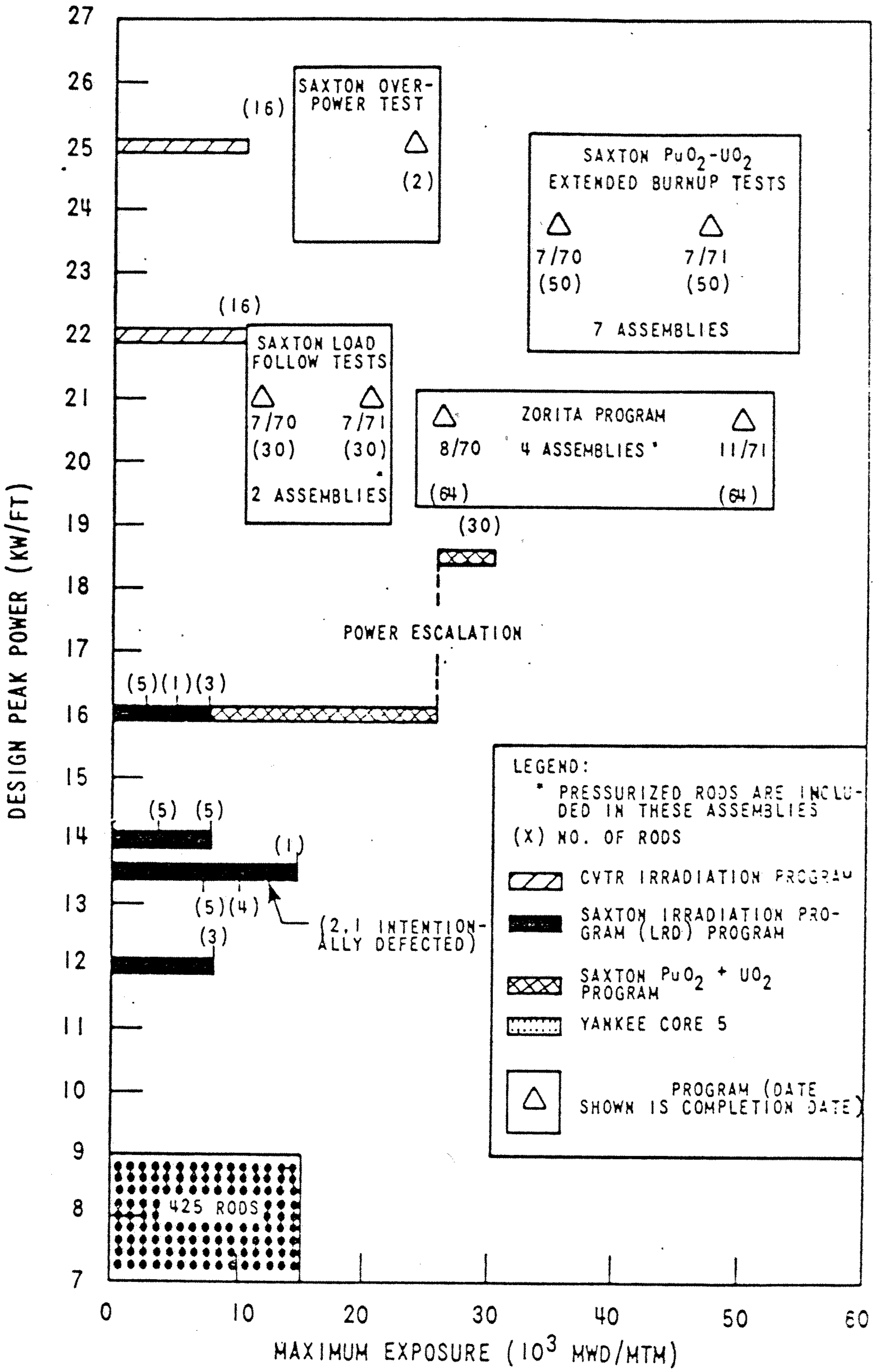
T_m = 547°F

Avg. Erichment = 2.8

INDIAN POINT 3	FSAR UPDATE
DOPPLER CONTRIBUTIONS TO THE POWER COEFFICIENT VS. POWER LEVEL	
REV. 0 JULY, 1982	FIGURE NO. 3.2-12



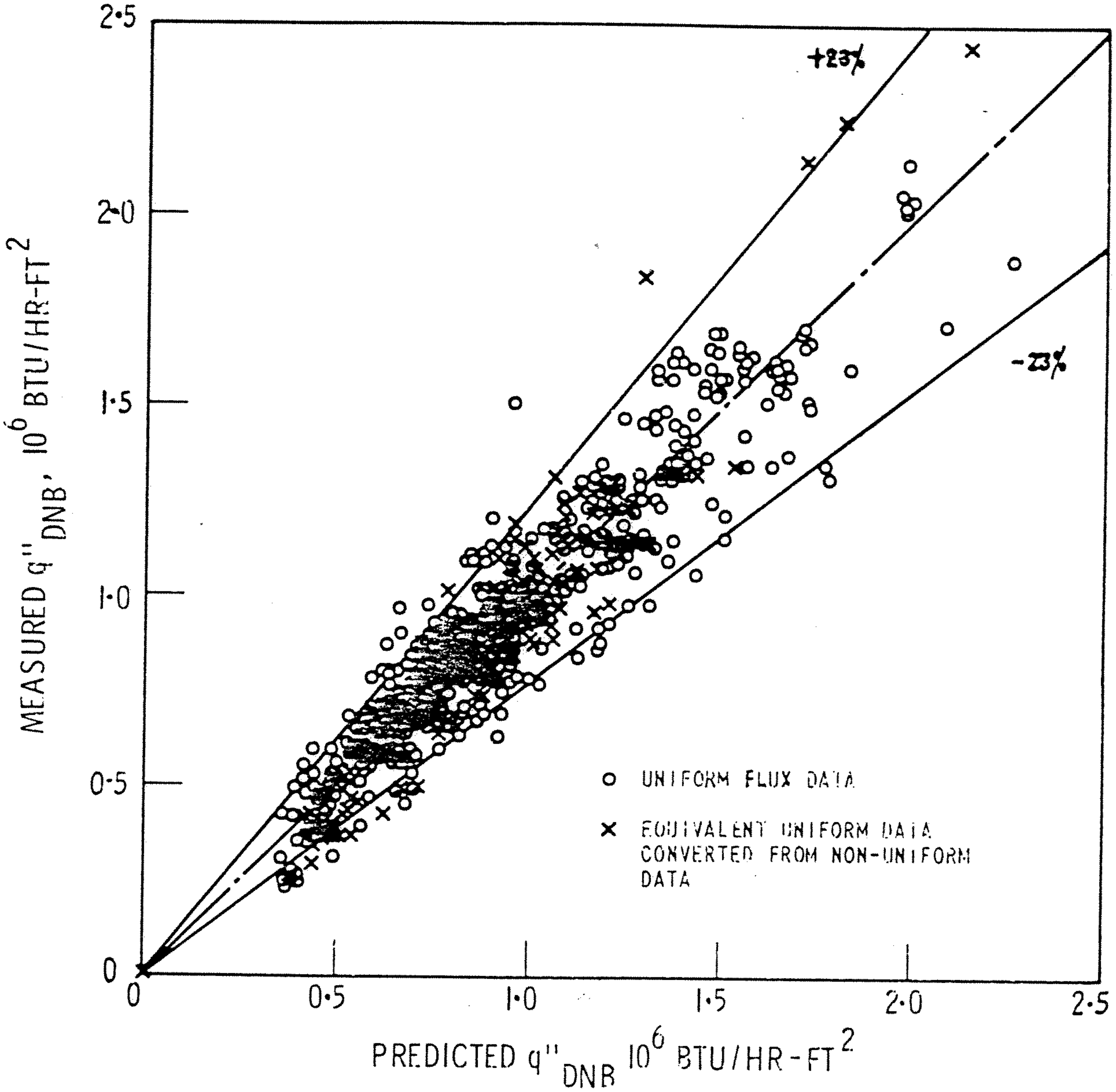
INDIAN POINT 3		FSAR UPDATE
THERMAL CONDUCTIVITY OF URANIUM DIOXIDE		
REV. 0	JULY, 1982	FIGURE NO. 3.2-13



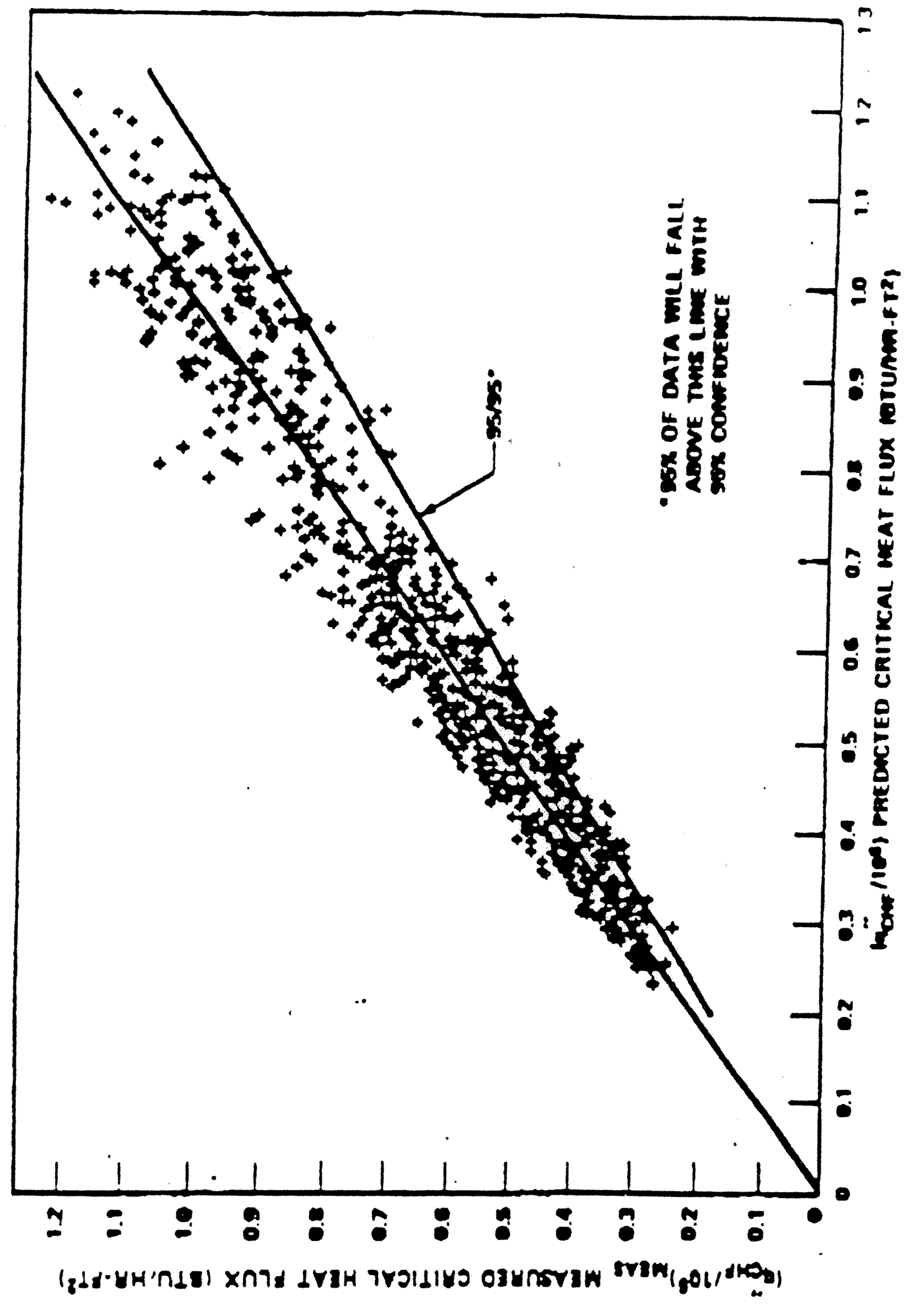
INDIAN POINT 3 | **FSAR UPDATE**

HIGH POWER FUEL ROD
EXPERIMENTAL PROGRAM

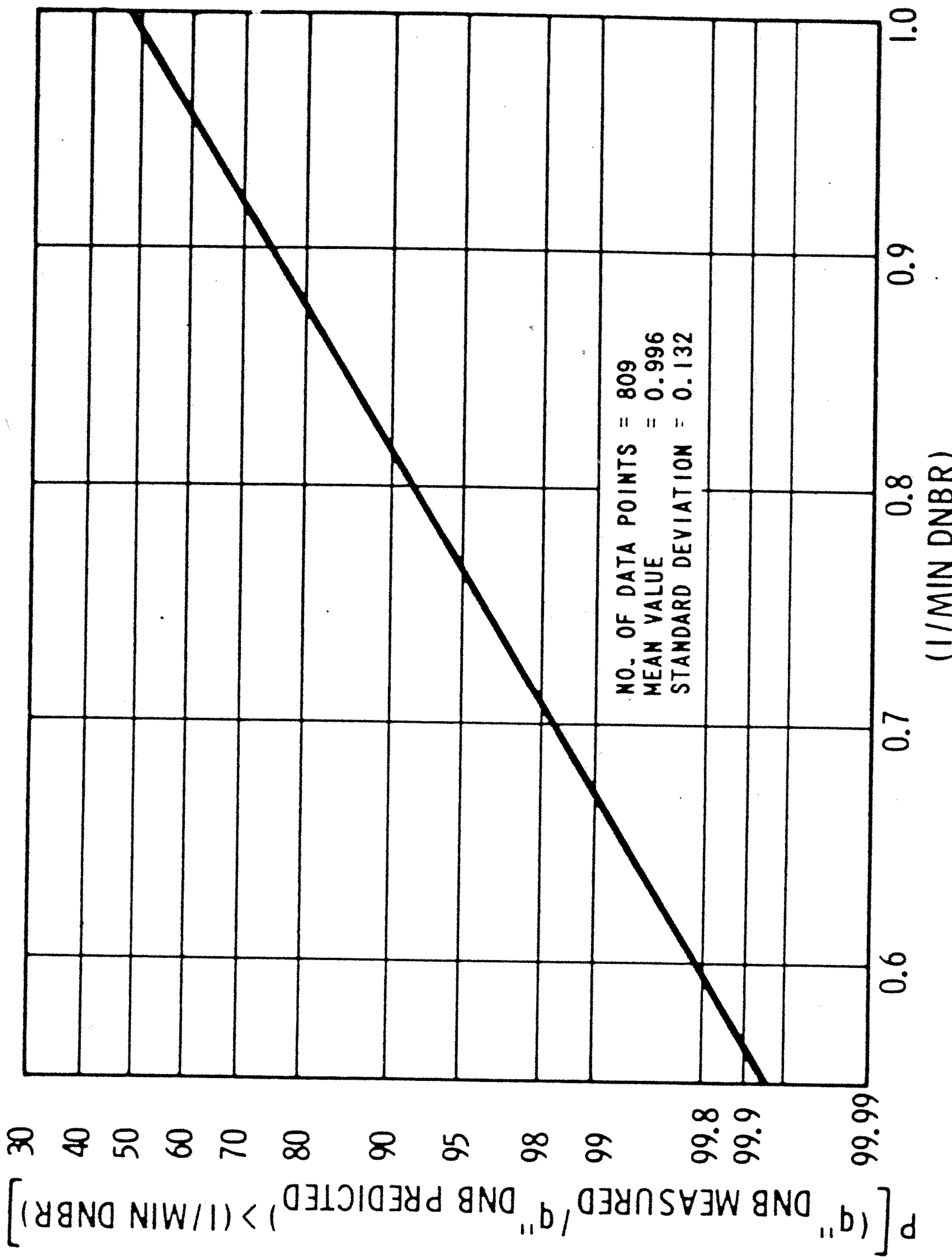
REV. 0 | JULY, 1982 | FIGURE NO. 3.2-14



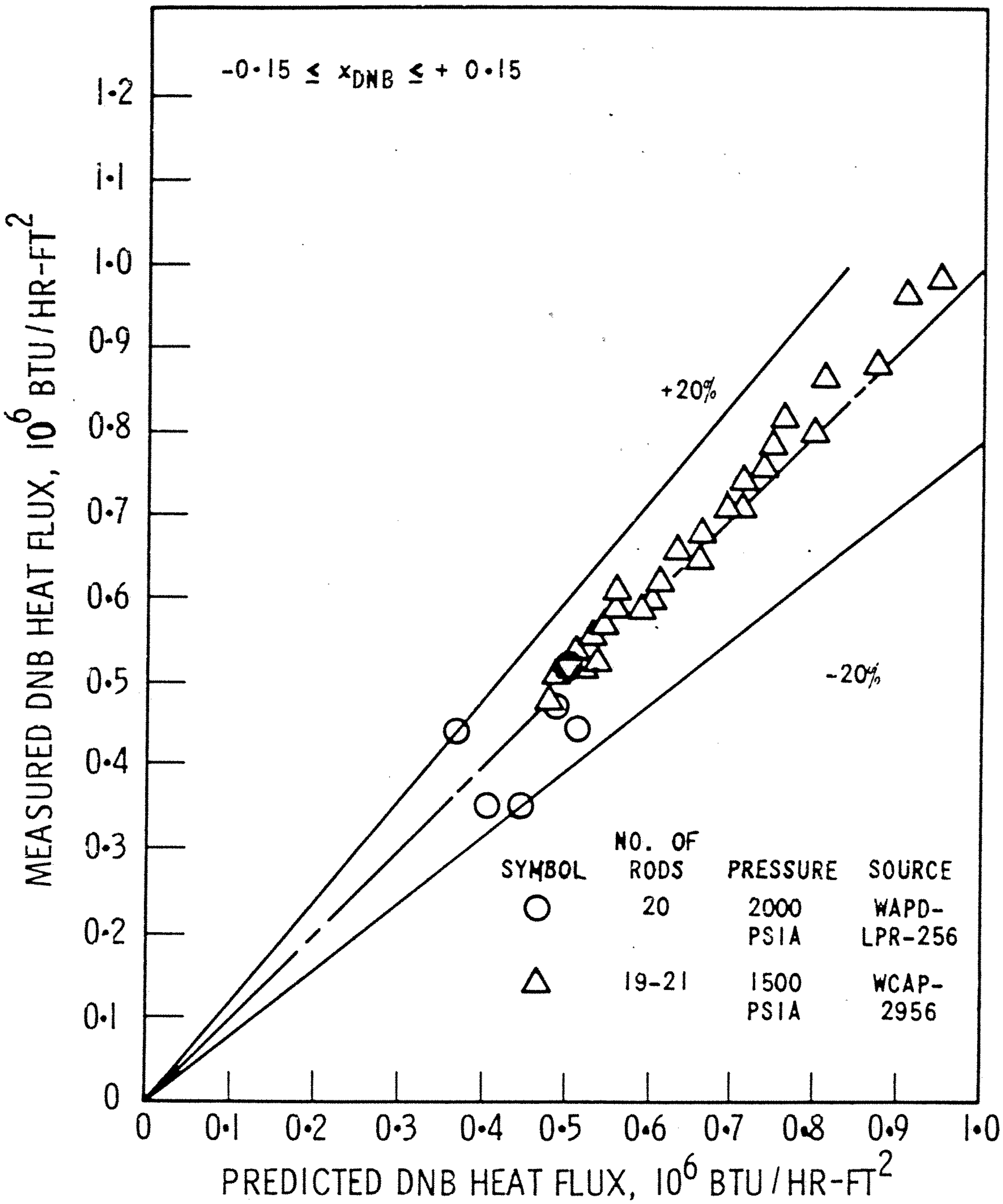
INDIAN POINT 3		FSAR UPDATE
COMPARISON OF W-3 PREDICTION AND UNIFORM FLUX DATA		
REV. 0	JULY, 1982	FIGURE NO. 3.2-15



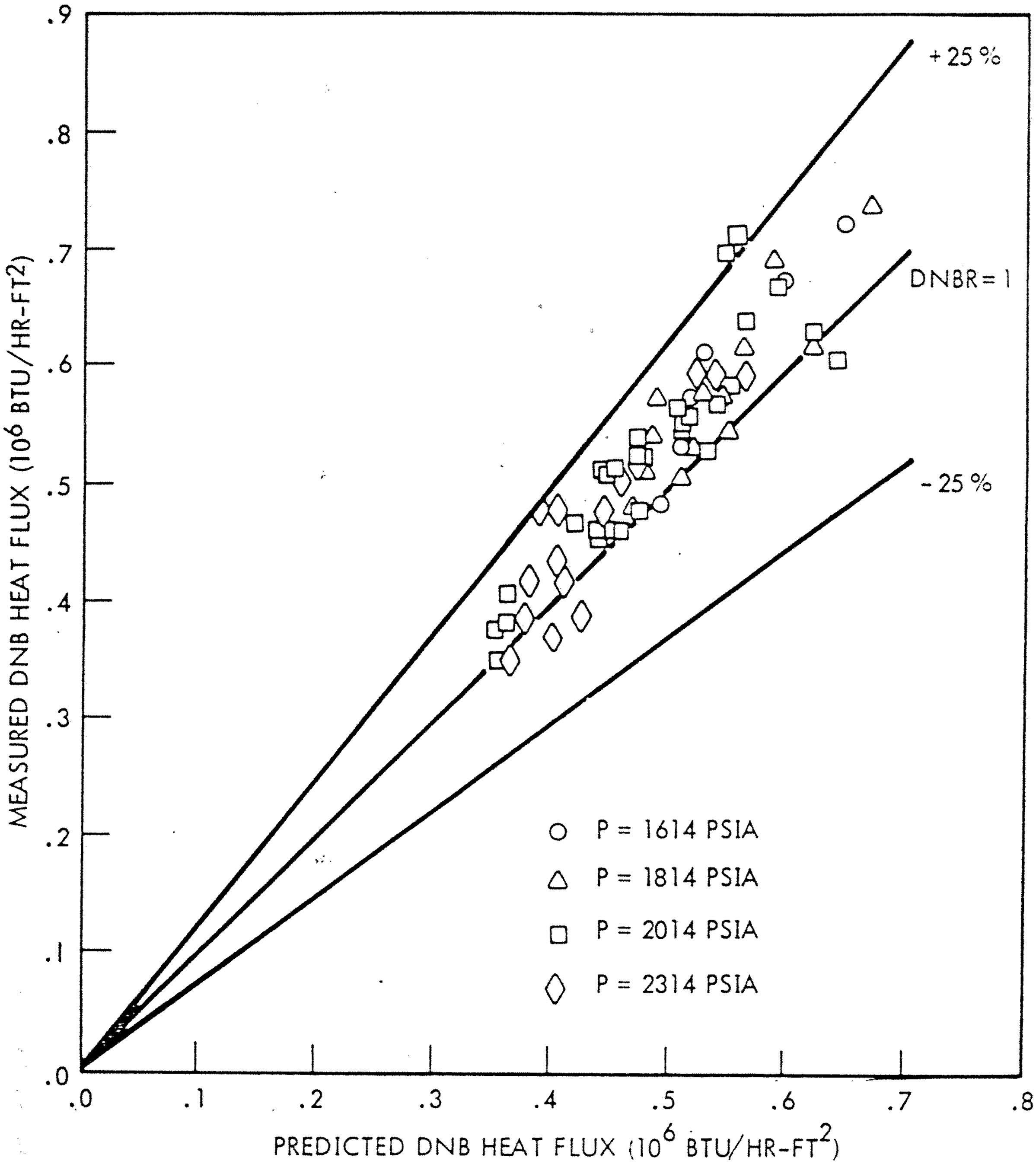
INDIAN POINT 3	FSAR UPDATE
MEASURED VERSUS PREDICTED CRITICAL HEAT FLUX WRB-1 CORRELATION	
REV. 0, JULY 1990	FIGURE NO. 3.2-15A



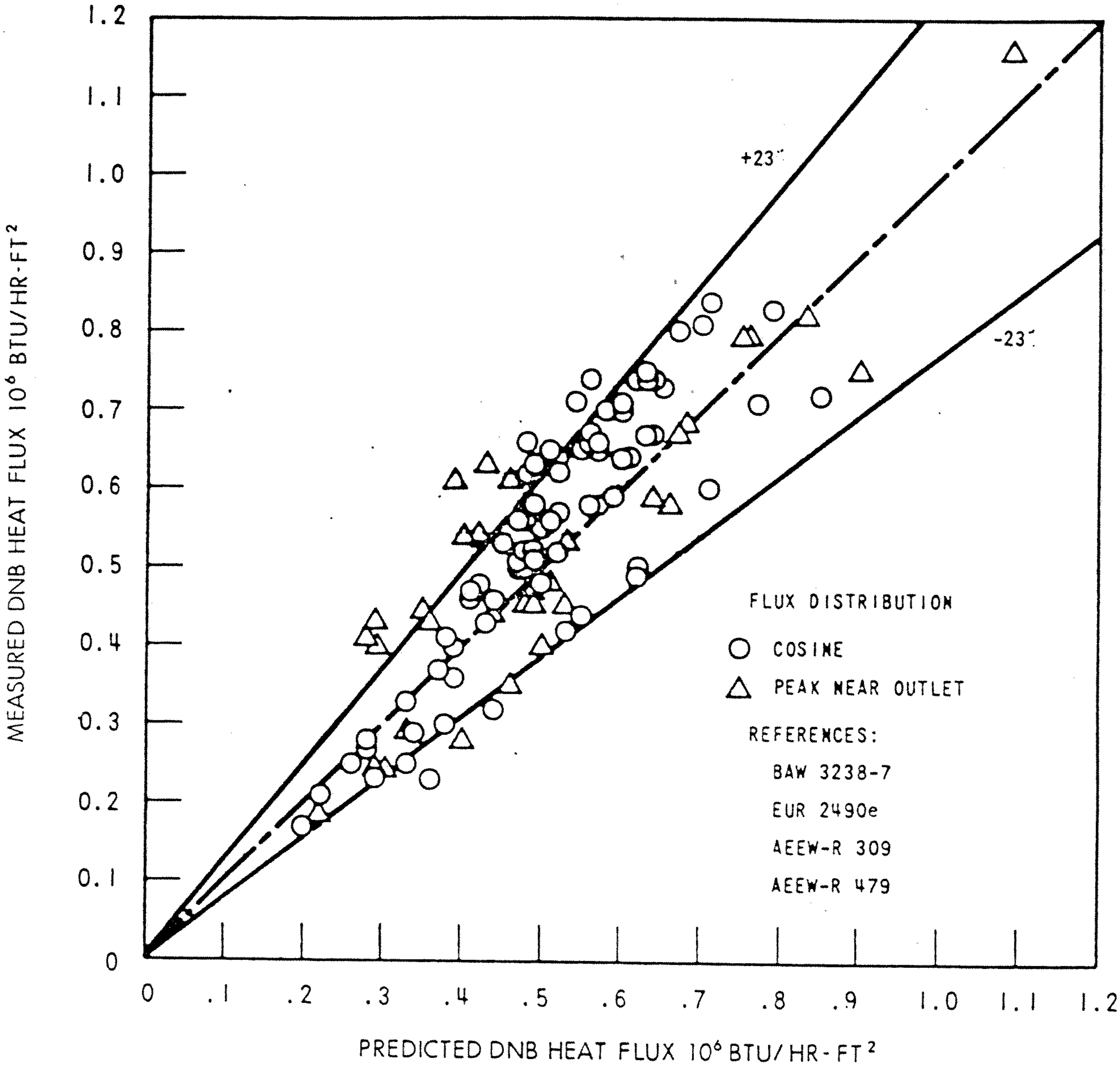
INDIAN POINT 3		FSAR UPDATE
W-3 CORRELATION PROBABILITY DISTRIBUTION CURVE		
REV. 0	JULY, 1982	FIGURE NO. 3.2-16



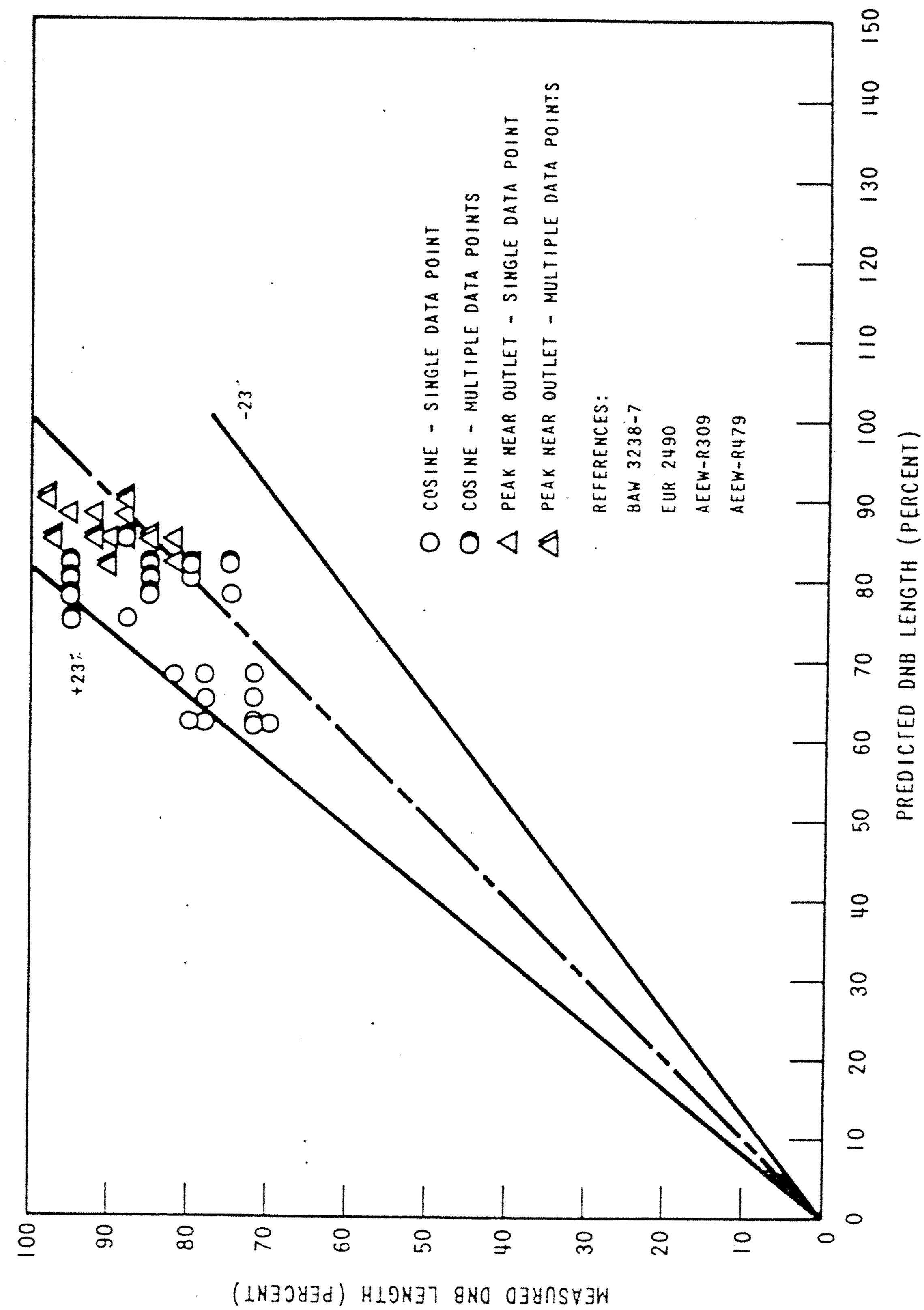
INDIAN POINT 3		FSAR UPDATE
COMPARISON OF W-3 CORRELATION WITH ROD BUNDLE DNB DATA (SIMPLE GRID WITHOUT MIXING VANE)		
REV. 0	JULY, 1982	FIGURE NO. 3.2-17



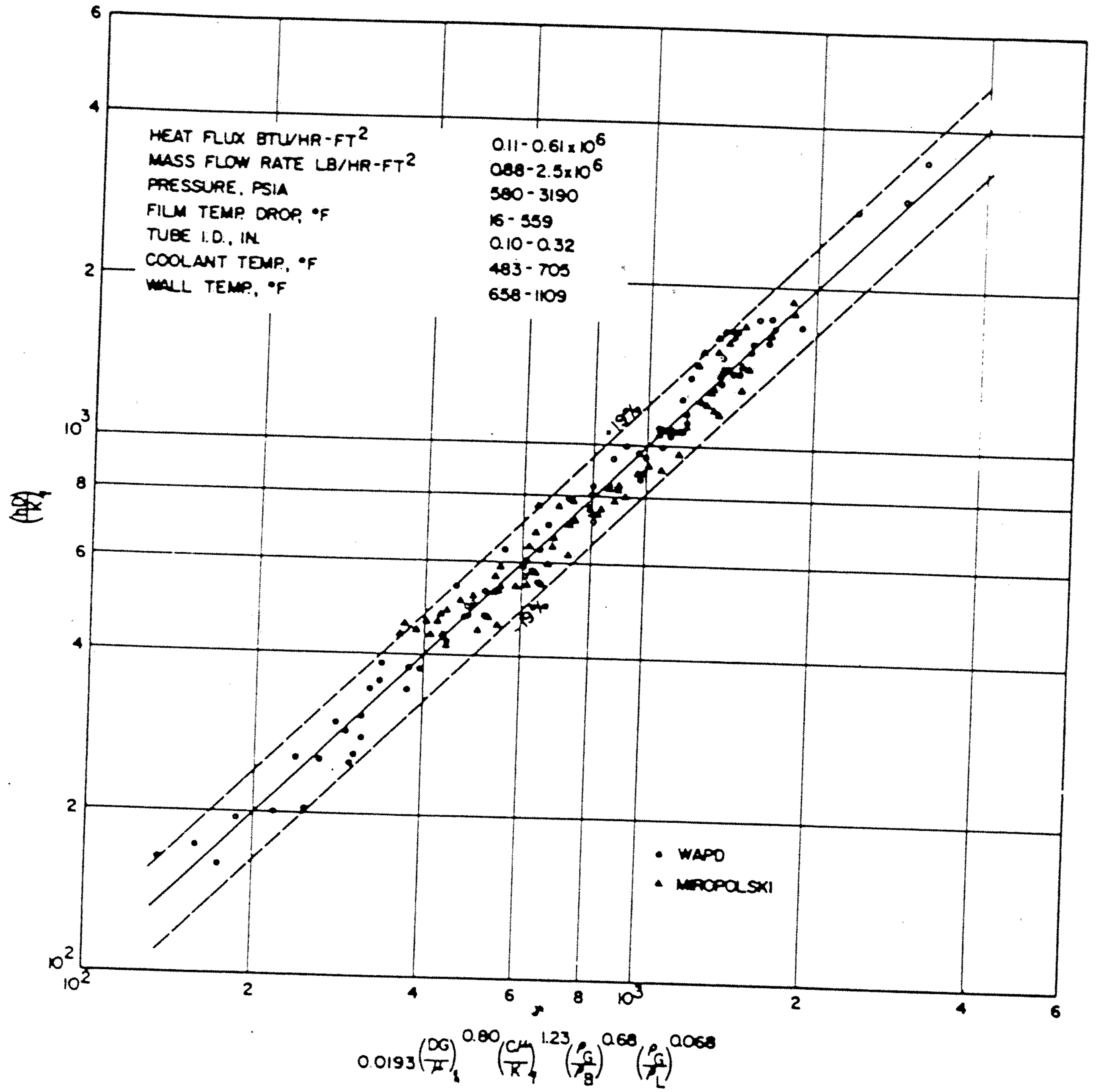
INDIAN POINT 3		FSAR UPDATE
COMPARISON OF W-3 CORRELATION WITH ROD BUNDLE DNB DATA (SIMPLE GRID WITH MIXING VANE)		
REV. 0	JULY, 1982	FIGURE NO. 3.2-18



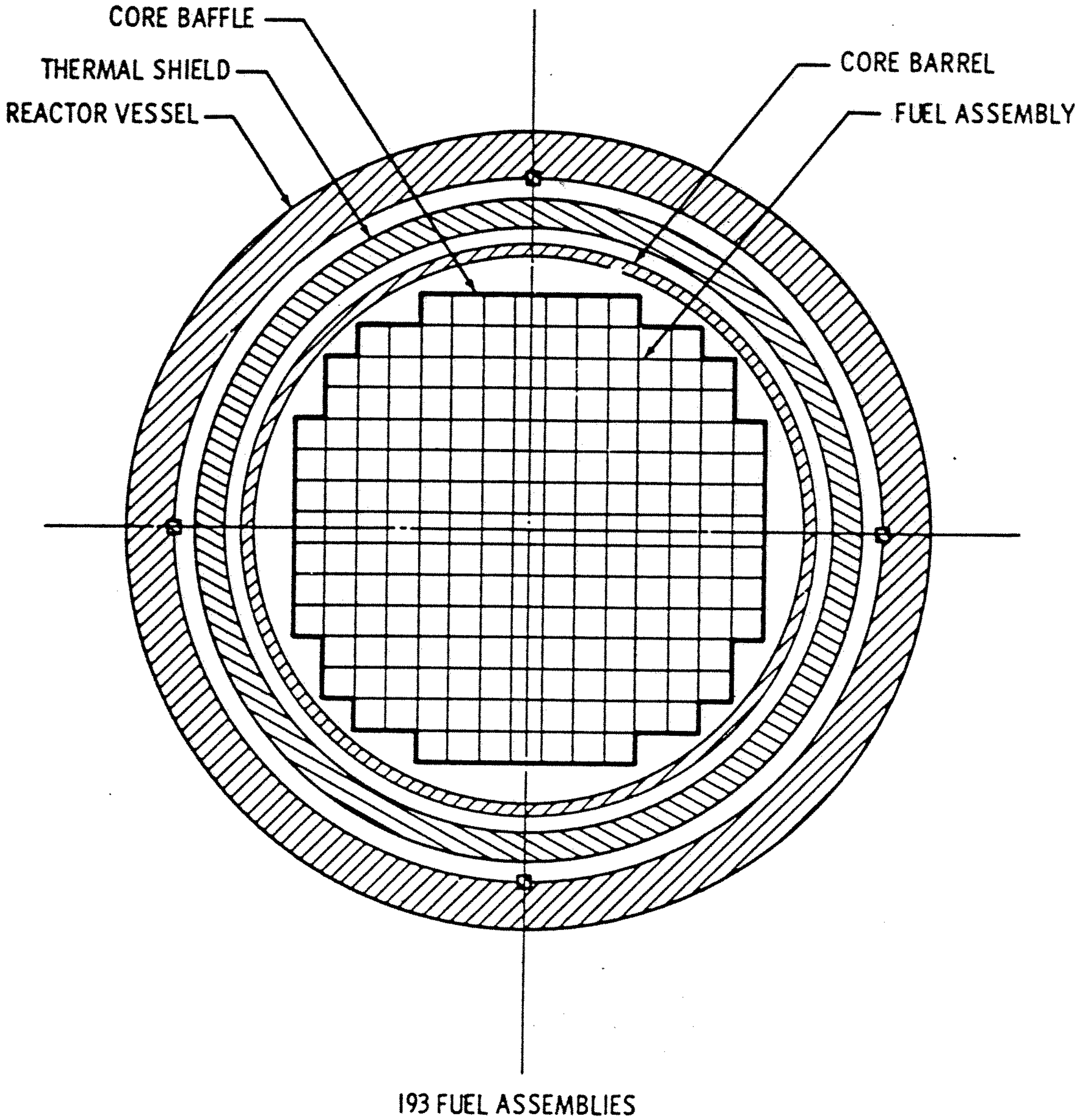
INDIAN POINT 3	FSAR UPDATE
COMPARISON OF NON-UNIFORM DNB DATA WITH W-3 PREDICTIONS	
REV. 0 JULY, 1982	FIGURE NO. 3.2-19



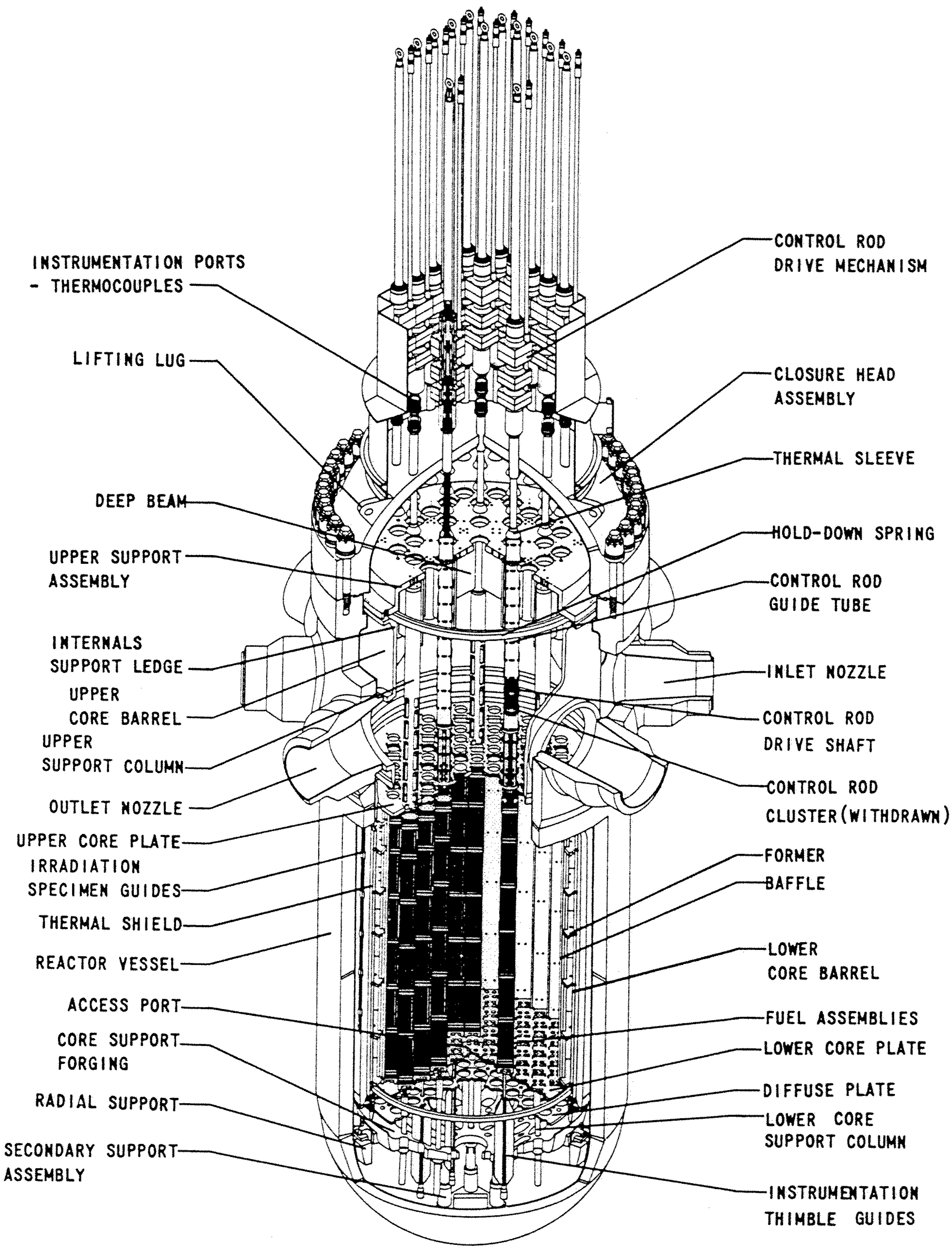
INDIAN POINT 3		FSAR UPDATE
COMPARISON OF W-3 PREDICTION WITH MEASURED DNB LOCATION		
REV. 0	JULY, 1982	FIGURE NO. 3.2-20



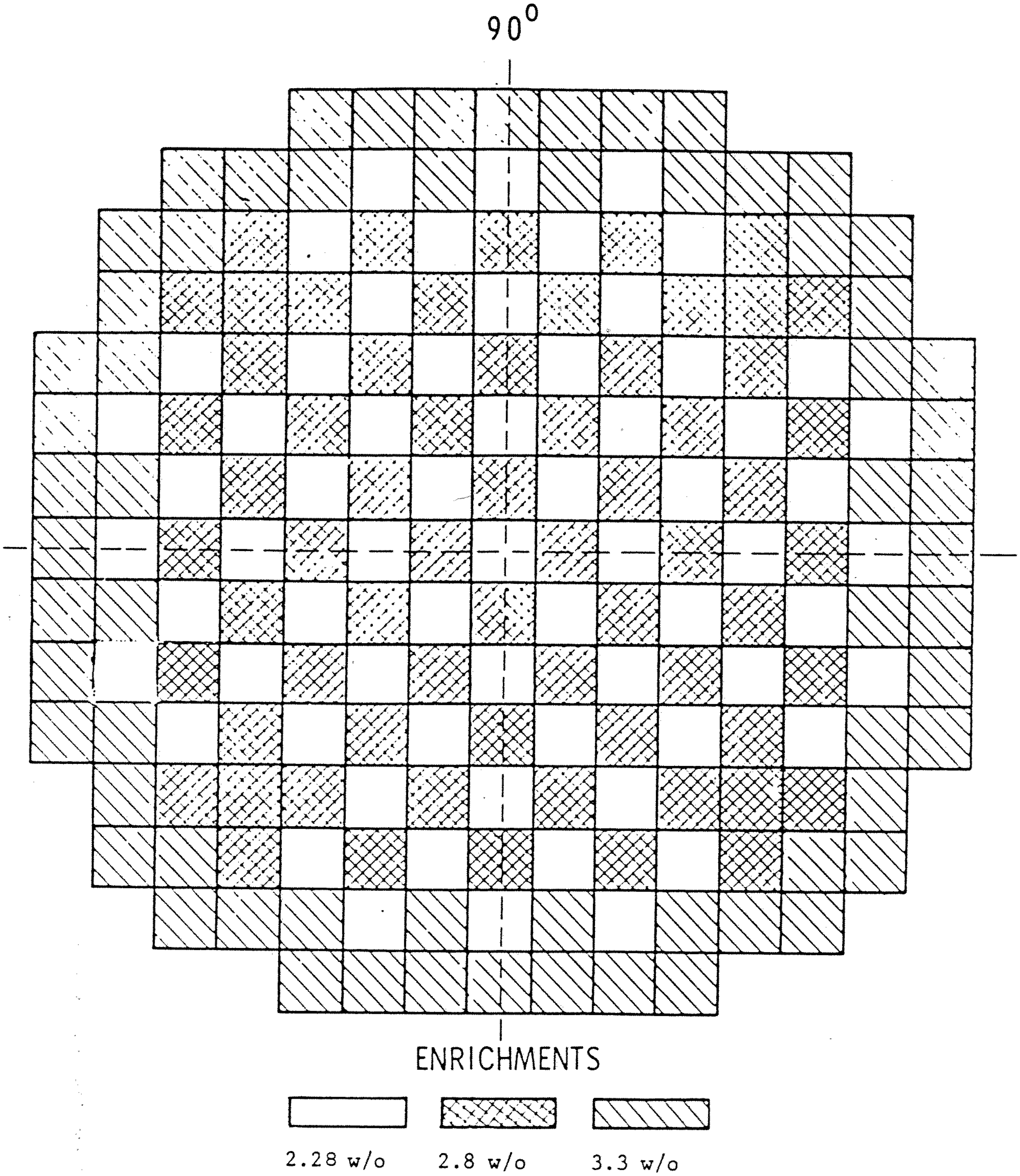
INDIAN POINT 3	FSAR UPDATE
STABLE FILM BOILING HEAT TRANSFER DATA AND CORRELATION	
REV. 0 JULY, 1982	FIGURE NO. 3.2-21



INDIAN POINT 3		FSAR UPDATE
CORE CROSS SECTION		
REV. 0	JULY, 1982	FIGURE NO. 3.2-22



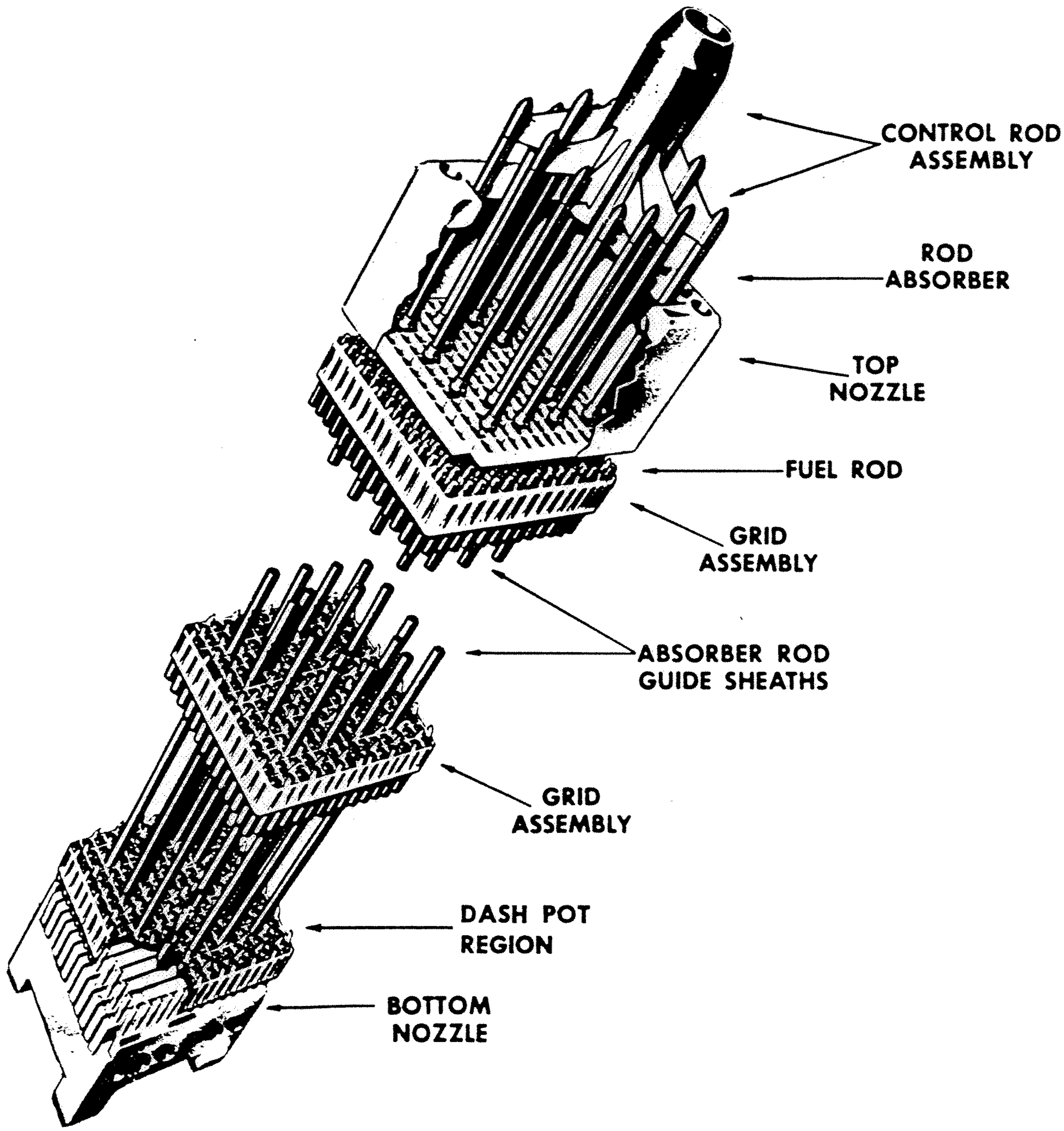
INDIAN POINT 3	FSAR UPDATE
REACTOR VESSEL AND INTERNALS	
REV. 0 JULY, 1982	FIGURE NO. 3.2-23



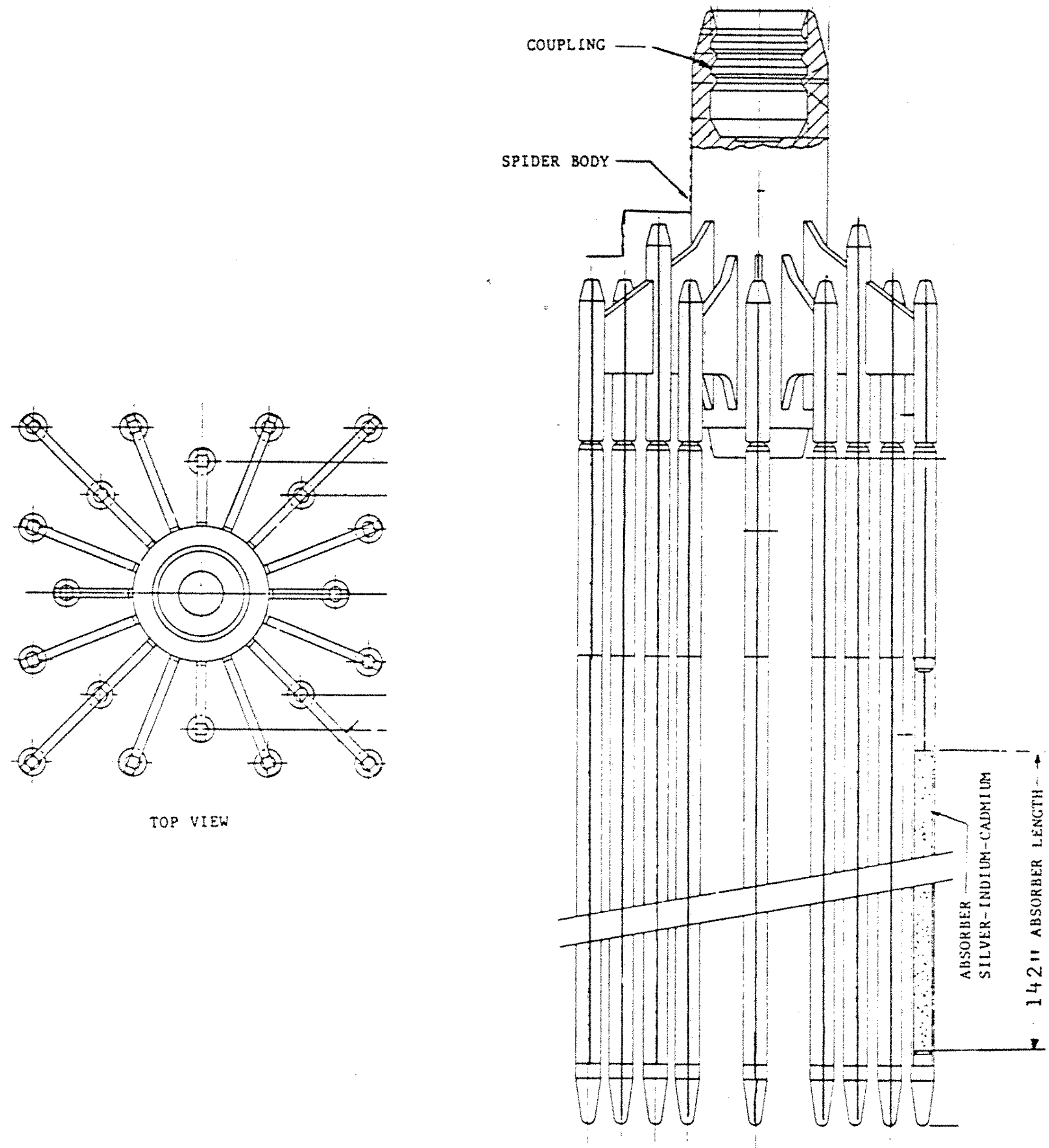
INDIAN POINT 3 FSAR UPDATE

CORE LOADING ARRANGEMENT (FIRST CYCLE)

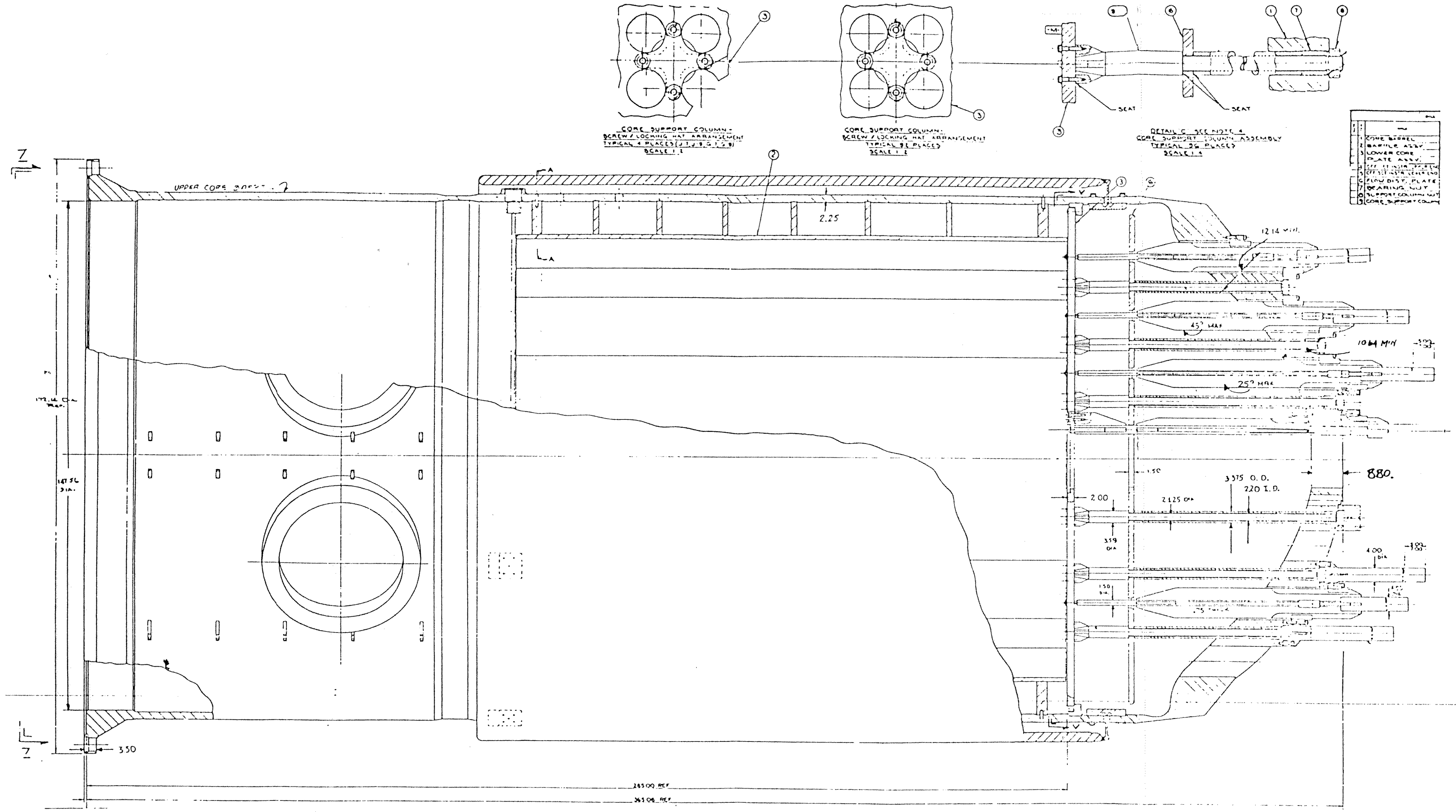
REV. 0 JULY, 1982 FIGURE NO. 3.2-24



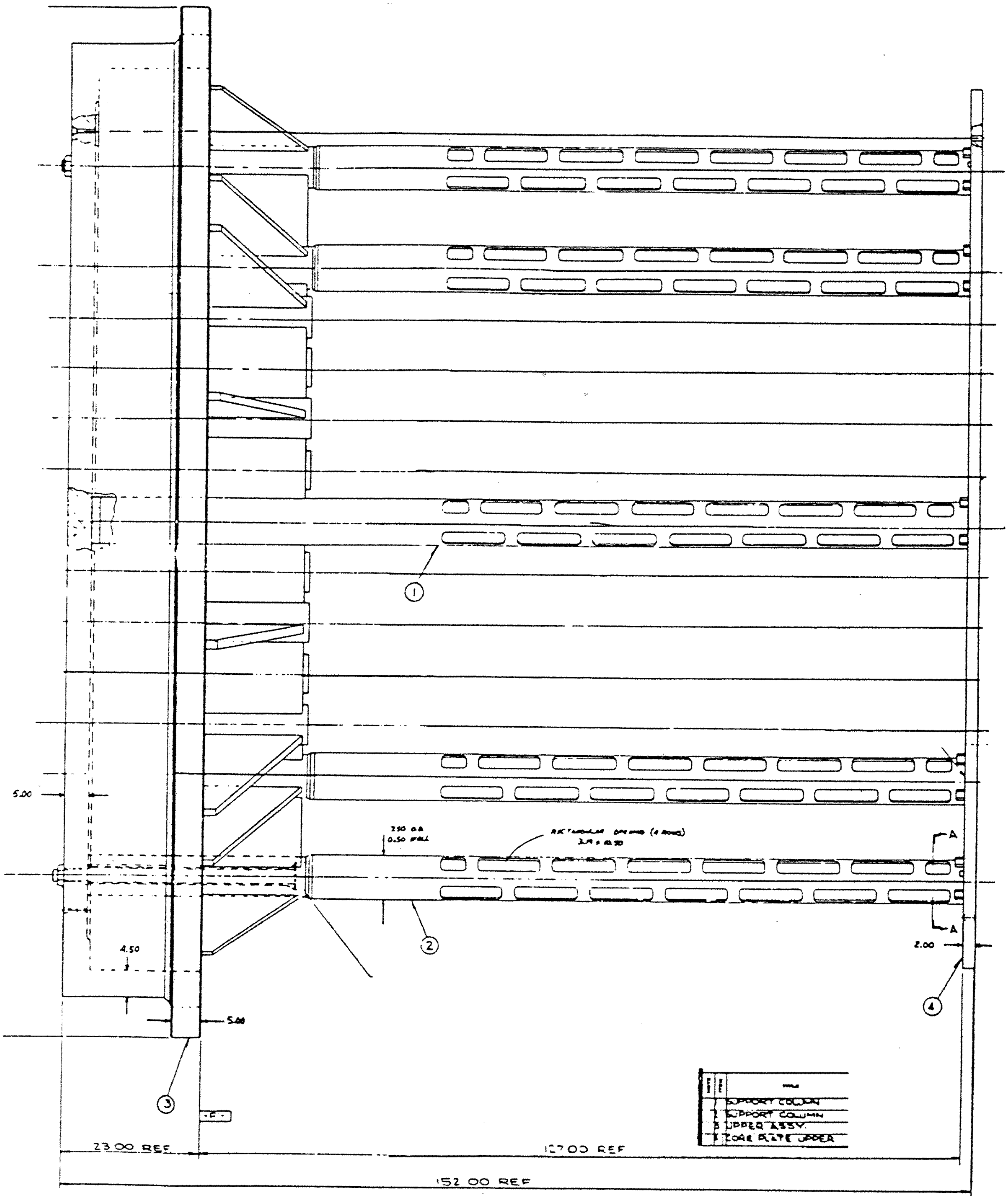
INDIAN POINT 3		FSAR UPDATE
TYPICAL ROD CLUSTER CONTROL ASSEMBLY		
REV. 0	JULY, 1982	FIGURE NO. 3.2-25



INDIAN POINT 3		FSAR UPDATE
ROD CONTROL CLUSTER ASSEMBLY OUTLINE		
REV. 0	JULY, 1982	FIGURE NO. 3.2-26



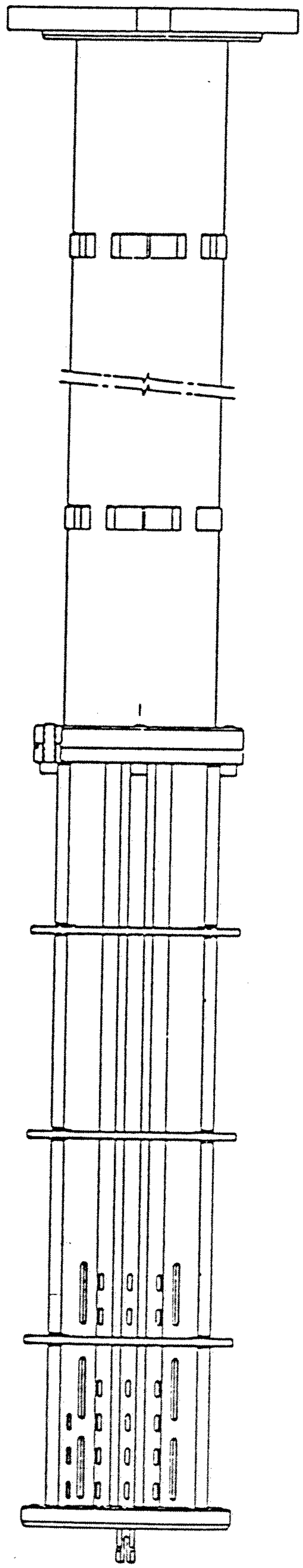
INDIAN POINT 3		FSAR UPDATE
CORE BARREL ASSEMBLY		
REV. 0	JULY, 1982	FIGURE NO. 3.2-27

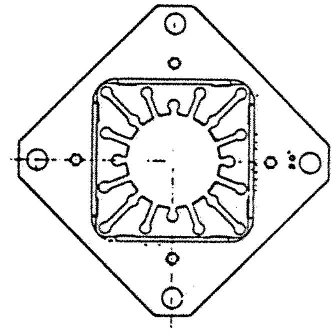


INDIAN POINT 3 FSAR UPDATE

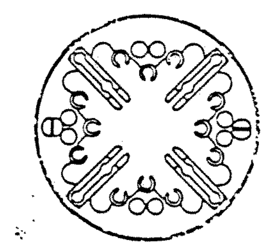
UPPER CORE SUPPORT STRUCTURE

REV. 0 JULY, 1982 FIGURE NO. 3.2-28



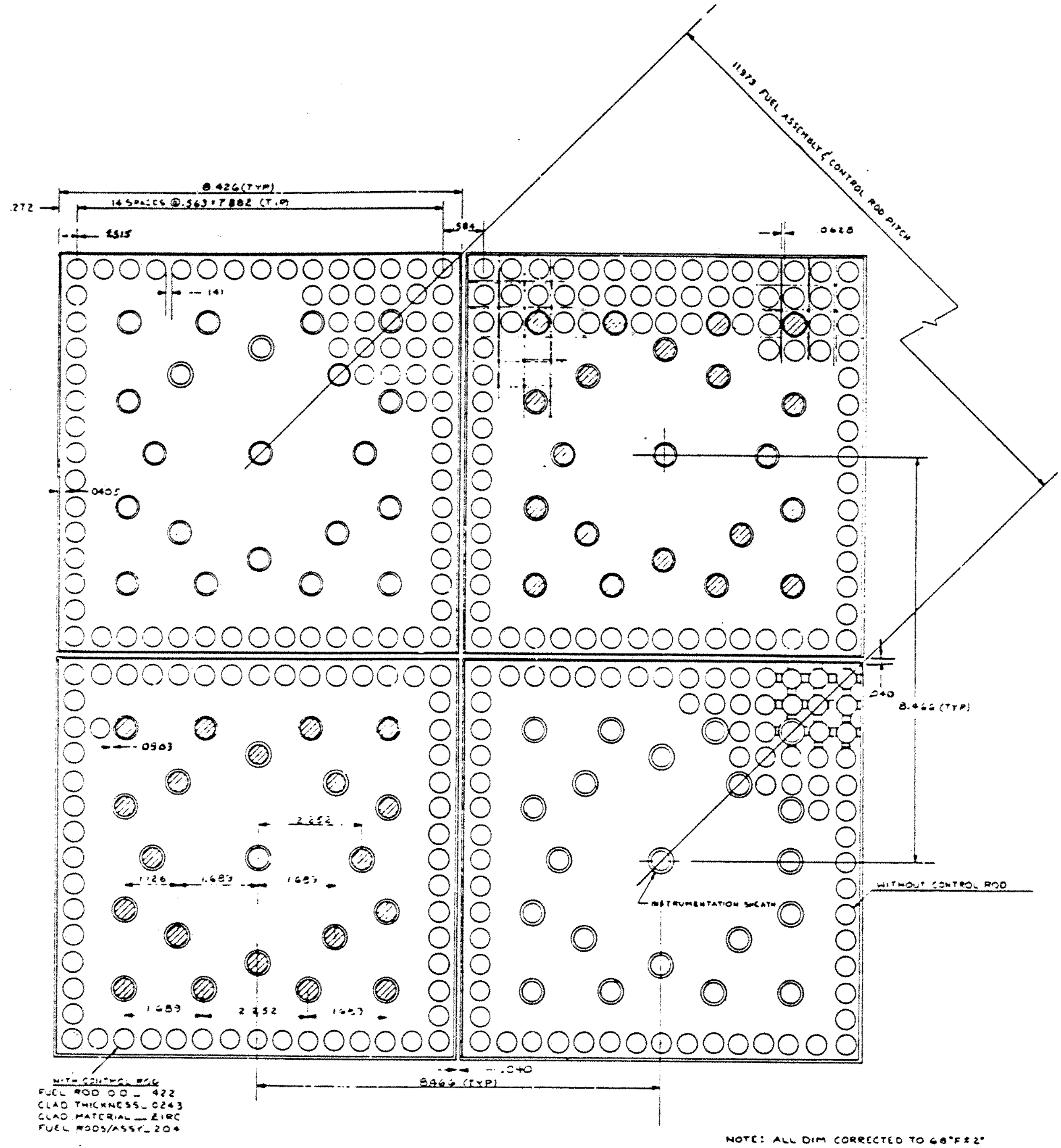


TOP VIEW

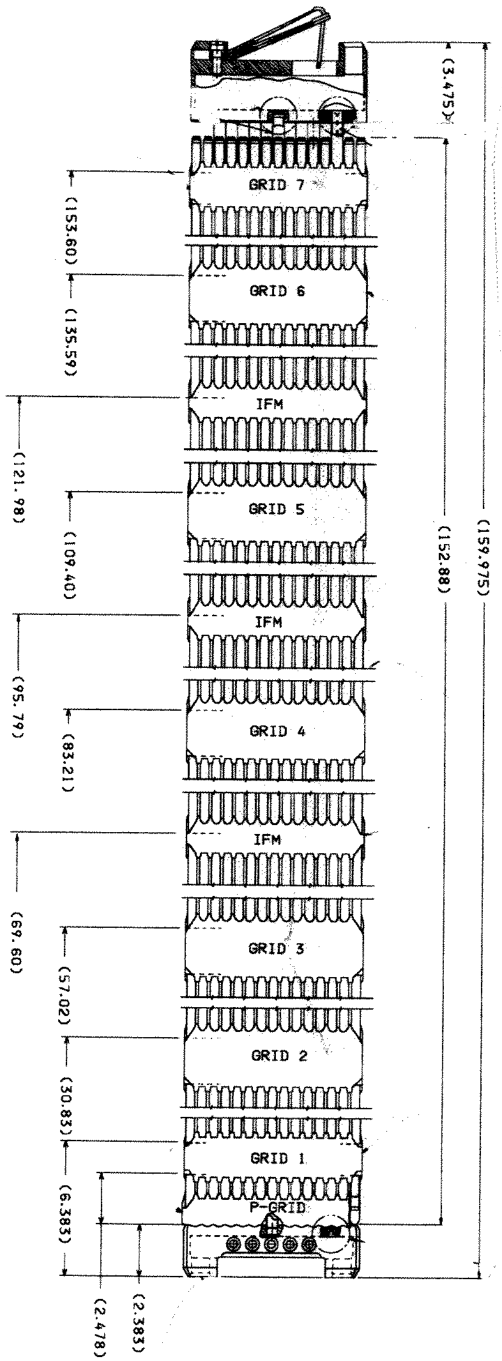


BOTTOM VIEW

INDIAN POINT 3		FSAR UPDATE
GUIDE TUBE ASSEMBLY		
REV. 0	JULY, 1982	FIGURE NO. 3.2-29



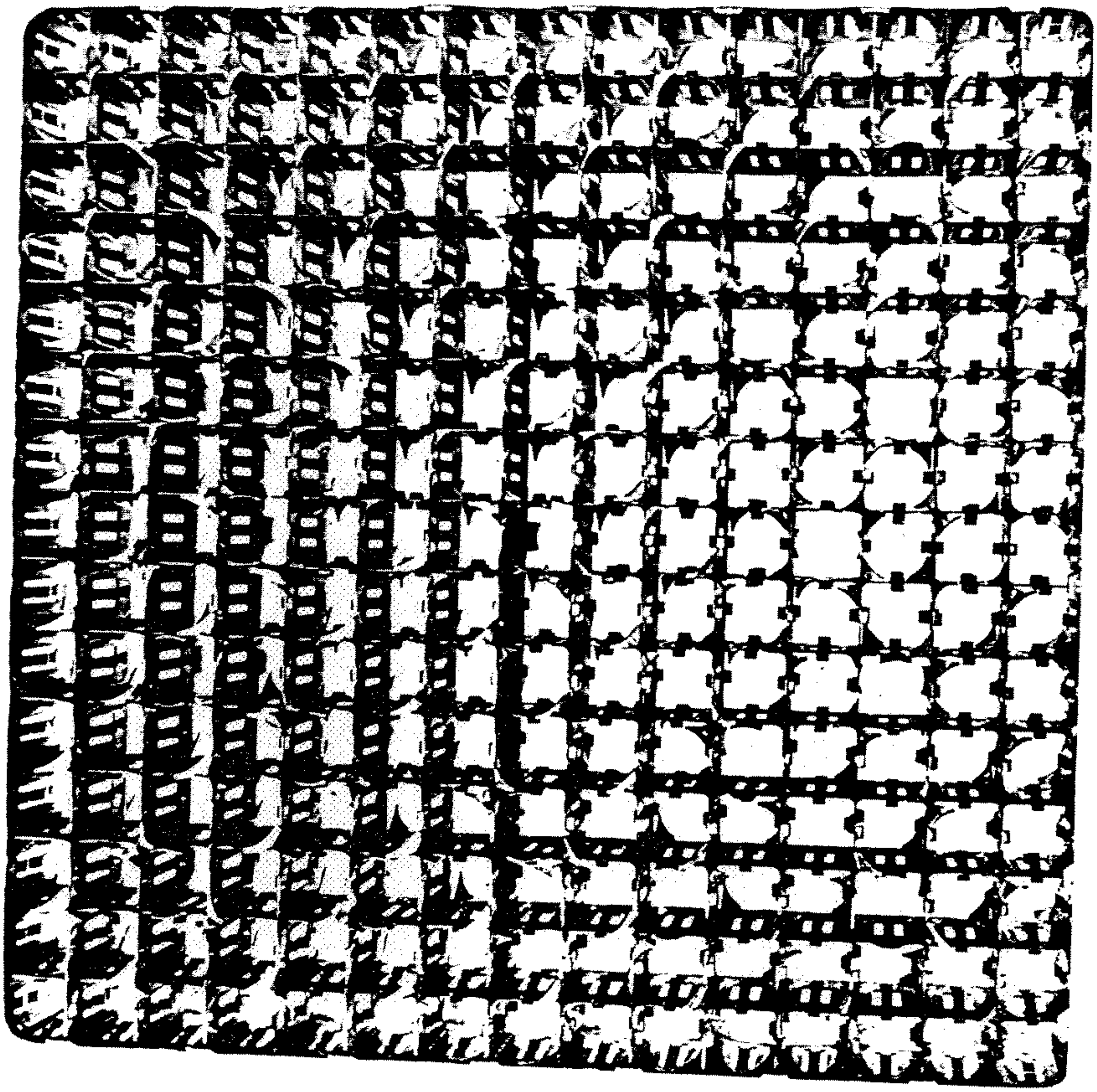
INDIAN POINT 3		FSAR UPDATE
FUEL ASSEMBLY AND CONTROL CLUSTER CROSS SECTION		
REV. 0	JULY, 1982	FIGURE NO. 3.2-30



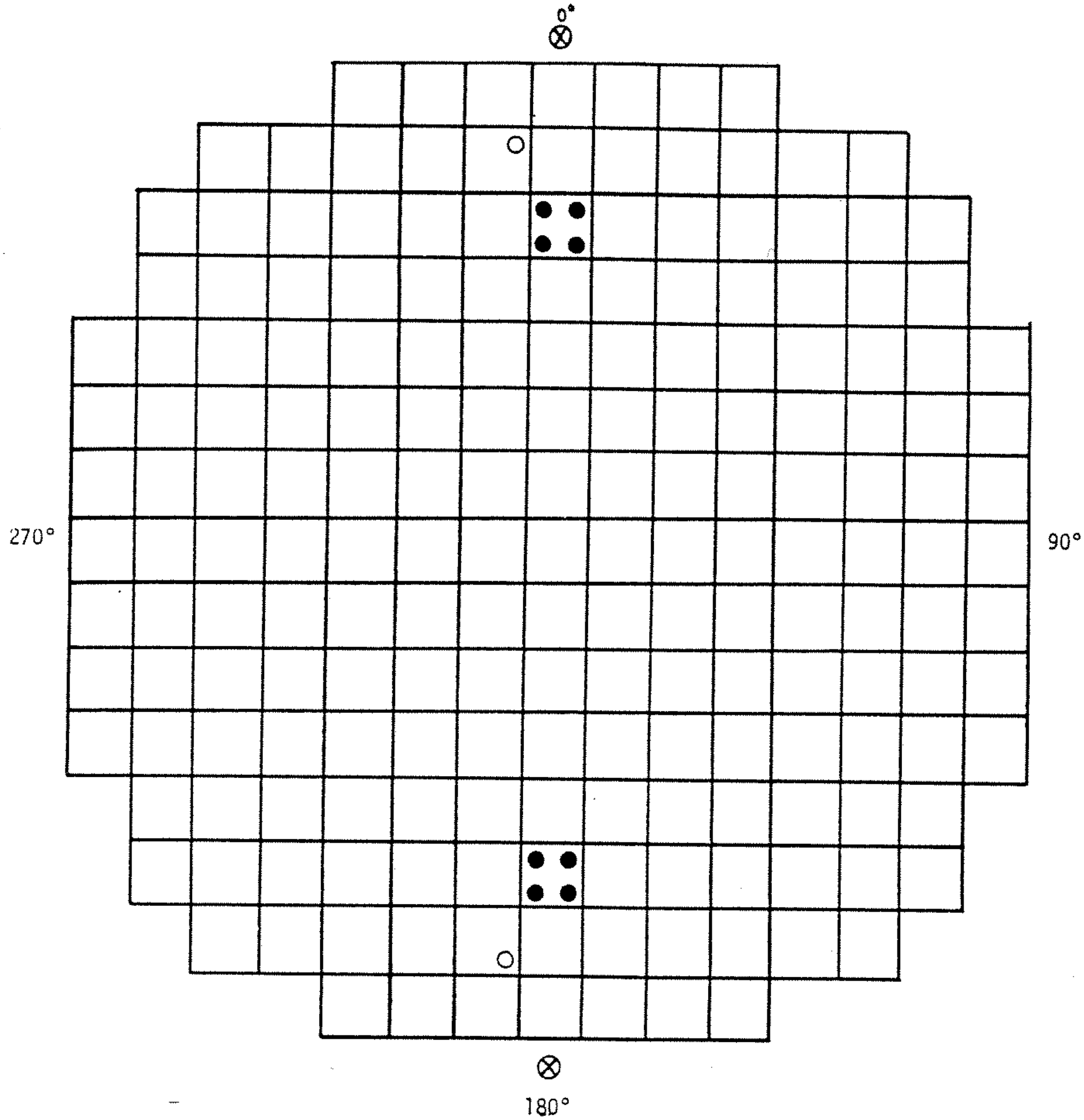
INDIAN POINT 3	FSAR UPDATE

FUEL ASSEMBLY OUTLINE
(Ref: Westinghouse Dwg 10006E64 r1)

FIGURE NO. 3.2-31

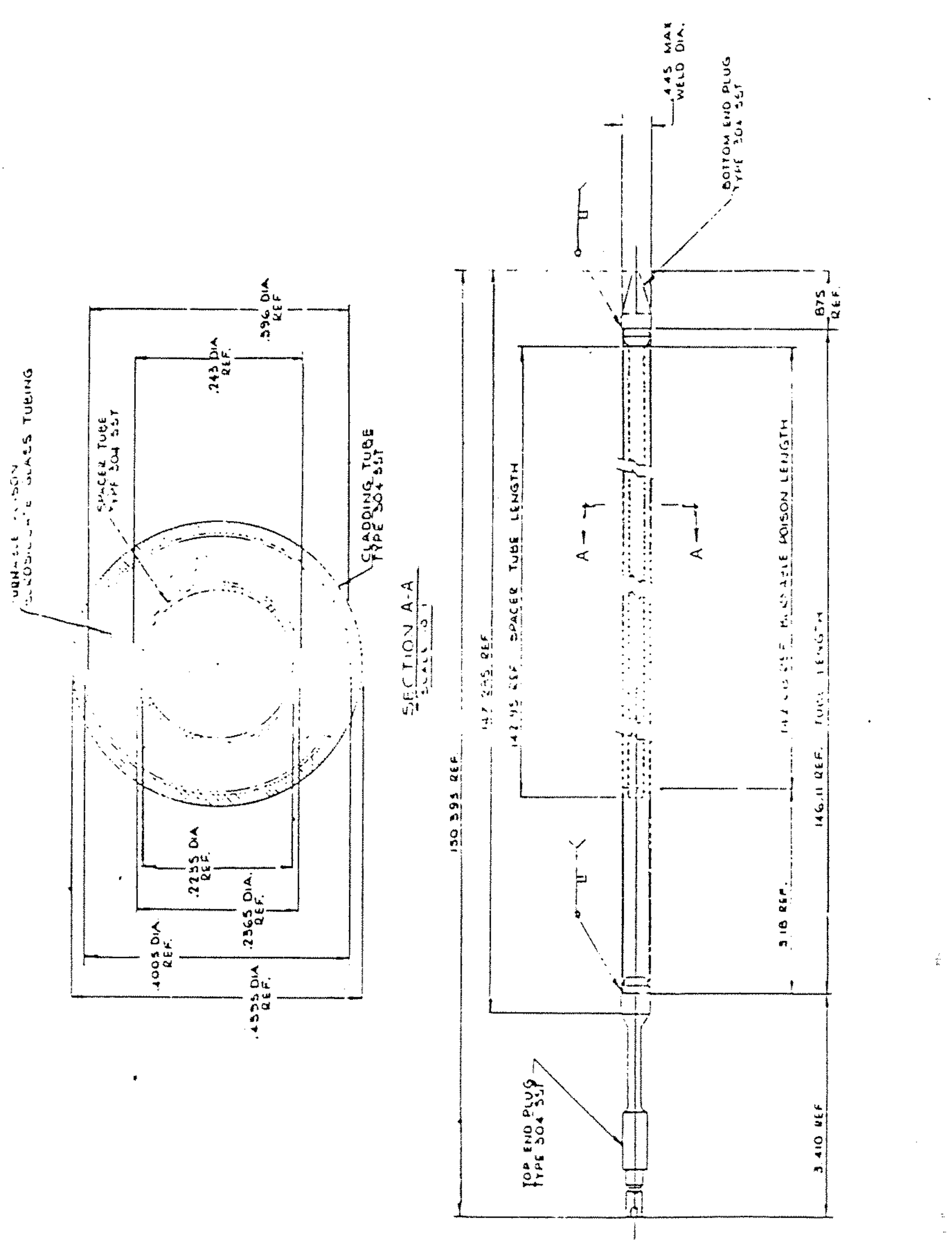


INDIAN POINT 3		FSAR UPDATE
SPRING CLIP GRID ASSEMBLY		
REV. 0	JULY, 1982	FIGURE NO. 3.2-32



○ Primary source rod
● Secondary source rod
⊗ Detector Location

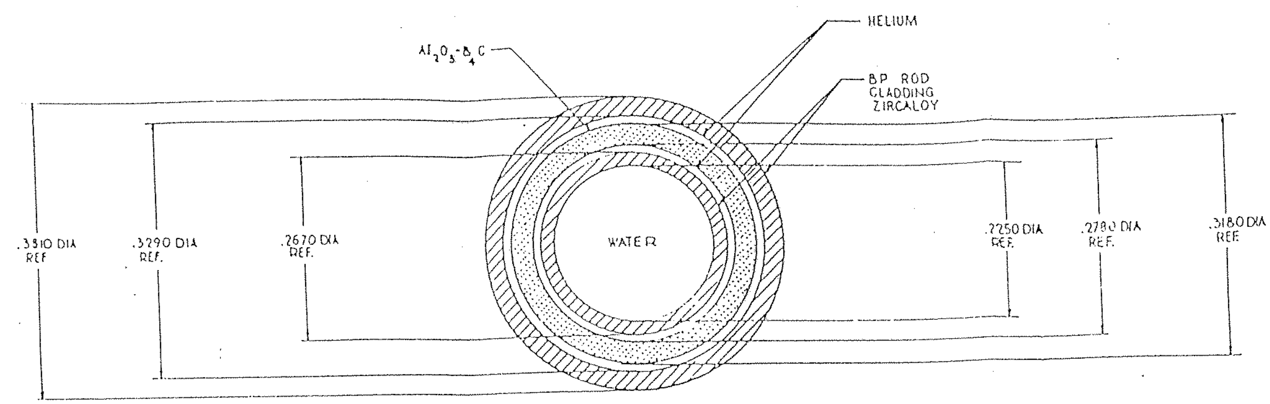
INDIAN POINT 3		FSAR UPDATE
NEUTRON SOURCE LOCATIONS (FIRST CYCLE)		
REV. 0	JULY, 1982	FIGURE NO. 3.2-33



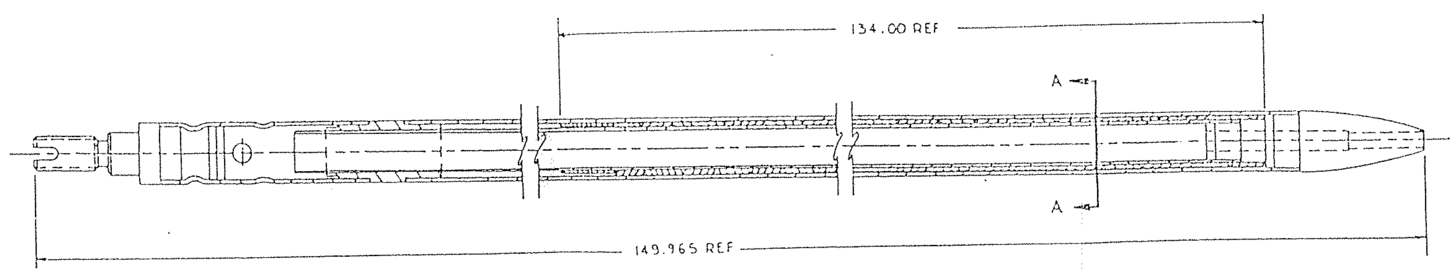
INDIAN POINT 3 FSAR UPDATE

OLD BURNABLE POISON ROD
(BOROSILICATE GLASS BURNABLE ABSORBER)

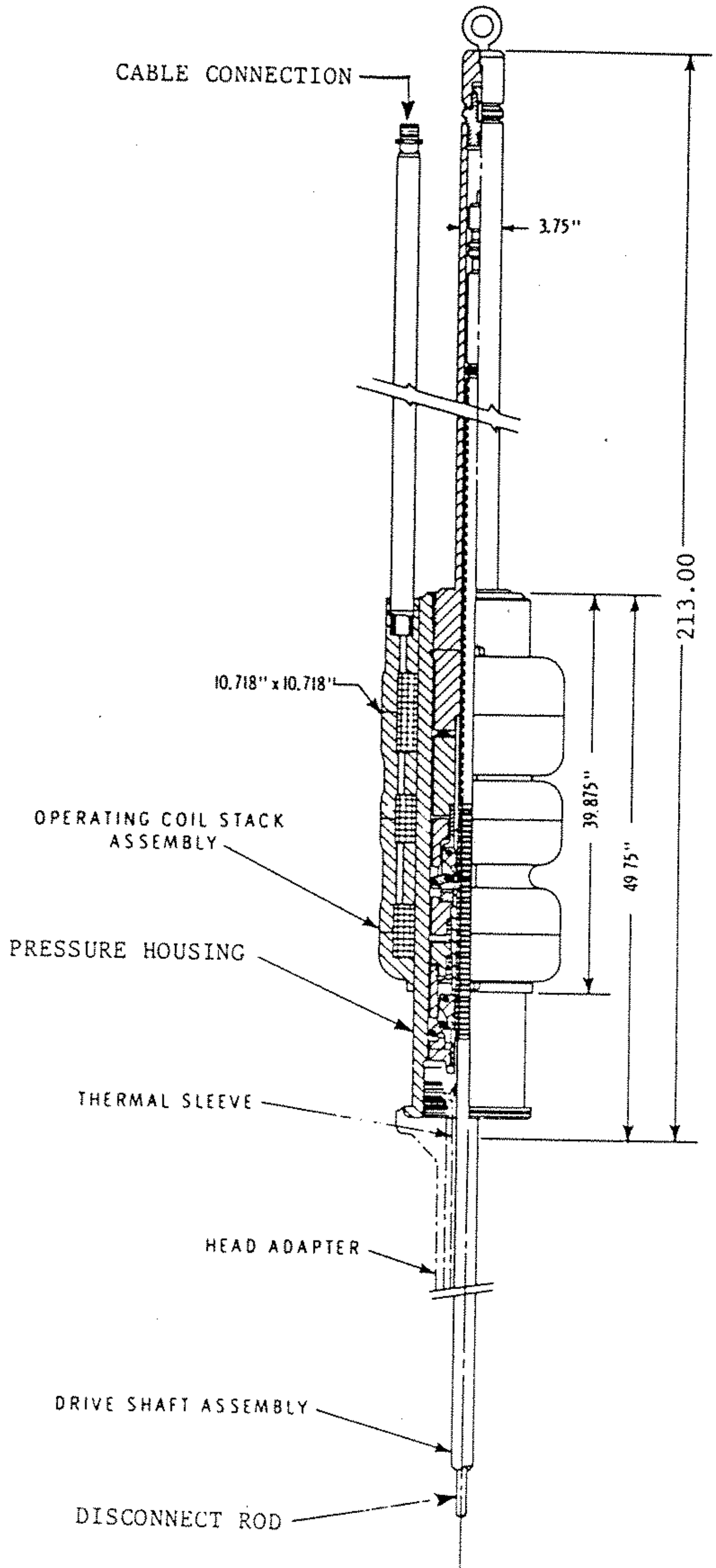
REV. 1	JULY, 1986	FIGURE NO. 3.2-34



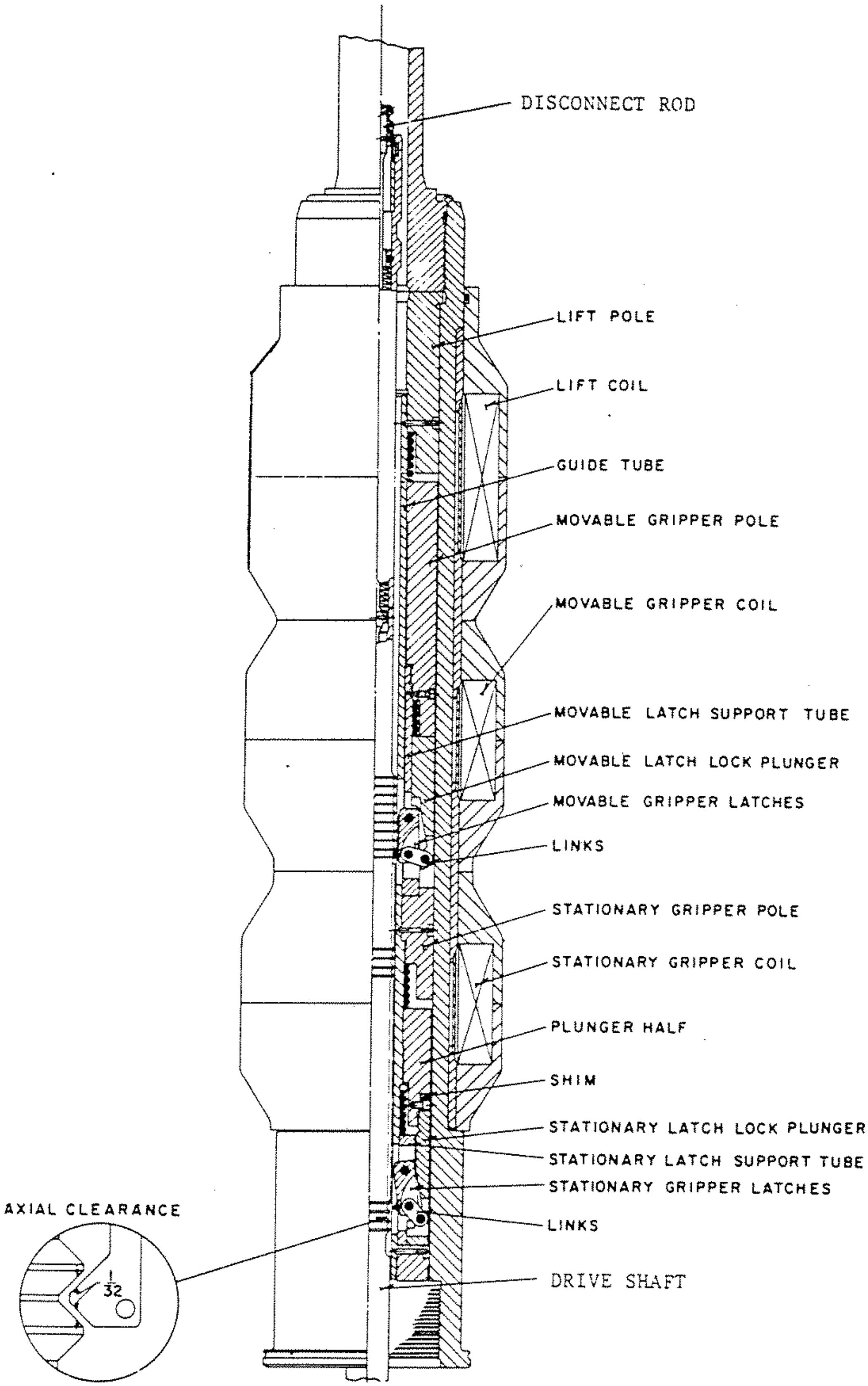
SECTION A-A
SCALE 2:1



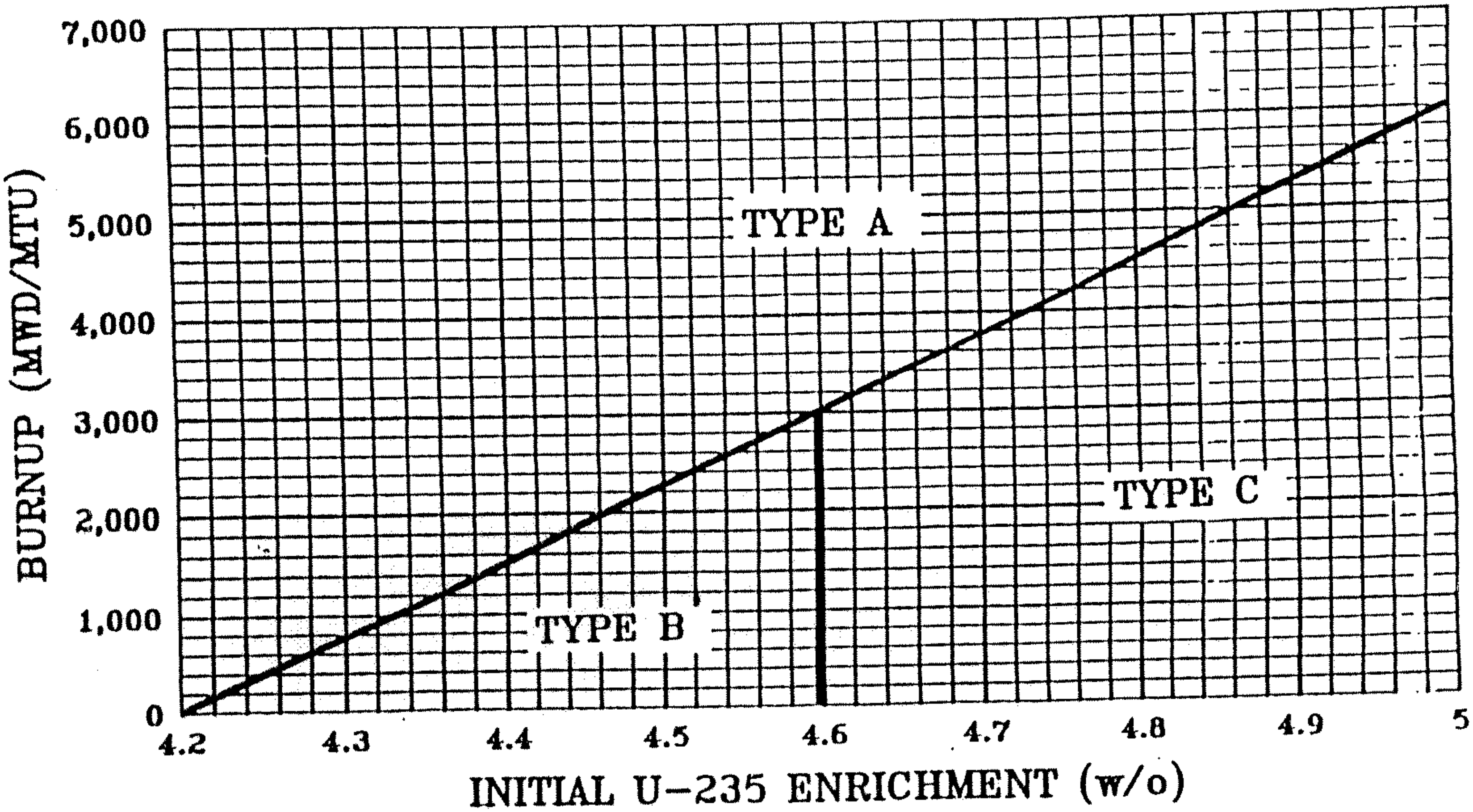
INDIAN POINT 3		FSAR UPDATE
NEW BURNABLE POISON ROD (WET ANNULAR BURNABLE ABSORBER)		
REV. 0	JULY, 1986	FIGURE NO. 3.2-34a



INDIAN POINT 3		FSAR UPDATE
CONTROL ROD DRIVE MECHANISM ASSEMBLY		
REV. 0	JULY, 1982	FIGURE NO. 3.2-35

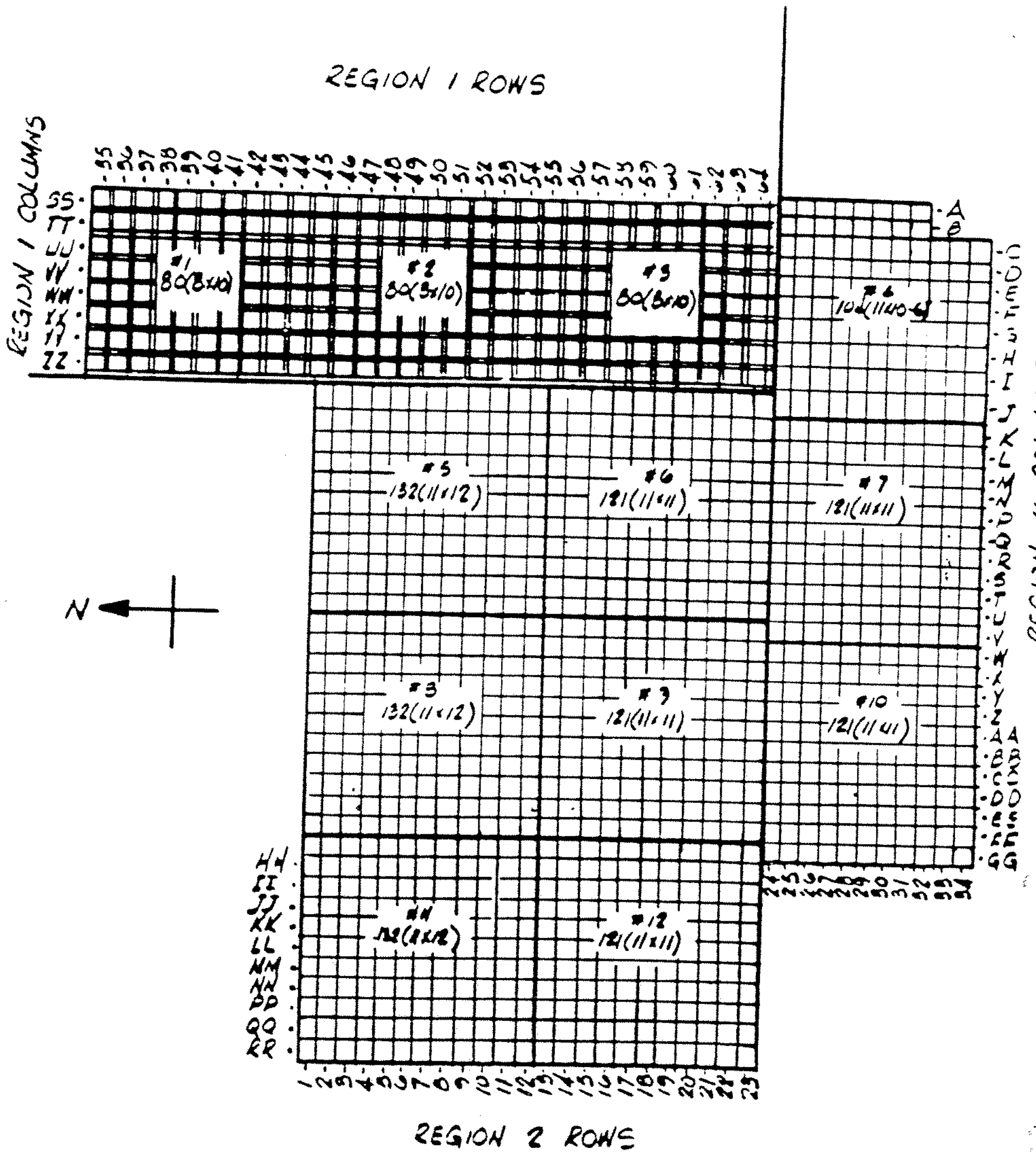


INDIAN POINT 3		FSAR UPDATE
CONTROL ROD DRIVE MECHANISM SCHEMATIC		
REV. 0	JULY, 1982	FIGURE NO. 3.2-36

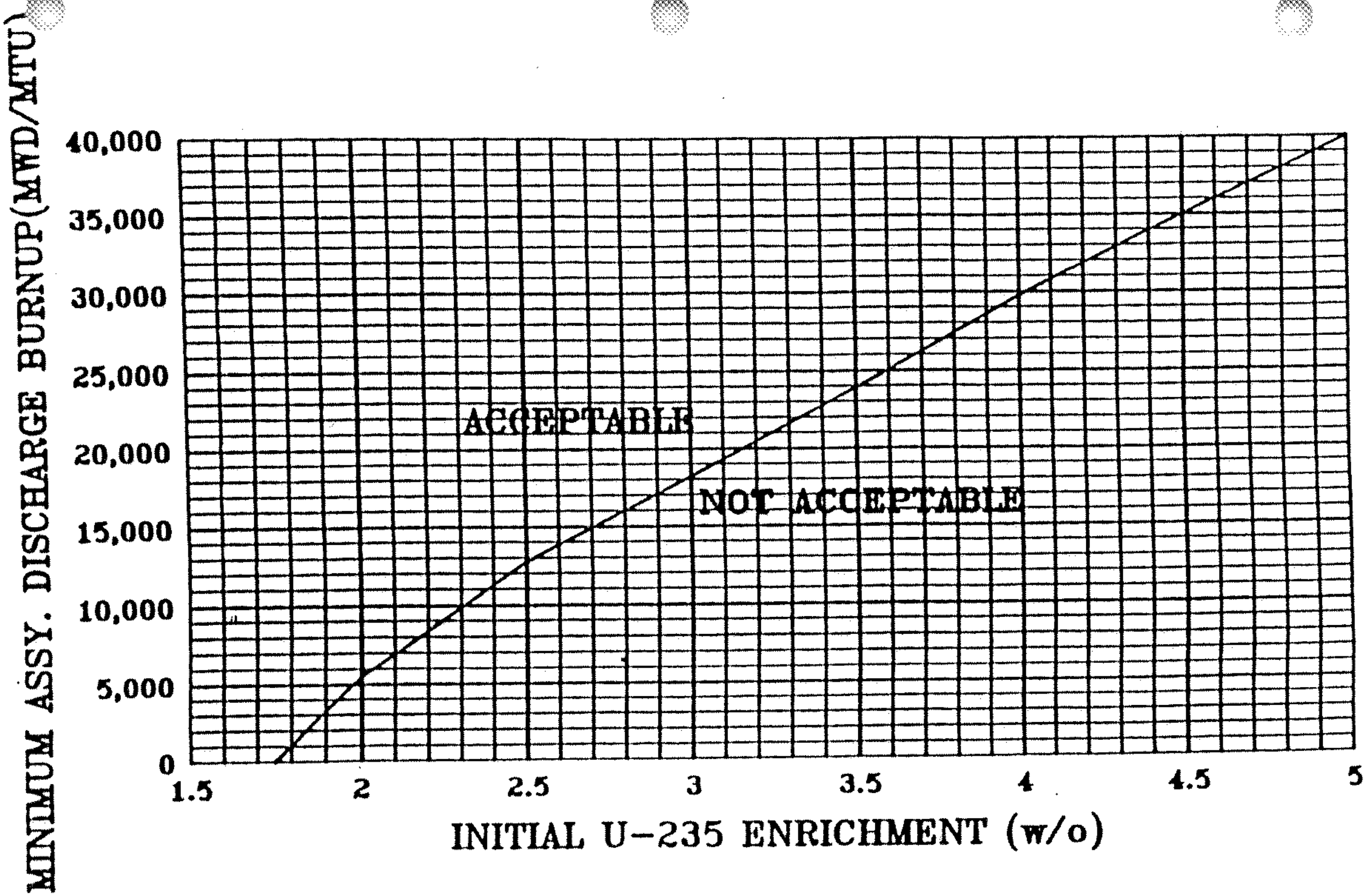


Note: Fresh (unburned) fuel is defined as fuel with a burnup of 0 MWD/MTU.

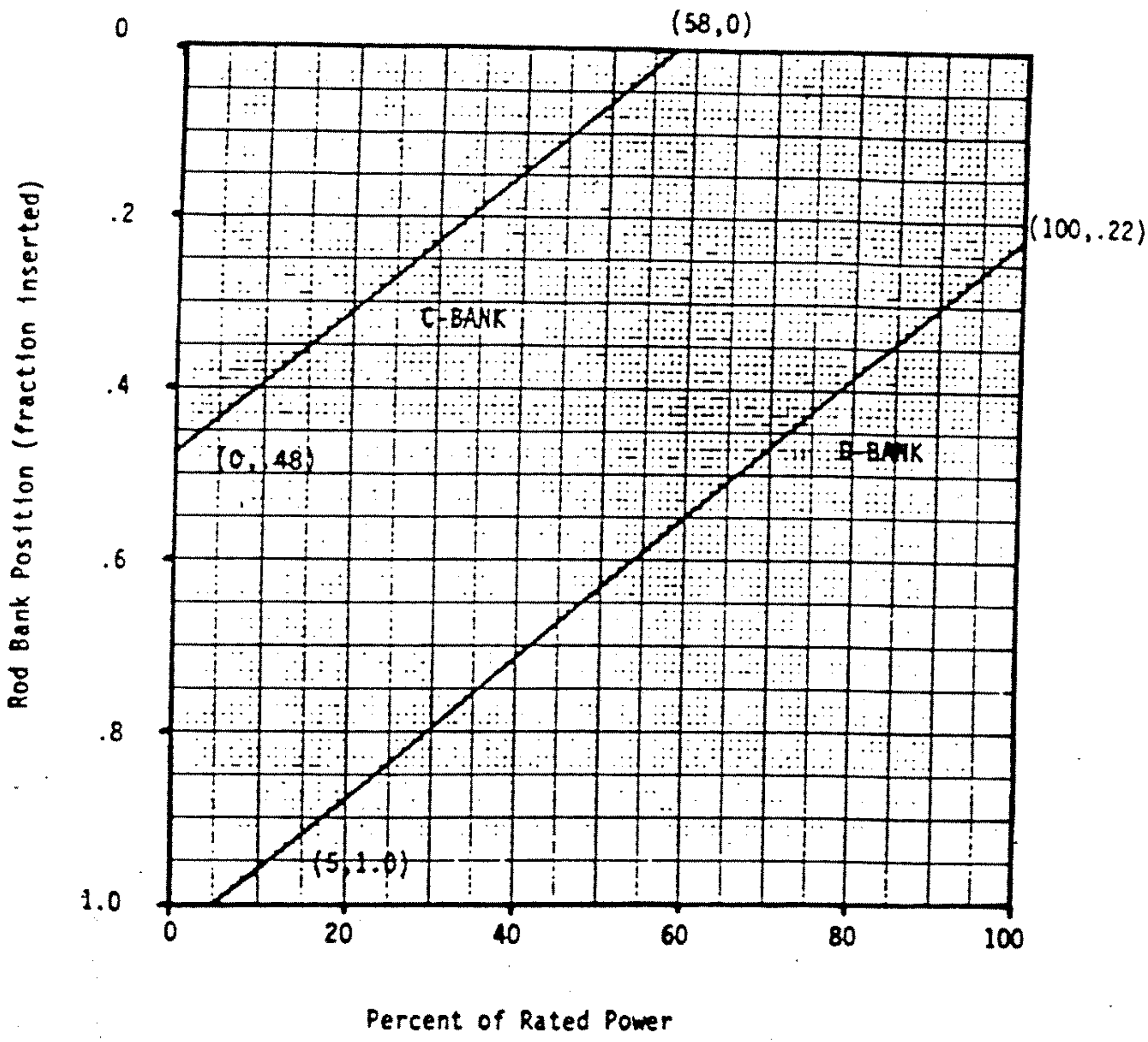
INDIAN POINT 3	FSAR UPDATE
SPENT FUEL PIT REGION 1 TYPE DEFINITION	
REV. 1 DEC 1997	FIGURE NO. 3.2-37A



INDIAN POINT 3	FSAR UPDATE
MAXIMUM DENSITY SPENT FUEL PIT (SFP) RACKS REGIONS AND INDEXING	
REV. 0, JULY 1990	FIGURE NO. 3.2-37B



INDIAN POINT 3 FSAR UPDATE
REGION 2 BURNUP REQUIREMENTS FOR FUEL ASSEMBLY STORAGE IN SPENT FUEL PIT
REV. 0 DEC 1997 FIGURE NO. 3.2-37C



NOTE: Banks A and B are fully withdrawn at zero power

INDIAN POINT 3	FSAR UPDATE
INSERTION LIMITS 100 STEP OVERLAP FOUR LOOP OPERATION (CYCLE 1)	
REV. 1, JULY 1990	FIGURE NO. 3.2-38

CHAPTER 4

REACTOR COOLANT SYSTEM

4.1 DESIGN BASIS

The Reactor Coolant System, shown in Plant Drawings 9321-F-27338 and -27473 [Formerly Figure 4.2-2A & -2B], consists of four similar heat transfer loops connected in parallel to the reactor vessel. Each loop contains a circulating pump and a steam generator. The system also includes a pressurizer, pressurizer relief tank, connecting piping, and instrumentation necessary for operational control.

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 Chapter 3. 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, to acceptable values, its uncontrolled release to the secondary system and to other parts of the plant under conditions of either normal or abnormal reactor behavior. During transient operation the system's heat capacity attenuates thermal transients. 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 coast-down which would result from a loss of flow situation. The layout of the system assures natural circulation capability following a loss of flow to permit decay heat removal without overheating the core. Part of the system's piping is 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 which apply to the Reactor Coolant System are given below.

The General Design Criteria presented and discussed in this section are those which were in effect at the time when Indian Point 3 was designed and constructed. These general design criteria, which formed the bases for the Indian Point 3 design, were published by the Atomic Energy Commission in the Federal Register of July 11, 1976, and subsequently made a part of 10 CFR 50.

The Authority has completed a study of compliance with 10 CFR Parts 20 and 50 in accordance with some of the provisions of the Commission's Confirmatory Order of February 11, 1980. The detailed results of the evaluation of compliance of Indian Point 3 with the General Design Criteria presently established by the Nuclear Regulatory Commission (NRC) in 10 CFR 50 Appendix A, were submitted to NRC on August 11, 1980, and approved by the Commission on January 19, 1982. These results are presented in Section 1.3.

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 of 7/11/67)

The Reactor Coolant System is of primary importance in protecting the health and safety of the public.

Quality standards of material selection, design, fabrication and inspection conformed 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 Section 4.3.1 and Section 4.5. Particular emphasis was placed on the quality assurance of the reactor vessel to obtain material whose properties were uniformly within tolerances appropriate to the application of the design methods of the code.

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 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 of 7/11/67)

All piping, components and supporting structures of the Reactor Coolant System were designed to seismic Class I requirements (SEE Chapter 16). They are capable of withstanding:

a) The Operational Basis Earthquake accelerations within code allowable working stresses.

b) The Design Basis Earthquake accelerations acting in the horizontal and vertical direction simultaneously with no loss of function.

The Reactor Coolant System is located in the Containment, whose design, in addition to being a seismic Class I structure, also considered accidents or other applicable natural phenomena with sufficient margin to compensate for the limits of accuracy on measurements, the quantity of data and the period of time in which historical data have been accumulated on the natural phenomena. Details of the containment design are given in Chapter 5.

Records Requirements

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 of 7/11/67)

Records of the design of the major Reactor Coolant System components and the related engineered safety features components are maintained by the Authority and will be retained throughout the life of the plant as dictated by plant procedures.

Records of fabrication were maintained in the manufacturers' plants as required by the appropriate code. They will be available to the Authority throughout the life of the plant. Construction records are available at the site and/or in the office of the Authority where they will be retained for the life of the plant.

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 of 7/11/67)

The dynamic effects during blowdown following a Loss-of-Coolant Accident were evaluated in the detailed layout and design of the high pressure equipment and barriers which afford missile protection. Support structures were designed with consideration given to fluid and mechanical thrust loadings.

The steam generators are supported, guided and restrained in a manner which 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.

The mechanical consequences of a pipe rupture are restricted by design such that the functional capability of the Engineered Safety Features is not impaired.

4.1.3 Principal Design Criteria

The criteria which apply solely to the Reactor Coolant System are given below.

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 or significant uncontrolled leakage throughout its design lifetime. (GDC 9 of 7/11/67)

The Reactor Coolant System in conjunction with its control and protective provisions was 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.

Fabrication of the components which constitute the pressure retaining boundary of the Reactor Coolant System was carried out in strict accordance with the applicable codes. In addition, there were 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 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 automatically initiated to maintain the required cooling capability and to limit system conditions so that continued safe operation is possible.

The system is protected form overpressure by means of pressure relieving devices, as required by Section III of the ASME Boiler and Pressure Vessel Code.

Isolatable 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.

Monitoring Reactor Coolant Leakage

Criterion: Means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary. (GDC 16 of 7/11/67)

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. This equipment provided 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 was the detection of deviations from normal containment environmental conditions including air particulate activity, radiogas activity, humidity, and condensate runoff. In addition, assuming no operator action, the liquid inventory in the process systems and containment sump can be used for gross indication of leakage. However, sensitivity of the processing systems and containment sump system can be improved with incorporation of a CR alarm (VC Sump Pump running) or operator actions to increase monitoring of the processing system (i.e. sump flow monitor once every 4 hours).

Further details are supplied in Section 4.2.7.

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 of 7/11/67)

The Reactor Coolant Pressure Boundary is capable of accommodating, without further rupture, the static and dynamic loads which would be imposed as a result of a sudden reactivity insertion such as a rod ejection. Details of this analysis are provided in Chapter 14.

The operation of the reactor is such that the severity of an ejection accident is inherently limited. Since core depletion is primarily followed with boron dilution, only the rod cluster control assemblies in the controlled 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 assure that this condition is met.

By using the flexibility in the selection of control rod groupings, radial locations and position 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, 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 was evaluated as a theoretical, thought 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 is adequately protected. (See Chapter 14)

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 provision 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. (GC 34 of 7/11/67)

The Reactor Coolant Pressure Boundary was designed to reduce to an acceptable level the probability of a rapidly propagating type failure.

Assurance of adequate fracture toughness in the reactor vessel material was provided by compliance, insofar as possible, with the requirements for fracture toughness testing included in the Summer 1972 Addenda to Section III of the ASME Boiler and Pressure Vessel Code. In cases where it as not possible to perform all tests in accordance with these requirements, conservative estimates of material fracture toughness were made using information available at the time of design.

Assurance that the fracture toughness properties remain adequate throughout the service life of the plant is provided by a radiation surveillance program.

Safe operating heatup and cooldown limits are established according to Section III, ASME Boiler and Pressure Vessel Code, Appendix G 2000, Protection Against Nonductile Failure, issued in the Summer 1972 Addenda.

Changes in fracture toughness of the core region plates, weldments, and associated weld heat affected zone due to radiation damage are monitored by a surveillance program based on ASTM E-185, Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels(1). The evaluation of the radiation damage in this surveillance program is based on pre-irradiation testing by Charpy V-notch, dropweight test, and tensile specimens and post-irradiation testing of Charpy V-notch, tensile, and wedge opening loading specimens carried out during the lifetime of the reactor vessel. Specimens are irradiated in capsules located near the core mid-height and removed from the vessel at specified intervals. Further details are given in Section 4.1.6.

All pressure containing components of the Reactor Coolant System were designed, fabricated, inspected and tested in conformance with the applicable codes. Further details are given in Section 4.1.7.

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 leak-tight 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 of 7/11/67)

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 and certain 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 pipe within the primary shielding concrete. Further details are given in Section 4.5.

4.1.4 Design Characteristics

Design Pressure

The Reactor Coolant System design and operating pressures together with the safety, power relief and pressurizer spray valves set points, and the protection system set point 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.

Design Temperature

The design temperature for each component was 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.

Seismic Loads

The seismic loading conditions were established by the “Operational Basis Earthquake” and “Design Basis Earthquake”. The former was selected to be typical of the largest probable ground motion based on the site seismic history. The latter was selected to be the largest potential ground motion at the site based on seismic and geological factors and their uncertainties.

For the “Operational Basis Earthquake” (OBE) loading condition, the Nuclear Steam Supply System was designed to be capable of continued safe operation. Therefore, for this loading condition, critical structures and equipment needed for this purpose were required to operate within normal design limits. The seismic design for the “Design Basis Earthquake” (DBE) was intended to provide a margin in design that assured capability to shut down and maintain the nuclear facility in a safe condition. In this case, it was only necessary to ensure that the reactor coolant system components did not lose their capability to perform their safety function. This is 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 Chapter 16. These criteria assure the integrity of the Reactor Coolant System under seismic loading.

For the combination of normal and Operational Basis Earthquake loadings, the stresses in the support structures were kept within the limits of the applicable codes.

For the combination of normal and Design Basis Earthquake loadings the stresses in the support structures were limited to values as necessary to assure their integrity and to maintain the stresses in the Reactor Coolant System components within the allowable limits as previously established.

4.1.5 Cyclic Loads

All components in the Reactor Coolant System were 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 operation. The number of thermal and loading cycles used for design purposes and the bases thereof are given in Table 4.1-8. 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.

To provide the necessary high degree of integrity for the equipment in the Reactor Coolant System, the transient conditions selected for equipment fatigue evaluation were based on a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from normal operation, normal and abnormal load transients. To a large extent, the specific transient operating conditions considered for equipment fatigue analyses were based upon engineering judgement and experience. Those transients were chosen which are representative of transients to be expected during plant operation and which are sufficiently severe or frequent to be of possible significance to component cyclic behavior.

Clearly it is difficult to discuss in absolute terms the transients that the plant will actually experience during the 40 years operating life. For clarity, however, each transient condition is discussed in order to make clear the nature and basis for the various transients.

1. Heatup and Cooldown

   For design evaluation, the heatup and cooldown cases were represented by continuous heatup or cooldown at a rate of 100°F per hour which corresponds to a heatup or cooldown rate under abnormal or emergency conditions. The heatup occurs from ambient to the no load temperature and pressure condition and the cooldown represents the reverse situation. In actual practice, the rate of temperature change of 100°F per hour will not be usually attained because of the Over Pressure Protection System (OPS) discussed in Section 4.3 and other limitations such as:

   a) Criteria for protection against non-ductible failure which establish maximum permissible temperature rates of change, as a function of plant pressure and temperature.

   b) Slower initial heatup rates when using pumping energy only.

   c) Interruptions in the heatup and cooldown cycles due to such factors as drawing a pressurizer steam bubble, rod withdrawal, sampling, water chemistry and gas adjustments.

   The heatup and cooldown rates, administratively imposed by plant operating procedures, are limited to 50°F per hour for normal operation. Ideally, heatup and cooldown would occur only before and after refueling. In practice, additional scheduled and unscheduled plant cooldowns may be necessary for plant maintenance.

2. Unit Loading and Unloading

   The unit loading and unloading cases were conservatively represented by a continuous and uniform ramp power change of 5% per minute between no load and full load. The reactor coolant temperature varies with load as prescribed by the temperature control system. The number of each operation was specified at 14,500 times or once per day for the 40-year plant design life. In practice, the plant is operated at base load.

3. Step Increase and Decrease of 10%

The ±10% step change in load demand is a control transient which is assumed to be a change in turbine control valve opening which might be occasioned by disturbances in the electrical network into which the plant output is tied. The Reactor Control System was designed to restore plant equilibrium without reactor trip following a ±10% step change in demand. The turbine load power range for automatic reactor control initiated from nuclear plant equilibrium conditions, is in the range between 15% and 100% of full load. In effect, during load change conditions, the Reactor Control System attempts to match turbine and reactor outputs so that peak reactor coolant temperature is minimized. Concurrently, the reactor coolant temperature is restored to its programmed set point at a sufficiently slow rate to prevent excessive pressurizer pressure decrease.

Following a step load decrease in turbine load, the secondary side steam pressure and temperature initially increase since the decrease in nuclear power lags behind the step decrease in turbine load. During the same increment of time, the Reactor Coolant System average temperature and pressurizer pressure also initially increase.

Because of the power mismatch between the turbine and reactor, the increase in reactor coolant temperature is ultimately reduced from its peak value to a value below its initial equilibrium value at the inception of the transient. The reactor coolant average temperature set point change is made as a function of turbine-generator load as determined by first stage (inlet) turbine pressure measurement. The pressurizer pressure will also decrease from its peak pressure value and follow the reactor coolant's decreasing temperature trend. At some point during the decreasing pressure transient, the saturated water in the pressurizer begins to flash, which reduces the rate of pressure decrease.

Subsequently, the pressurizer heaters come on to restore the plant pressure to its normal value.

Following a step load increase in turbine load, the reverse situation occurs, i.e., the secondary side steam pressure and temperature initially decrease and the reactor coolant average temperature and pressure initially decrease. The control system automatically withdraws the control rods to increase core power. The decreasing pressure transient is reversed by actuation of the pressurizer heaters and eventually the system pressure is restored to its normal value. The reactor coolant average temperature is raised to a value above its initial equilibrium value at the beginning if the transient.

The number of each operation was specified at 2000 times or 50 per year for the 40-year plant design life.

4. Large Step Decrease in Load

This transient applies to a step decrease in turbine load from full power of such magnitude that the resultant rapid increase in reactor coolant average temperature and secondary side steam pressure and temperature automatically initiates a secondary side steam dump that prevents a reactor shutdown or the lifting of steam generator safety valves.

This transient capability definition brackets the transient design bases used for the Regulating Systems as discussed in Section 7.3.

The number of occurrences of this transient was specified at 200 times or 5 per year for the 40-year plant design life.

5. Reactor Trip from Full Power

A reactor trip from full power may occur for a variety of causes resulting in temperature and pressure transients in the Reactor Coolant System and in the secondary side of the steam generator. This is the result of continued heat transfer from the reactor coolant in the steam generator. The transient continues until the reactor coolant and steam generator secondary side temperatures are in equilibrium at zero power conditions. A continued supply of feedwater and controlled dumping of secondary steam removes the core residual heat and prevents the steam generator safety valves from lifting. The reactor coolant temperature and pressure undergo a rapid decrease from full power values as the reactor protection system causes the control rods to move into the core.

The number of occurrences of this transient was specified at 400 times or 10 per year for the 40-year plant design life.

6. Hydrostatic Test Conditions

The pressure tests outlined below apply to field pressure tests conducted on the erected Reactor Coolant System. The number of tests given below does not include any allowance for pressure tests conducted on a specific component in the manufacturer's shop in accordance with vessel code requirements.

a. Primary Side Hydrostatic Test before Initial Startup at 3110 psig

This hydrostatic test was performed at a minimum water temperature of 100 F, imposed by a reactor vessel material NDTT value of 100 F at beginning of life, and a maximum test pressure of 3110 psig. In this test, the primary side of the steam generator was pressurized to 3110 psig coincident with the secondary side pressure of 0 psig. The Reactor Coolant System was designed on the basis of 5 cycles of this hydro test.

b. Reactor Coolant System Leakage Test

This test is performed at normal operating pressure following each refueling outage prior to startup in accordance with ASME Section XI. Additional tests are performed following repairs, replacements or modifications of the RCS in accordance with ASME Section XI.

7. <u>Loss of Load Without Immediate Turbine or Reactor Trip</u>

This transient applies to a step decrease in turbine load from full power occasioned by the loss of turbine load without immediately initiating a reactor trip, and represents the most severe transient on the Reactor Coolant System. The reactor and turbine eventually trip as a consequence of a high pressurizer level trip initiated by the Reactor Protection System. Since redundant means of tripping the reactor are provided as a part of the Reactor Protection System, transients of this nature are not expected but are included to insure a conservative design.

8. <u>Loss of Flow</u>

This transient applies to a partial loss of flow from full power in which a Reactor Coolant Pump is tripped out of service as a result of a loss of power to the pump. The consequences of such an accident are a reactor and turbine trip, on low reactor coolant flow, followed by automatic opening of the steam dump system and flow reversal in the affected loop. The flow reversal results in reactor coolant being passed at cold leg temperature through the steam generator and being cooled still further. This cooler water then passes through the hot leg piping and enters the reactor vessel outlet nozzles. The net result of the flow reversal is a sizeable reduction in the hot leg coolant temperature of the affected loop.

All components in the Reactor Coolant System were designed to withstand the effects of transients that result in system temperature and pressure changes.

Stress intensity values at all critical points in the reactor vessel due to these excursions of pressure and temperature were determined for each of these transients through systematic analytical procedures. These stress intensity values S_{ij} (i, j = 1,2,3) were plotted against a time interval for each cycle. This plot may represent one or more stress cycles. For each cycle, extreme values of S_{max} and S_{min} were determined. From these values, the largest S_{alt} (alternating stress intensity) was found.

For this largest value of S_{alt}, an allowable number of cycles (N) was determined through design fatigue curves established for different materials. The ratio of design cycles (n) to allowable cycles (N) gave the usage factor ^{u}i (i = 1,2,3…, n). Usage factor was determined in this manner for all transients. The cumulative usage factor was determined by summing the individual usage factors. The cumulative usage factor ($U = {}^{u}1^{+}\ {}^{u}2^{+u}3^{+}\ldots^{+u}n$) was never allowed to exceed a value of 1.0.

Although loss of flow and loss of load transients were not included in the tabulation, since the tabulation was only intended to represent normal design transients, the effects of these transients were analytically evaluated and were included in the fatigue analysis for primary system components.

Over the range from 15% of full power up to and including, but not exceeding, 100% of full power, the Reactor Coolant System and its components were designed to accommodate 10% of full power step changes in plant load and 5% of full power per minute ramp changes without reactor trip. The Reactor Coolant System can accept a complete loss of load from full power with reactor trip. In addition, the turbine bypass and steam dump systems make it possible to accept a 10% to 50% change in load, at a maximum turbine unloading rate of 200% per minute from approximately 100% load with steam dump without reactor trip (load rejection capability depends on full power Tavg; see Section 7.3.2). However, for component stress analysis purposes, this was analyzed as a step change in load from 100% to 50% load.

4.1.6 Service Life

The service life of the 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 which is exposed to a significant level of neutron irradiation and it is, therefore, the only component which is subject to material radiation damage effects.

The NDTT shift of the vessel material and welds, due to radiation damage effects, is monitored by a radiation damage surveillance program which conforms with ASTM-185 standards.

Reactor vessel design was 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 were established for the 40-year design life. These operating conditions included 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

All pressure containing components of the Reactor Coolant System were of U.S. manufacture and were designed, fabricated, inspected and tested in conformance with the applicable codes listed in Table 4.1-9.

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 combine with the primary steady state stresses.

References

1) WCAP-8475, “Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program”, Westinghouse Class 3, January 1975.

TABLE 4.1-1

REACTOR COOLANT SYSTEM PRESSURE SETTINGS

	Pressure, psig
Design Pressure	2485
Operating Pressure (at pressurizer)	2235
Safety Valves	2485
Power Relief Valves	2335
Pressurizer Spray Valve (Begin to open)	2260
(Fully open)	2310
High Pressure Trip	2385
Low Pressure Trip	1800
Hydrostatic Test Pressure	3110

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 O.D. to top of Control Rod Mechanism Housing)	43-9 $\frac{11}{16}$
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$\frac{1}{16}$
OD of Flange, in	205
ID at Shell, in	173
Inlet Nozzle ID, in	27½
Outlet nozzle ID, in	29
Clad Thickness, min, in	$\frac{5}{32}$
Lower Head Thickness, min, in	5$\frac{5}{16}$
Vessel Belt-Line Thickness, min, in	8$\frac{5}{8}$
Closure Head Thickness, in	7
Reactor Vessel Inlet Temperature, F	517.3
Reactor Vessel Outlet Temperature, F	611.7
Reactor Coolant Flow, lb/hr	1.388×10^8

TABLE 4.1-3

PRESSURIZER & PRESURIZER 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
Steam Volume, Full Power, ft^3	720
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/valve	179,000
Safety Valves	
Number	3
Set Pressure, psig	2485
Capacity, lb/hr Saturated steam/valve	420,000
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

* 60% of net internal volume

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/Stem, °F	650/600
Reactor Coolant Temperature to Steam Generator, °F	602.5
Reactor Coolant Temperature from Steam Generator, °F	540.7
Reactor Coolant Flow, (gpm)	88,600
Total Heat Transfer Surface Area, ft^2	43,467
Heat Transferred at Design (Licensed) Power Level, (807.5 Mwt) Btu/hr	2755×10^6
Steam Conditions at Full Load, Outlet Nozzle:	
Steam Flow, 10^6 lb/hr	3.2914 / 3.4906
Steam Temperature, °F	510.9 / 510.8
Steam Pressure, psia	750.4 / 749.6
Feedwater Temperature, °F	390.0 / 433.6
Overall Height, ft-in	63 – 1.62
Shell OD, upper/lower, in	166/127
Shell Thickness, upper/lower, (minimum), in	3.5/2.62
Number of U-tubes	3214
U-tube Diameter, in	0.875
Tube Will Thickness, (Average) in	0.050
Number of manways/ID, in	4/16
Number of handholes/ID, in	6/6
Number of inspection ports/ID, in	1/3

*The values on this table apply at the design full-load power level of 3230 MWt (four loops) unless noted otherwise. The values shown are for a single steam generator.

TABLE 4.1-4
(Cont.)

STEAM GENERATOR DESIGN* DATA

	3230 MWt	Zero Power
Reactor Coolant Water Volume ft^3 (cold)	924.1	924.1
Primary Side Fluid Heat Content, Btu	24.1×10^6	23.86×10^6
Secondary Side Water Volume, ft^3**	1626.9	2666
Secondary Side Steam Volume, ft^{3**}	3100	2061
Secondary Side Fluid Heat Content, Btu**	45.05×10^6	72.8×10^6

* The values on this table apply to the design full load power level of 3230 MWt (four loops) unless noted otherwise. The values shown are for a single steam generator.

** These values correspond to a normal operating water level of 52% narrow range span, and may vary with changes in water level.

*** Values provided for 3230 MWt correspond to a feed temperature of 433.6^{o}F.

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
Net Positive Suction Head, Ft	170
Developed Head, ft	272
Capacity, gpm	89,700
Seal Water Injection, gpm	8
Seal Water Return, gpm	3
Pump discharge Nozzle ID, in	27½
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 Thermalastic Epoxy
Phase	3
Frequency, cps	60

TABLE 4.1-5
(Cont.)

REACTOR COOLANT PUMPS DESIGN DATA

Starting Current, amp	2950
Input (hot reactor coolant), kW	4250
Input (cold reactor coolant) kW	5600
Power, hp (nameplate)	6000

TABLE 4.1-6

REACTOR COOLANT PIPING DESIGN DATA

Reactor Inlet Piping, ID, in	27½
Reactor Inlet Piping, nominal thickness, in	2.375
Reactor Outlet Piping, ID, in	29
Reactor Outlet Piping, nominal thickness,	2.50
Coolant Pump Suction Piping, ID, in	31
Coolant Pump Suction Piping, nominal thickness, in	2.625
Pressurizer Surge Line Piping, ID, in	11.5
Pressurizer Surge Line Piping, nominal thickness,	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	1.1
Across Vessel, including nozzles	46.7
Across Hot Leg	1.3
Across Steam Generator	32.3
Across Pump Suction Leg	3.0
Total Pressure Drop	84.4

TABLE 4.1-8

THERMAL AND LOADING CYCLES

Transient Condition	Design Cycles+**	Loading Conditions
1. Plant heatup at 100°F per hour	200 (5/yr*)	Normal
2. Plant cooldown at 100°F per hour	200 (5/year)	Normal
3. Plant loading at 5% of full power per minute	14,500 (1/day)	Normal
4. Plant unloading at 5% of full power per minute	14,500 (1/day)	Normal
5. Step load increase of 10% of full power (but not to exceed full power)	2,000 (50/year)	Normal
6. Step load decrease of 10% of full power	2,000 (50/year)	Normal
7. Step load decrease of 50% of full power	200 (5/year)	Normal
8. Reactor trip	400 (10/year)	Upset
9. Hydrostatic test at 3110 psig pressure, 100 F temperature	5 (pre-operational)	Test
10. Hydrostatic test at 2485 psig pressure and 400 F temperature	200 (post-operational)	Test
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		
12. Loss of load, without immediate turbine trip or reactor trip	80 (2 /year)	Upset
13. Partial loss of flow, one pump only	80 (2/year)	Upset

TABLE 4.1-8
(Cont.)

THERMAL AND LOADING CYCLES

	Transient Condition	Design Cycles+**	Loading Conditions
14.	Operating Basis Earthquake (OBE)	5++	Upset
15.	Design Basis Earthquake (DBE)	1++	Faulted

\+ Estimated for equipment design purposes (40-year life) and not intended to be an accurate representation of actual transients or to reflect actual operating experience. See Section 4.1.5.

* This transient includes pressurizing to 2235 psig.

** Piping and Valves included in the Reactor Coolant System boundary are designed, analyzed and fabricated in accordance with their applicable codes.

++ The upset conditions include the effect of the specified earthquake for which the system must remain operational or must regain its operational status. The faulted conditions include the earthquake for which safe shutdown is required. For fatigue studies, Class I components were analyzed for five OBE's and one DBE in addition to other fatigue producing events in the above listed four loading conditions. Each earthquake is considered to produce ten peak stress magnitudes.

TABLE 4.1-9

REACTOR COOLANT SYSTEM – CODE REQUIREMENTS

The edition of the ASME Code, Section III and addenda to which the major components in the Reactor Coolant System were designed and fabricated are:

Component	Code Edition	Class	Applicable Addenda
Reactor Vessel	1965	A	Winter 1965 and Code Cases 1332, 1335, 1339, 1359
Rod Drive Mechanism	1965	A	Summer 1966
Replacement Steam Generators			
- Tube side	1983	1	Summer 1984
- Shell side	1983	1	Summer 1984
Pressurizer	1965	A	Summer 1966
Pressurizer Relief Tank	1965	C	Summer 1966
Pressurizer Safety Valves	1965		Summer 1966
Reactor Coolant Pump Volute	- Westinghouse design per ASME III Article 4.		

In addition the reactor coolant pipe was designed to ANSI B31.1 – 1955.

The loop 32 hot leg elbow replaced during the cycle 6/7 refueling outage in conjunction with the steam generator replacement was designed and fabricated to ASME Code Section III, 1983 edition, class 1 requirements, including addenda through summer 1984, although the elbow was required to meet only B31.1-1955 criteria.

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 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 and auxiliary system connections is shown in Plant Drawing 9321-F-27383 and -27473 [Formerly Figures 4.2.2A and 4.2-2B]. A schematic flow diagram denoting principal parameters under normal steady state full power operating conditions is shown in Figure 4.2-2.

The containment boundary shown on the flow diagram in Plant Drawing 9321-F-27473 [Formerly Figure 4.2-2B] 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. A power level of 100% rated output for 80% of the time is considered an estimate of ideal operation over the service life of the system.

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

Reactor Vessel

The reactor vessel is cylindrical in shape with a hemispherical bottom and a flanged and gasketed removable upper head. The vessel was 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 ninety-five percent of the total coolant flow is effective for heat removal from the core. The remainder of the flow includes the flow through the RCC 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 which result from heat generated by the absorption of gamma energy. This protection is further described in Section 3.2.3.

Forty-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 O-rings seal the reactor vessel when the reactor closure head is bolted in place. A leakoff connection is provided between the two O-rings to monitor leakage across the inner O-ring. In addition, a leakoff connection is also provided beyond the outer O-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 were 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 are examined at selected intervals to evaluate reactor vessel material NDTT changes as described in Section 4.5.2.

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 3.2-23.

Reactor vessel design data are listed in Table 4.1-2.

<u>Pressurizer</u>

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 basis for establishing the pressure relieving capacity for the Reactor Coolant Pressure Boundary is the loss of 100 percent of the heat sink i.e. steam flow to the turbine, when the thermal output of the reactor is at 100 percent of its rated power, with appropriate credit taken for operation of the secondary system safety valves and the Reactor Protection System.

Overpressure protection is described in Section 4.3.4.

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 was 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 the hot leg of a reactor coolant loop. During a positive surge, caused by a decrease in plant load, the pressurizer 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 set point 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 switch in the Control Room. Spray valve position is monitored by indicating lights in the Control Room. A small continuous spray flow is provided to assure that the pressurizer liquid is homogeneous with the coolant and to prevent excess cooling of the spray piping. Isolation valves and bypasses are provided at each pressurizer spray valve. The outer bypass, which was provided to allow a small continuous spray flow to the pressurizer during on line maintenance of the valves, was removed by a subsequent modification.

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.

<u>Steam Generators</u>

Each loop contains a Westinghouse Model 44F 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 were designed and manufactured in accordance with Section III (Nuclear Vessels) of the ASME Boiler and Pressure Vessel Code and were installed during the cycle 6/7 refueling outage as replacements for the original Model 44 steam generators.

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. Primary side manways are provided to permit access into the channel head and to the U-tubes.

Feedwater to the steam generator enters just above the top of the U-tubes through an elevated feedwater ring. The water exits the feedwater ring through inverted J-tubes installed on the top of the ring. This minimizes the potential for damaging water hammer events to occur. The water flows downward through an annulus between the tube wrapper and the shell and then upward through the tube bundle where it is converted to a steam-water mixture.

The steam-water mixture from the tube bundle passes through a low pressure drop, swirl vane modular primary separator assembly which imparts a centrifugal motion to the mixture and reduces the water content in the mixture. The separated water passes through a perforated section of riser tube, impinges on a small external downcomer sleeve, and combines with the feedwater for another pass through the tube bundle.

The remaining higher quality steam-water mixture rises through additional secondary separators which limit the moisture content of the steam to one tenth of one percent or less under all design load conditions. The steam outlet nozzle is equipped with an integral steam flow limiting device. Manways, handholes, and an inspection port are provided for maintenance and inspection of the secondary side.

The steam generator pressure boundary is constructed of low alloy steel. The heat transfer tubes are thermally treated Inconel 690. The interior surface of the channel head 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-tubesheet joint is welded.

Reactor Coolant Pumps

Each reactor coolant loop contains a vertical single stage centrifugal pump which 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. Auxiliary system piping connections to the reactor coolant pumps are shown in Plant Drawing 9321-F-27383 [Formerly Figure 4.2-2A]. 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% design flow introduces no operational restrictions, since the pumps operate at full speed.

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 out through passages in the diffuser and out through a discharge nozzle in the side of the casing. The motor-impeller can be removed from the casing for maintenance or inspection without removing the casing from 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 which directs the controlled leakage out of the pump, and a third seal which 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 which leaks through the second seal, is also collected and removed from the pump. Pump seal injection flow is indicated in the Control Room as described in Section 9.2

Component cooling water is supplied to the motor bearing oil coolers and the thermal barrier cooling coil.

Each of the reactor coolant pumps is provided with an anti-reverse rotation device in the motor. This anti-reverse mechanism consists of eleven pawls mounted on the outside diameter of the flywheel, a serrated ratchet plate mounted on the motor frame, a spring return for the ratchet plate, and two shock absorbers.

After the motor has come to a stop, one pawl engages the ratchet plate and, as the motor tends to rotate in the opposite direction, the ratchet plate also rotates until stopped by the shock absorbers. The rotor remains in this position until the motor is energized again. After the motor has come up to speed, the ratchet plate is returned to its original position by the spring return.

When the motor is started, the pawls drag over the ratchet plate until the motor reaches approximately 80 revolutions per minute. At this time, centrifugal force holds the pawls in the elevated position until the speed falls below the above value.

Normal loop flow, with all pumps running is approximately 9474 pounds/second (per loop). The following table indicates the loop flow distribution for less than four pumps running:

	Running Pump Loop Flow	Stopped Pump Loop Flow	Core Flow
3 Pumps Running*	10,244 (lb/sec)	-3,048 (lb/sec)	27,686 (lb/sec)
2 Pumps Running*	10,771 (lb/sec)	-1,944 (lb/sec)	17,655 (lb/sec)
1 Pump Running*	11,035 (lb/sec)	- 913 (lb/sec)	8,295 (lb/sec)

The above table assumes that the anti-reverse flow mechanism functions properly.

In the highly unlikely event that the anti-reverse flow device does not function properly and allows the pump to rotate freely in reverse, the following flow distributions would be realized:

	Running Pump Loop Flow	Stopped Pump Loop Flow	Core Flow
3 Pumps Running*	10,360 (lb/sec)	-4,837 (lb/sec)	26,244 (lb/sec)
2 Pumps Running*	10,856 (lb/sec)	-2,924 (lb/sec)	15,863 (lb/sec)
1 Pump Running*	11,067 (lb/sec)	-1,313 (lb/sec)	7,127 (lb/sec)

Each loop is provided with flow and temperature instruments. The loop temperature information is valid for all loops, active or inactive. The loop flow information is valid only for active loops. Inactive loop flow can be estimated by calculating total flow from a heat balance and then subtracting the active loop flow.

It is important to note that the Reactor Control and Protection Systems were designed to prevent abnormally induced transients or abnormal operating conditions. Plant operating instructions describe the various combinations of plant power level and reactor coolant flows which will result in plant trips.

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.

*NOTE: Reactor power operation is not permitted with less than four loops operating.

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 was also available to confirm the seal design.

The primary coolant pump flywheel is shown in Figure 4.2-8. The flywheel was fabricated from two rolled, vacuum-degassed, ASTM A-533 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 material specification is ASTM A-533 Grade B Class I, plus supplementary material testing requirements and Charpy tests, as detailed in Section 4.3.

Pressurizer Relief Tank

Principal design parameters of the pressurizer relief tank are given in Table 4.1-3. It is shown on Figure 4.2-9.

Steam and water discharge from the power relief and safety vales pass to the pressurizer relief tank which is partially filled with water at or near ambient containment conditions. 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 the full power pressurizer steam volume.

The tank is protected against a discharge exceeding the design value by two rupture discs which discharge into the Reactor Containment. The rupture discs 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 discs setting) is twice the calculated pressure resulting from the maximums safety valve discharge described above. This margin is to prevent deformation of the disc. The tank and rupture disc 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 set point pressure at full flow.

The pressurizer relief tank, by means of its connection to the Waste Disposal System, provides a means for removing any non-condensable gases 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.

Piping

The general arrangement of the Reactor Coolant System piping is shown on the plant layout drawing in Chapter 1. Piping design data are presented in Table 4.1-6.

The austenitic stainless steel reactor coolant piping and fittings which make up the loops are 29 inch ID in the hot legs, 27.5 inches ID in the cold legs and 31 inches ID between each loop's steam generator outlet and its reactor coolant pump suction. The pressurizer relief line, which connects the pressurizer safety and relief valves' outlet to the inlet nozzle flange on the pressurizer relief tank was 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 relief and safety valves.

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)
2) Both ends of the pressurizer surge line,
3) Pressurizer spray line connection to the pressurizer, and
4) Charging lines and auxiliary charging line connections.

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 or migrate the potential for valve stem leakage due to modulating service.

Component Supports

The support structures for the reactor coolant system components are described in Appendix 4B.

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 Plant Drawing 9321-F-27473 [Formerly Figure 4.2-11] and the valve design parameters are given in Table 4.1-3.

A reactor head vent system is also provided to exhaust non-condensible gases or steam from the primary system that could inhibit natural circulation core cooling.

Two power operated relief valves (PORVs) and three code safety valves (SVs) are provided to protect against pressure surges which are beyond the pressure limiting capacity of the pressurizer spray. The PORVs also operate from the Overpressure Protection System described in Section 4.3 to prevent RCS pressure from exceeding the limits of Appendix G of Section III of the ASME Pressure Vessel and Boiler Code (1972 Summer Addenda) and ASME Code Case N-514.

The instrument N_2 system for the PORVs is tapped from the N_2 supply line to the safeguards accumulators. The actual take-off point for this N_2 system is downstream of the pressure regulator valve NNE-863. Each PORV is supplied with its own accumulator. The PORV accumulators individually hold 6 cu ft of N_2 at a minimum pressure of 550 psig. During low temperature shutdown operations, the Overpressure Protection System requires an N_2 supply of sufficient capacity which, in case of loss of the main N_2 supply, can support the number of PORV cycles resulting from an overpressure event of 10 minute duration. This N_2 supply is provided by one Safety Injection Accumulator having its associated N_2 fill valve blocked open.

A platform within the pressurizer missile shield provides access to the pressurizer safety valves for testing purposes. This access platform does not interfere with maintenance which may require removal of the safety and/or power operated relief valves.

The pressurizer relief tank is protected against a steam discharge exceeding the design pressure valve by two rupture discs which discharge into the Reactor Containment. The rupture disc relief conditions are given in Table 4.1-3.

The design criteria require that transient flow loads from valve discharge be combined with deadweight loads, seismic loads, and loads due to internal pressure. Thrust, bending, and torsion (as applicable) from each of these loads was determined. Primary stress in the supports under this load combination was limited to 1.33 times allowable stress or rated load in accordance with Manufacturer’s Standardization Society Standard MSS-SP-58 for standard supports or AISC-69 requirements for non-standard supports. In those instances where the integrity of supports is also dependent on reinforced concrete anchorage, the concrete behavior limits and anchor bolt behavior were in accordance with I.E. Bulletin 79-02. (8)

In response to NUREG-0737, Item II.D.1, the Electric Power Research Institute (EPRI), under contract with the Westinghouse Owners Group, performed full scale tests on typical safety and relief valves and associated piping configurations under simulated transient conditions.

EPRI issued the results of the test program to participating utilities along with evaluation guidelines for applying the test results to plant specific pressurizer safety and relief valves and associated piping. These guidelines have been used to evaluate the adequacy and integrity of the pressurizer safety and relief valve piping during plant transients which challenge the safety and relief valves.

The evaluation of the pressurizer safety and relief valve piping considered the following two transient cases:

Case 1: Sequential actuation of the power operated relief valves (PORV’s) and safety valves (SV’s) at their respective set pressures with pressurizer pressure increasing at a rate of 130 psi/sec and loop seal temperature at the inlet to the SV’s of 260°F.

Case 2: Block valves upstream of the PORV's closed and only SV's open at their set pressures with pressurizer pressure increasing at a rate of 144 psi/sec and loop seal temperature at the inlets to the SV's of 260°F.

The peak transient pressures, temperatures and flow rates in the discharge piping were computed using the RELAP-5 MOD 1, Cycle 14 computer code which has been verified during the EPRI test program to yield results which closely match actual pressures, temperatures and flow rates witnessed during the testing. A post processor code (FORCE) was used to translate the thermal hydraulic results of the RELAP code into transient forces on the piping and a structural code (STARDYNE) was used to determine the resulting pipe stresses and support transferred loads. In addition, the structural code NUPIPE-II was used to determine the pipe stresses and support transferred loads due to deadweight loads, normal pressure loads, thermal loads and seismic loads.

These evaluations indicated the need to upgrade the original design of the pressurizer safety and relief valve discharge system for purposes of complying with NUREG-0737 Item II.D.1. This upgrade was accomplished by a series of modifications which included: discharge piping support upgrade, discharge piping branch connection reinforcement, replacement of pressurizer safety valve internals, and re-routing of pressurizer safety valve loop seal drain lines to provide for continuous drainage back to the pressurizer. Continuous drainage is provided to preclude the occurrence of water hammer in the discharge piping in the event the valves open.

These modifications served to reduce the stresses in the safety and relief valve discharge piping to acceptable levels which are below the stress limits in the EPRI guidelines for the various load combinations considered. In addition, the stresses in the piping due to thermal loads are below the ANSI B31.1 1967 Power Piping Code stress limits. The constant supports were designed and fabricated in accordance with ANSI B31.1 whereas the hydraulic restraints were designed and fabricated in accordance with the ANSI B31.7 Nuclear Power Piping Code. Finally, the weld reinforcing pads were manufactured and installed in accordance with the requirements of ANSI B31.1.

Subsequent to the analyses and modifications described above, various pipe supports were further upgraded based on reanalyses of the pressurizer PORV inlet and outlet piping and supports. These reanalyses were performed to assess the effect to upgrading the PROV block valve operators with larger and heavier operators to enhance their design capabilities. Upgrade of the supports was deemed necessary to ensure that the integrity of the piping they support and the safety equipment they service will remain intact and to demonstrate that the design criteria used for Seismic Class I pipe supports is not violated with the added loading.

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 concrete shield walls. These walls provide shielding to permit access into the Containment during full power operation for inspection and maintenance of miscellaneous equipment. These shielding walls also provide 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 was 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 presented in Chapter 5.

4.2.5 Materials of Construction

Each of the materials used in the Reactor Coolant System was selected for the expected environment and service conditions. The major component materials are listed in Table 4.2-1.

All reactor coolant system materials which are exposed to the coolant are corrosion-resistant. They consist of stainless steels and Inconel, and they were chosen for specific purposes at various locations within the system for their superior compatibility with the reactor coolant. The chemical composition of the reactor coolant is maintained within the specification given in Table 4.2-2. Reactor coolant chemistry is further discussed in Section 4.2.8.

The water in the secondary side of the steam generators is held within the appropriate chemistry specifications to control deposits and corrosion inside the steam generators. Secondary side chemistry is monitored as described in the Technical Specifications.

The phenomena of stress-corrosion cracking and corrosion fatigue are not generally encountered unless a specific combination of conditions is present. The necessary conditions are a susceptible alloy, an aggressive environment, a stress, and time.

It is characteristic of stress corrosion that combinations of alloy environment which result in cracking are usually quite specific. Environments which 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. In the presence of this environment, stress-corrosion cracking can occur in stainless steels having high residual stresses resulting from manufacturing procedures.

The use of lead in the materials of the secondary side of Indian Point 3 was minimized to the practical limit of that occurring as trace elements in alloys and as such is insignificant.

All external insulation employed in the Reactor Coolant System is compatible with the structural materials of the component. The cylindrical shell exterior and closure flanges to the reactor vessel are insulated with metallic reflective insulation. The closure head is insulated with low halide-content insulating material.

Assurance of adequate fracture toughness of the reactor vessel is provided by compliance with the requirements for fracture toughness included in the Summer 1972 Addenda to Section III of the ASME Boiler and Pressure Vessel Code. In cases where it is not practicable to perform all tests in accordance with this Code, conservative estimates of material fracture toughness were made to demonstrate compliance with the Code requirements.

The techniques used to measure and predict the integrated fast neutron (E greater than 1 MeV) fluxes at the sample locations are described in Appendix 4A. The calculation method used to obtain the maximum fast neutron 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 is obtained from the measured sample exposure by appropriate application of the calculated azimuthal neutron flux variation.

The maximum integrated fast neutron exposure of the vessel inner surface was computed to be 0.922×10^{19} n/cm^2 at license expiration, or 27.1 effective full power years, at a power of 3216 MWt). Similarly, the maximum integrated fast neutron exposure at the ¼T location was computed to be 5.5×10^{18} n/cm^2 at EOL (11). With this exposure the end-of-life ΔRT_{NDT} was estimated to be 170.6°F, and RT_{PTS} temperature was not expected to be over 268°F, which is below the NRC screening criteria of 270°F. Each core redesign is evaluated to assure that leakage is less than assumed in analyses to predict the effect of neutron embrittlement.

The first reactor vessel material surveillance capsule was removed during the 1978 refueling outage. This capsule has been tested by Westinghouse Corporation and the results have been evaluated and reported. (1) Based on the Westinghouse evaluation, heatup and cooldown curves were developed for up to 9.26 EFPYs of reactor operation.

Generic Letter 88-11 requested that licensees use the methodology of Regulatory Guide 1.99, revision 2. “Radiation Embrittlement of Reactor Vessel Material”, to predict the effect of neutron radiation on reactor vessel materials as required by paragraph V.A of 10 CFR part 50, Appendix G. Capsules X and Z were analyzed (9)(10) and new pressure-temperature curves were developed using this methodology.

The maximum shift in RT_{NDT} after 20 EFPYs of operation is projected to be 230.1°F at the 1/4T and 188.8°F at the 3/4T vessel wall locations for Plate B2803-3 the controlling plate. Plate B2803-3 was also the controlling plate for the operating periods of 2 EFPYs, 9 EFPYs, 11.00 EFPYs, 13.3 EFPYs and 16.2 EFPYs.

Heatup and cooldown limit curves are calculated using the most limiting value or RT_{NDT} at the end of 20 EFPYs. This service period assures that all components in the Reactor Coolant System will be operated conservatively in accordance with Code recommendations.

The highest RT_{NDT} of the core region material is determined by adding the radiation induced ΔRT_{NDT} for the applicable time period to the original RT_{NDT} shown in Table 4.4.1.

Changes in fracture toughness of the core region plates or forgings, weldments, and associated heat treated zones due to radiation damage will continue to be monitored by a surveillance program which conforms with ASTM E-185, Recommended Practice for

Surveillance Tests for Nuclear Reactor Vessels. (3) Specimens are irradiated in capsules located near the core mid-height and removed from the vessel at specified intervals. To this extent, the second, third and forth reactor vessel material surveillance capsules were removed in 1982, 1987 and 2003 respectively. These capsules have been tested and the results have been evaluated and reported. (4)(5)(6)(9)(10) No change to the technical specification heatup and cooldown limit curves was required as a result of the Capsule Y and Capsule Z analyses. However, the new analytical methodology introduced in Reg. Guide 1.99, Rev. 2, resulted in the generation of new heatup-cooldown curves for 9, 11, 13.3, 16.2 and 20 EFPY.

The evaluation of the radiation damage in this surveillance program is based on pre-irradiation testing by Charpy V-notch, drop weight and tensile specimens and post-irradiation testing by Charpy V-notch, tensile, and wedge opening loading specimens carried out during the lifetime of the reactor vessel.

Information from the radiation surveillance program described above is also used for evaluations required by the NRC's Pressurized Thermal Shock (PTS) Rule (10 CFR 50.61). Using the prescribed PTS Rule methodology, RTPTS values were generated for allbeltline region materials of the reactor vessel as a function of several fluence values and pertinent vessel lifetimes. All of the RTPTS values remain below the NRC screening values for PTS using the projected fluence exposure through the expiration date of the operating license. (7)

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 normal system heat-up rate is ≤ 60°F/hr, and cooling rate is ≤ 50°F/hr. Sufficient electrical heaters are installed in the pressurizer to permit a heatup rate, starting with a minimum water level, of 55°F/hr. This rate takes into account the small continuous spray flow provided to maintain the pressurized liquid homogeneous with the coolant.

The fastest cooldown rates which result form the hypothetical case of a break of a main stream line are discussed in Chapter 14. Refer to Section 4.4 for further information.

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) Two radiation sensitive instruments provide the capability for detection of leakage from the Reactor Coolant System. The containment air particulate monitor is quite sensitive to low leak rates. The containment radiogas monitor is much less sensitive but can be used as a backup to the air particulate monitor.

2) A third instrument used in leak detection is the humidity detector. This provides a means of measuring overall leakage from all water and steam systems within the containment but furnishes a less sensitive measure. The humidity monitoring method provides backup to the radiation monitoring methods.

3) A leakage detection system is included which determines leakage losses from all water and steam systems within the Containment including that from the Reactor

Coolant System. This system collects and measures moisture condensed from the containment atmosphere by the cooling coils of the fan cooling units. It relies on the principle that all leakages up to sizes permissible with continued plant operation will be evaporated into the containment atmosphere. This system provides a dependable and accurate means of measuring integrated total leakage, including leaks from the cooling coils themselves which are part of the containment boundary.

4) An increase in the amount of coolant makeup water which is required to maintain normal level in the pressurizer. Assuming no operator action, the increase in containment sump level provides a less sensitive means of detecting leakage. However, sensitivity of the processing systems and containment sump system can be improved with incorporation of a CR alarm (VC Sump Pump running) or operator actions to increase monitoring of the processing system (i.e. sump flow monitor once every 4 hours).

Leakage detection methods are described in detail and evaluated in Chapter 6.

Leakage Prevention

Reactor Coolant System components were manufactured to exacting specifications which exceeded normal code requirements (as outlined in Section 4.1.3). In addition, because of the welded construction of the Reactor Coolant System and the extensive non-destructive testing to which it was 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 the reactor coolant pump seals. Also, all sealed joints are potential sources of leakage even though the most appropriate sealing device was selected in each case. Thus, because of the large number of joints and the difficulty of assuring 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.

Whereas leakage prevention is important from the standpoint of RCS inventory loss, it can also be of concern with regard to corrosion of carbon steel components within the reactor coolant pressure boundary. Leakage past the canopy seals of the reactor vessel closure head penetrations has been reported at several plants including Indian Point 3. While this type of leakage is not a safety issue due to the minimal amounts possible, it does represent a boric acid corrosion concern. For these reasons, CRDM and CET Seal Clamp Assemblies designed by the reactor manufacturer have been installed at the twelve spare reactor vessel head penetrations and at the five core exit thermocouple penetrations to prevent canopy seal weld leaks in these locations. The CRDM and CET Seal Clamp Assemblies meet all of the ASME Boiler and Pressure Vessel Code requirements applicable to the Indian Point 3 reactor vessel.

Locating Leaks

Experience has shown that hydrostatic testing is successful in locating leaks in a pressure containing system.

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.

Following a significant boration or dilution, the operator normally will sample both the Reactor Coolant System hot legs and the pressurizer liquid for record purposes and to check that homogenization of the pressurizer liquid with the recirculating reactor coolant has been completed. For a cold shutdown, the operator borates the system prior to the start of cooldown.

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 reactor coolant water qualify listed in Table 4.2-2. Maintenance of the water quality to minimize corrosion is accomplished using the Chemical and Volume Control System and the Sampling System which are described in Chapter 9.

4.2.9 Reactor Coolant Flow and Temperature Measurements

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 $\Delta P/\Delta P_o = (w/w_o)^{2.0}$ where ΔP_o is the referenced pressure differential with the corresponding referenced flow rate w_o and ΔP is the pressure differential with the corresponding flow rate w. 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 plants. The expected absolute accuracy of the channel is within 10% and field results have shown the repeatability of the trip point to be within 1%. The analysis of the loss of flow transient presented in Section 14.1.6 provides for an allowance of 3% measurement inaccuracy.

As a result of the calibration techniques used, the absolute accuracy of the coolant flow measurement is not relevant. As indicated in Chapter 14, the limiting trip point assumed for analysis was 87% loop flow. As indicated above, this represents a 3% of flow allowance below the allowable value (low flow trip point) of 90% or higher which is dictated by the Technical Specifications. Since the trip point is calibrated as a function of full flow output of the instrument and since the flow rate of the reactor is verified to be equal or greater than the design flow rate during startup testing, the actual trip point would be 89% or greater based on the 1% repeatability.

Startup tests provide a means for verifying that reactor coolant flow is equal to or greater than the design flow rate. The core flow rate cap be verified with an accuracy better than 10% by correlating a secondary system heat balance and the inlet and outlet core temperatures. In addition measurements of pump input power and loop ΔP can be made at hot shutdown condition for various configurations of running pumps, and the absolute flow rate of each pump is verified to be greater than the design flow.

Hot leg temperature measurement of each loop is accomplished with three fast response, narrow range single element RTD's mounted in the three scoops of each hot leg. Cold leg temperature measurement of each loop is accomplished with one fast response, narrow range duel element RTD mounted in the manifold of each cold leg. In addition a single wide range RTD is provided in each cold leg at the discharge of the RCP and near the entrance to each steam generator.

4.2.10 Reactor Coolant System Vent Collection System

A vent collection system consisting of stainless steel pipes is provided as a means to collect water and contain gases during venting of the Reactor Coolant System during cold shutdown.

The system is located inside the Containment and consists of a vent collecting header which is connected to the Reactor Coolant System venting points. The vent header discharges gases inside the Containment in the area of the purge exhaust system. Liquids are continuously drained to the Reactor Coolant Drain Tank or to the Containment Sump.

The Vent Collection System is used only when the reactor is in cold shutdown; vent connections are disconnected and the Reactor Coolant System venting points flanged off before the reactor is put into hot shutdown mode.

4.2.11 Reactor Head Vent System

The basic function of the Reactor Vessel Head Vent System is to remove non-condensible gases or steam from the Reactor Vessel Head. This system is designed to mitigate a possible condition of inadequate core cooling due to a loss of natural circulation resulting from the accumulation of non-condensible gases in the Reactor Coolant System. The Reactor Head Vent System connects to the reactor head via an existing part length control rod drive mechanism conoseal port. The vent piping downstream of the concoseal port connects to two parallel paths. Each path contains two series normally closed valves. The two flow paths join together and discharge into the pressurizer safety and relief valve discharge header. The common vent line downstream of the valves contains a flanged spool piece which is removable for refueling. These flanged connections are therefore outside the reactor coolant pressure boundary.

The Reactor Head Vent System is operated manually from the main control room. The solenoid operated isolation valves are full open/close type valves which are controlled from individual control switches in the main control room.

4.2.12 Reactor Vessel Level Indication System (RVLIS)

The basic function of the RVLIS is to monitor the water level in the reactor vessel or relative voids in the RCS during accident conditions. However, the system is designed to function under all plant normal, abnormal, and accident conditions, including LOCA, post-LOCA, and during and after a seismic event. Refer to Section 7.5.2 for more detailed information about the operation of this system.

References

(1) Davidson, J. A., S. L. Anderson and W. T. Kaiser. "Analysis of Capsule T from the Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program," WCAP-9491, April, 1979.

(2) Hazeton, W.S., S. L. Anderson and S. E. Yanichko. "Basis for Heatup and Cooldown Limit Curves," WCAP-7924, July, 1972.

(3) Yankchko, S. E. and J. A. Davidson, "Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program," WCAP-8475, January, 1975.

(4) Yanichko, S. E. and Anderson. "Analysis of Capsule Y From the Power Authority of the State of New York Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program, Volume 1," WCAP-10300-1. March, 1983.

(5) Kaiser, W. T. "Analysis of Capsule Y From the Power Authority of the State of New York Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program, Volume 2," WCAP-10300-2, March, 1983: Heatup and Cooldown Curves for Normal Operation.

(6) Yanichko, S. E. "Analysis of Capsule Y From the Power Authority of the State of New York Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program, Volume 3," WCAP-10300-3, March, 1983: J-Integral Testing of IX-WOL Fracture Mechanics Specimens.

(7) Perone, V. A. et al. "Indian Point Unit 3 Reactor Vessel Fluence and RT_{PTS} Evaluations," WCAP-11045, January, 1986 and Revision 1 thereto dated June,1989.

(8) NRC IE Bulletin 79-02, dated March 8, 1979, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts."

(9) Yanichko, S. E. "Analysis of Capsule Z from the New York Power Authority Indian Point 3 Reactor Vessel Radiation Surveillance Program," WCAP-11815, March, 1988.

(10) Entergy Report to NRC "Capsule X Material Surveillance Report, NL-04-092," 07/29/04.

(11) Calculation CN-RCDA-03-88, Rev. 0, "Indian Point Unit 3 Stretch Power Uprate Evaluation for Reactor Vessel Intergrity," Westinghouse Electric Co., December 2003.

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 & Nozzle Forgings	A-508 Class 2
	Cladding, Stainless Weld Rod	type 304 equivalent
	Thermal Shield and Internals	A-240, type 304
	Insulation	Stainless steel, Aluminum
Replacement Steam Generator	Pressure Plate	SA-533, Gr. B Class 1
	Cladding, Stainless Weld Rod	type 304 equivalent
	Cladding for Tube Sheets	Inconel Weld Deposit
	Tubes	Inconel (SB-163)
	Channel Head Forgings	SA-508 Class 3
Pressurizer	Shell	SA-302 Gr. B
	Heads	SA-216 WCC
	External Plate	SA-302, Gr. B
	Cladding, Stainless	type 304 equivalent
	Internal Plate	SA-240 type 304
	Internal Piping	SA-376 type 316
Pressurizer Relief Tank	Shell	A-285 Gr. G
	Heads	A-285 Gr. C
	Internal surface coating	Amercoat 55
Piping	Pipes	A-376 type 316
	Fittings	A-351, CF8M
	Nozzles	A-182 F316
Pump	Shaft	type 304
	Impeller	A-351, CF8
	Casing	A-351, CF8M
Valves	Pressure Containing Parts	A-351, CF8M and A-182 F316

TABLE 4.2-2

REACTOR COOLANT WATER CHEMISTRY SPECIFICATION

Chemistry Parameter	MODE	Requirement
Electrical Conductivity	1, 2, 3, 4, 5	Determined by the concentration of boric acid and alkali present.
Solution pH (at 25 C)	1, 2, 3, 4, 5	Determined by the concentration of boric acid and alkali present. Expected values range between 4.2 (high boric acid concentration) to 10.5 (low boric acid concentration).
Oxygen, ppm, max. (>250 F)	1, 2, 3, 4	0.10
Chloride, ppm, max. (>250 F)	1, 2, 3, 4	0.15
Flouride, ppm, max. (>250 F)	1, 2, 3, 4	0.15
Hydrogen, cc (STP)/kg H_20	1, 2	25-50
Total Suspended Solids, ppm, max.	1, 2, 3, 4, 5	1.0
pH Control Agent (Li^7OH)	1, 2	Varying to maintain minimum RCS $pH_{(t)}$ of 6.9, not to exceed 3.5 ppm Lithium
Boric Acid as ppm Boron	1, 2, 3, 4, 5, 6	Variable from 0 to 4000

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 the design and stress analyses, the material selection and fabrication, and the quality control and operations control.

Reactor Vessel

A stress evaluation of the reactor vessel has been carried out in accordance with the rules of Section III of the ASME Boiler and 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 State of New York has adopted ASME Code Section III and imposes no additional design requirements beyond those listed in this code.

A summary of fatigue usage factors for components of the reactor vessel is given in Table 4.3-2.

For the Indian Point 3 reactor vessel, the maximum thermal stress due to gamma ray heating occurs in the cylindrical portion of the vessel adjacent to the core and its value is about 2500 psi. This additional thermal stress does not augment the stress intensity values considerably. The maximum stress intensity values under steady state and transient operating conditions are still far below the allowable limits of N-414 of ASME Boiler and Pressure Vessel Code Section III. 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 presently in service. These cycles include five heatup and cooldown cycles per year, a conservative selection when the vessel may not complete more than one cycle per year during normal operation. A complete list of thermal and loading cycles and their frequencies used for design purposes are given in Table 4.1-8.

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.

Assurance of adequate fracture toughness of the Reactor Coolant System is provided by compliance, insofar as possible, with the requirements for fracture toughness included in Appendix G to Section III of the ASME Boiler and Pressure Vessel Code.

The heatup and cooldown curves for the plant are based on the actual measured fracture toughness properties of the vessel materials (see the Technical Specifications) determined in accordance with the above mentioned fracture toughness requirements. Where sufficient tests to comply with these requirements for fracture toughness testing were not performed, conservative estimates of fracture toughness properties are used. Maximum allowable pressures as a function of the rate of temperature change and the actual temperature relative to the vessel RT_{NDT} are established according to the methods given in Appendix G, Protection Against Nonductile Failure, published in Section III of the ASME Boiler and Pressure Vessel Code. Curves incorporating allowances for instrument error in measurement of temperature and pressure are given in the Technical Specifications.

These curves are based on the predicted RT_{NDT} of the vessel and include appropriate estimates of ΔRT_{NDT} caused by radiation. Estimated ΔRT_{NDT} values are derived by using Figure 4.4-1 and the fluence at 1/4T corresponding to the maximum for the service period applicable. (See Figure 4.2-10)

The results of the radiation surveillance program are used to verify the ΔRT_{NDT} predicted from Figure 4.4-1 as discussed in Section 4.4.

The use of an RT_{NDT} that includes a ΔRT_{NDT} to account for radiation effects on the core region material automatically provides additional conservatism for the non-irradiated regions. Therefore, the flanges, nozzles, and other regions not affected by radiation will be favored by additional conservatism approximately equal to the assumed ΔRT_{NDT}.

Change in fracture toughness of the core region plate or forging, weld metal and of the associated heat affected zone due to radiation damage are monitored by a surveillance program based on ASTM E-185, Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels. For additional details of the irradiation program, see Section 4.5. In addition, assurance of adequate fracture toughness is further provided by the evaluations required by the NRC's Pressurized Thermal Shock (PTS) Rule (10CDF50.61). Information from the radiation surveillance program referenced above is used to demonstrate that the RT_{PTS} values generated in accordance with PTS rule methodology remain below the NRC screening values for PTS.

The vessel closure contains fifty-four 7-inch studs. The stud material has a minimum yield strength of 104,400 psi at design temperature. The membrane stress in the studs when they are at the steady state operational condition is approximately 54,210 psi. This means that twenty eight of the fifty four 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.

As part of the plant operator training program, supervisory and operating personnel are instructed in reactor vessel design, fabrication and testing, as well as present and future precautions necessary for pressure testing and operating modes. The need for record keeping is stressed as such records are helpful for future summation of time at power levels and temperatures which tend to influence the irradiated properties of the material in the core region.

The following components of the reactor pressure vessel were analyzed in detail through systematic analytical procedures:

Control Rod Housing

An interaction analysis was performed on the CRDM housings. The flange was assumed to be a ring and the tube a long cylinder. The different values of Young's Modulus and coefficients of thermal expansion of the tubes were taken into account in the analysis. The local flexibility was considered at appropriate locations. The closure head was treated as a perforated spherical shell with modified elastic constants. The effects of redundants on the closure head were assumed to be local only. Using the mechanical and thermal stresses from this analysis, a fatigue evaluation was made for the "J" weld.

Closure Head Flange and Shell

The closure head, closure head flange, vessel flange, vessel shell and closure studs were all evaluated in the same analysis. An analytical model was developed by dividing the actual structure into different elements such as long sphere, ring, long cylinder, cantilever beam, etc. An interaction analysis was performed to determine the stresses due to mechanical and thermal loads. These stresses were evaluated in light of the strength and fatigue requirements of the ASME Boiler and Pressure Vessel Code Section III.

Main Closure Studs

A similar analysis to the one described for the closure head flange and shell was performed for the vessel flange to vessel shell juncture and for the main closure studs.

Inlet Nozzle and Vessel Support

For the analysis of nozzle and nozzle to shell juncture, the loads considered were internal pressure, operating transients, thermally induced and seismic pipe reactions, static weight of vessel, earthquake loading, expansion and contraction, etc. A combination of methods was used to evaluate the stresses due to mechanical and thermal loads and external loads resulting from seismic pipe reactions, earthquake, pipe break, etc.

For fatigue evaluation, peak stresses resulting from external loads and thermal transients were determined by concentrating the stresses as calculated by the above described methods. Combining these stresses enabled the fatigue evaluation to be performed.

Outlet Nozzle and Vessel Support

The method of analysis for the outlet nozzle and vessel support was the same as for the inlet nozzle.

Vessel Wall Transition

Vessel wall transition was analyzed by means of a standard interaction analysis. The thermal stresses were determined by the skin stress method where it was assumed that the inside surface of the vessel was at the same temperature as the reactor coolant and that the mean temperature of the shell remained at the steady state temperature. This method was considered conservative.

Core Barrel Support Pads

Thermal, mechanical and pressure stresses were calculated at various locations on the pad and at the vessel wall. Mechanical stresses were calculated by the flexure formula for bending stress in a beam, pressure stresses were taken from the analysis of the vessel to bottom head juncture and thermal stresses were determined by the conservative method of skin stresses. The stresses due to the cyclic loads were multiplied by a stress concentration factor where applicable and used in the fatigue evaluation.

Bottom Head to Shell Juncture

The standard interaction analysis and skin methods were employed to evaluate the stresses due to mechanical and thermal stresses, respectively. The fatigue evaluation was made on a cumulative basis where superposition of all transients was taken into consideration.

Bottom Head to Instrument Penetrations

An interaction analysis was performed by dividing the actual structure into an analytical model composed of different structural elements. The effects of the redundants on the bottom head were assumed to be local only. It was also assumed that, for any condition where there was interference between the tube and the head, no bending at the weld could exist. Using the mechanical and thermal stresses from this analysis, a fatigue evaluation was made for the “J” weld.

The location and geometry of the areas of discontinuity and/or stress concentration are shown in Figures 4.3.-1, 4.3.-2 and 4.3-3.

In addition to the analyses of reactor vessel components, the following transients were analyzed since they cause temperature and pressure excursions influencing the cumulative fatigue of the reactor vessel:

Transient Analyses

Loss of Load Transient. This is the most severe anticipated transient on the Reactor Coolant System. This transient applies to a step decrease in turbine load from full power occasioned by the loss of turbine load without immediately initiating a reactor trip. The reactor and turbine eventually are assumed to trip as a consequence of a high pressurizer level trip initiated by the Reactor Protection System.

Figure 4.3-4 gives the pressure-temperature transient assumed in the analysis for usage factor. This design basis transient is more severe than that reported in Section 14.1.8.

Loss of Flow Transient. This transient applies to a partial loss of flow accident from full power in which a reactor coolant pump is tripped out of service as a result of a loss of power to that pump. The consequences of such an accident are a reactor and turbine trip, followed by automatic opening of the steam dump system and flow reversal in the affected loop. The net result of the flow reversal is a sizeable reduction in the hot leg coolant temperature of the affected loop. Figure 4.3-5 gives the temperature transient assumed in the analysis for usage factor. This design basis transient is more severe than that reported in Section 14.1.6.

The number of occurrences of the loss of load and the loss of flow transients was generally specified at two (2) for each year of plant design life.

Reactor Coolant System Stress Analyses

All components in the Reactor Coolant System were designed to withstand the effects of these and other transients that would result in system temperature and pressure changes.

Stress intensity values at all critical points in the reactor vessel due to these excursions of pressure and temperature were determined for each of these transients through systematic analytical procedures. These stress intensity values S_{ij}(i, j = 1, 2, 3) were plotted against a time interval for each cycle. This plot would represent one or more stress cycles. For each cycle, extreme values of S_{max} and S_{min} were determined. From these values, the largest S_{alt} (alternating stress intensity) was found.

For this value of S_{alt} , an allowable number of cycles (N) was determined through design fatigue curves established for different materials. The ratio of design cycles (n) to allowable cycles (n) gave the usage factor (u). The usage factor was determined in this manner for all transients. The cumulative usage factor was determined by summing the individual usage factors. The cumulative usage factor ($U=u_1 + u_2 + u_3$….) was never allowed to exceed a value of 1.0.

Subsection N415.2 of the 1965 Edition of the ASME Code was used for calculating the usage factors.

Details of thermal and seismic analyses are summarized below[15]. These analyses are directly applicable to Indian Point 3.

Thermal Stresses

Maximum thermal stresses in the core barrel would occur if cold water were injected from the accumulators due to the occurrence of a Loss-of-Coolant Accident. The barrel is exposed to cold water in the downcomer annulus and to somewhat hotter water in the compartments between barrel and baffle, producing a thermal gradient across the barrel wall.

The lower support structure is cooled more uniformly because of the large and numerous flow holes, and consequently, thermal stresses are lower.

The method used to obtain the maximum barrel stresses was as follows:

1) Temperature distribution across the barrel wall was computed as a function of time taking into consideration water temperatures and film coefficients

2) Assuming that the obtained thermal gradients were axisymetrically distributed, which is conservative for stresses, maximum thermal stresses were computed in the barrel which was considered as an infinite cylinder

3) Thermal stresses were added to primary stresses, including seismic, in order to obtain the maximum stress state of the barrel.

Results of the analyses showed that the maximum thermal stresses in the barrel wall were well below the allowable criteria given for design by Section III of the ASME Code.

Seismic Analysis of Reactor Internals

The maximum stresses were obtained by combining the contributions from the horizontal and vertical earthquakes in the most conservative manner. The following paragraphs describe the horizontal and vertical contributions.

Horizontal Earthquake Model and Procedure

The Containment Building along with the reactor vessel support, the reactor vessel, and the reactor internals were included in this analysis. The mathematical model of the building, attached to ground, was identical to that used to evaluate the building structure. The reactor internals were mathematically modeled as beams, concentrated masses, and linear springs.

All masses, water, and metal were included in the mathematical model. All beam elements had the component weight or mass distributed uniformly, e.g., the fuel assembly mass and barrel mass. Additionally, wherever components were attached somewhat uniformly, their mass was included as an additional uniform mass, e.g., baffles and formers acting on the core barrel. The water near and about the beam elements was included as a distributed mass.

Horizontal components were considered as concentrated masses acting on the barrel. These concentrated masses also included components attached to the horizontal members since these were the media through which the reaction was transmitted. The water near and about these separated components was considered as being an additive at these concentrated mass points.

The concentrated masses attached to the barrel represented the following: a) the upper core support structure, including the upper vessel head and one-half the upper internals; b) the upper core plate, including one-half the thermal shield and the other half of the upper internals: c) the lower core plate, including one-half of the lower core support columns; d) the lower one-half of the thermal shield; and e) the lower core support, including the lower instrumentation and the remaining half of the lower core support columns.

The modulus of elasticity was chosen at its hot value for the three major materials found in the vessel, internals, and fuel assemblies. In considering shear deformation, the appropriate cross-sectional areas were selected along with a value for Poisson's ratio. The fuel assembly moment of inertia was derived from experimental results by static and dynamic tests performed on fuel assembly models. These tests provided stiffness values for use in this analysis.

The fuel assemblies were assumed to act together and were represented by a single beam. The following assumptions were made with regard to connection restraints. The vessel is pinned to the vessel support which is the surrounding concrete structure and part of the Containment Building. The barrel is clamped to the vessel at the barrel flange and spring connected to the vessel at the lower core barrel radial support. This spring corresponds to the radial support stiffness for two opposite supports acting together. The beam representing the fuel assemblies is pinned to the barrel at the locations of the upper and lower core plates.

Model analysis plus the response spectrum method[16] were used in this analysis. The modal analysis was studied by the use of a transfer matrix method.

The maximum deflection, acceleration, etc., were determined at each particular point by summing the absolute values obtained for modes. With the shear forces and bending moments determined, the earthquake stresses were then calculated.

Figure 4.3-6 shows the mathematical model studied.

Analytical Model for Vertical Earthquake Model and Procedure

The reactor internals were modeled as a single degree of freedom system for vertical earthquake analysis. The maximum acceleration at the vessel support was increased by the amplification due to the building-soil interaction.

There were no interfaces in the analysis (e.g., dynamic to static, elastic to plastic).

Reactor Coolant Pumps Evaluation

Pump Motor Overspeed

During normal operation, the reactor coolant pumps are supplied from the unit auxiliary bus and therefore are tied to the turbine generator frequency (speed). On occurrence of unit (turbine) trip, the pump electrical buses are transferred from the auxiliary transformer without any intentional delay.

On most electrical and mechanical events which cause the turbine to be tripped, the reactor coolant pump buses and the unit are tripped simultaneously and the pumps will therefore not exceed their normal or pretrip running speed. If for some unlikely reason the only plant trip is a turbine overspeed trip (mechanical – hydraulic trip), then the pump trip will be initiated by the turbine hydraulic system and the trip point will be between 106 and 110 percent of the turbine generator synchronous speed. The turbine overspeed trip point is set at a value not to exceed 106 percent of synchronous speed (1908 rpm).

Missile Prevention

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. Low oil levels would signal an alarm in the Control Room and require shutting down the pump. Each motor bearing contains embedded temperature detectors, and so initiation of failure, separate from loss of oil, would be indicated and alarmed in the Control Room as a high bearing temperature. This, again, would require 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 constitutes a loss of coolant flow in the one loop, the effect of which is analyzed in Chapter 14.

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 into 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. Indication of pump malfunction under these conditions would be initially by high temperature signals from the bearing water temperature detector and later by excessive No. 1 seal leakoff indications. Following these signals, pump vibration levels would be checked. These would show excessive levels, indicating some mechanical trouble. The pump would then be shut down for investigation.

The design specifications for the reactor coolant pumps included as a design condition the stresses generated by a design basis earthquake ground acceleration of 0.15g. 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 would any bearing stress in the pump or motor exceed or even approach a value which the bearing could not carry.

Precautionary measures, taken to preclude missile formation from primary coolant pump components, assure that the pumps will not produce missiles under any anticipated accident condition. Each component of the primary pump motors has been analyzed for missile generation. Any fragments of the motor rotor would be contained by the heavy stator. The same conclusion applies to the pump impeller because the small fragments that might be ejected would be contained by the heavy casing.

<u>Reactor Coolant Pump (RCP) Flywheel</u>

a) Methods of Fabrication

The pump flywheels were fabricated from vacuum degassed steel accordance with ASTM A-533 CL-1, Grade B.

The flywheel blanks were flame-cut from 8 inch and 5 inch plates with allowance for exclusion of flame-affected metal. They were then machined to the specific dimensions, and the bolt holes were drilled. These plates were subjected to 100 percent volumetric ultrasonic examination.

The two plates, machined to 7.25 inches and 4.45 inches in thickness, were then bolted together, the finished flywheel was attached to the motor shaft, and the whole unit was balanced to yield vibration levels at operating speed less than 0.001 inches double amplitude.

The supplier certification data, presented in Table 4.3-5, shows the Charpy V-notch test results for the Indian Point 3 flywheels at +10° F.

Acceptability of the above flywheel material for Indian Point 3, in comparison to Safety Guide No. 14 toughness criteria, was determined by the following two steps:

1) A reference curve describing the lower bound fracture toughness behavior for the material in question was established

2) Charpy (C_V) impact energy values obtained in certification tests (Table 4.3-5) at 10° F were used to fix the position of the heat in question on the reference curve.

A lower bound K_{Id} reference curve (see Figure 4.3-8) was constructed from dynamic fracture toughness data generated by Westinghouse on A-533 Grade B, Class I steel. (13)(14) All data points were plotted on the temperature scale relative to the NDT temperature. The construction of the lower bound, below which no single test point falls, combined with the use of dynamic data when flywheel loading is essentially static represents a large degree of conservatism.

The applicability of a 30 ft-lb Charpy energy reference value was derived from sections on Special Mechanical Property Requirements and Tests in Article 3, Section III of the ASME Boiler and Pressure Vessel Code. The implication was that the test temperature lies a safe margin above NDT. Indian Point 3 flywheel plates exhibited an average value greater than 30 ft-lbs in the weak direction and, therefore, met the specific requirement “a” stated in Safety Guide No. 14 that NDT must be no higher than 10° F. Making the conservative assumption that all materials in compliance with the Code requirements were characterized by an NDT temperature of 10° F, one was able to reassign the “zero” reference temperature position in Figure 4.3-8 a value of 10° F.

Flywheel operating temperature at the surface is 120° F. The lower bound toughness curve indicated a value of 116 ksi-in^{½} at the (NDT + 110) position corresponding to operating temperature. Safety Guide No. 14 requirement “c” was fulfilled with considerable margin for safety. The flywheel analysis was reviewed by Westinghouse (Reference 20) and it was concluded that a 10°F increase in flywheel surface temperature to 130° F has no adverse effect on the flywheel integrity or the conclusions of the analysis.

By assuming a minimum toughness at operating temperature in excess of 100 ksi-in^{½}, it was seen by examination of the correlation in Figure 4.3-9 that the C_V upper shelf energy must be in excess of 50 fl-lb; therefore, the requirement “b” that the upper shelf energy must be at least 50 ft-lb was satisfied.

Based on the above discussion, the flywheel materials met the Safety Guide No. 14 toughness criteria on the basis of supplier certification data.

The design requirements of the bearings were 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 were held at a very low value, and even under the most severe seismic transients or other accidents, would not begin to approach loads which cannot be adequately carried for short periods of time.

Because there were no established criteria for short-time stress-related failures in such bearings, it was 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, gave assurance of the adequacy of the bearings to operate without failure.

As was generally the case with machines of this size, the shaft dimensions were predicated on avoidance of shaft critical speed conditions rather than actual levels of stress.

There are many machines as large as and larger than the reactor coolant pumps designed to run at speeds in excess of first shaft critical. However, it was considered desirable in a superior product to operate below first critical speed, and the Reactor Coolant Pumps were designed in accordance with this philosophy. This resulted in a shaft design which, even under the severest postulated transient, gave 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 gave assurance of the conservative stress levels experienced during these transients.

So, in each of these cases, where the functional requirements of the component controlled its dimensions, it was seen that if the requirements discussed previously were met, the stress-related failure cases were more than adequately satisfied.

It was 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 flywheel.

b) Methods of Quality Assurance

An NDTT less than +10 F was specified. A minimum of three Charpy tests were made from each plate, parallel and normal to the rolling direction, to determine that each blank satisfied design requirements.

The finished flywheels were subjected to 100% volumetric ultrasonic inspection.

The finished machined bores were also subjected to magnetic particle, or liquid penetrant examination.

The design-fabrication techniques yielded flywheels with primary stress at operating speed (as shown in Figure 4.3-7) which was less than 50% of the minimum specified material yield strength at room temperature (100 to 150 F).

The flywheel calculated stresses at operating speed were based on stresses due to centrifugal forces (Figure 4.3-7). The stress resulting from the interference fit of the flywheel on the shaft was less than 2000 psi at zero speed, but this stress became zero at approximately 600 rpm because of radial expansion of the hub.

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% compared with maximum expected overspeed of 109%
2) A crack through the thickness of the flywheel at the bore
3) 400 cycles of startup operation in 40 years of plant design life.

Using critical stress intensity factors and crack growth data attained on flywheel material, the critical crack size for failure was greater than 17 inches radially and the crack growth data was 0.030" to 0.060" per 1000 cycles.

An ultrasonic inspection capable of detecting at least 1/2" deep cracks from the ends of the flywheel and a dye penetrant or magnetic particle test of the bore will be performed when the RCP motor is sent out for refurbishment in accordance with the Preventive Maintenance Program.

c) Normal Operating Speed

The primary coolant pumps run at 1189 rpm and may operate briefly at overspeeds up to 109% (1295 rpm) during loss of outside load.

For conservatism, however, 125% 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 seconds, pump operating temperatures would remain at about the design value.

d) Bursting Speed

Bursting speed of the flywheels has been calculated on the basis of Robinson's results(1) to be 3900 rpm, more than three times the operating speed. This is confirmed using Griffith-Irwin theory.(2)

IP3 License Amendment 221, issued July 2, 2004 revised the plant technical specifications to extend RCP flywheel inspection interval from 10 to 20 years. The amendment was based on NRC approval (May 5, 2003) of WCAP-15666 developed and submitted through the Westinghouse Owners Group. The technical justification provided in WCAP-15666 is based on a combination of deterministic fracture mechanics methods and risk based considerations of credible maximum flywheel speeds under specified operational or accident conditions.

<u>Steam Generators</u>

The pressure boundary integrity of the steam generator, including the primary to secondary pressure boundary is assured by compliance of the design, fabrication, analysis, inspection, and testing activities with the criteria and requirements of the ASME Boiler and Pressure Vessel Code. The stress report for the Model 44F steam generators currently installed in Indian Point Unit 3 included an evaluation for faulted conditions including large break LOCA and steam line break (loss of secondary pressure). The stress intensities calculated for these conditions are less than the applicable limits from the ASME Code. The criteria and requirements of Section III of the ASME Code (1965 edition, Winter 1965 Addenda) were used for the evaluation.

The evaluation of the stress intensity levels in the tubesheet and channel head for these faulted conditions is based on an evaluation of interactions of the complex structure of the channel head, tubesheet and lower shell. A finite element computer program is used for the evaluation. The structure is modeled in terms of discrete elements with loading and boundary conditions applied to these elements. The system of simultaneous linear equations resulting from the modeling is solved to determine the stress conditions. This method of stress analysis is well established for reactor coolant system components. For the tubesheet calculations, the guidelines of Article I-9, ASME Code, Section III were used to calculate the ligament efficiency based on nominal pitch dimension and maximum hold dimensions.

The postulated rupture of a primary pipe is assumed to impose a maximum pressure differential of 1100 psi across the tubes and tubesheet. The maximum local primary membrane plus primary bending stress in the tubesheet under these conditions is 21,620 psi. This is well below the ASME Code stress intensity allowable of 84,000 for this condition. The stress intensities in the channel head, channel head to tubesheet weld, and the tubesheet to lower shell weld have lower values and are well below applicable limits for this condition.

The postulated rupture of a secondary pipe is assumed to impose a maximum pressure differential of 2485 psi across the tubes and tubesheet. The maximum local primary membrane plus primary bending stress in the tubesheet under these conditions is 54,650 psi. This is well below the ASME Code stress intensity allowable of 84,000 for this condition (Tables 4.3-3 and 4.3-4). The stress intensities in the channel head, channel head to tubesheet weld, and the tubesheet to lower shell weld have lower values and are well below applicable limits for this condition.

The stress intensity in the tubes has also been considered for the postulated faulted conditions. In addition to the pressure differential stresses, LOCA blowdown forces result in a bending load on the tubes. The fatigue due to vibration of the tubes during a steam line break does not need to be considered in the evaluation. The requirements of the ASME Code are met for these faulted conditions.

In addition to analytical results, results of destructive pressure testing or representative tubes demonstrates a factor of safety against tube collapse due to external pressure. The results of the pressure testing have been used to calculate a collapse pressure for tubes of the size and material in the Model 44F steam generators. The lower bound collapse pressure for the tubes in the Indian Point steam generators is 2369 psi considering tube ovality, tube wear, and tube corrosion.

The structural evaluation of the tubes uses an allowance of 2 mils of uniform wall thinning. This value is based on published values and operating experience for corrosion and erosion-corrosion. The plugging limit for indication of tube degradation is based on the requirements of IWB-3521 of the ASME Boiler and Pressure Vessel Code Section XI or an analysis meeting the requirements of Regulatory Guide 1.121 and not on the allowance for uniform thinning in the structural evaluation.

The loading conditions considered include the maximum potential earthquake loading conditions superimposed on the loss of secondary pressure effects. The dynamic effects of the fluid and the acceleration of the steam generator result in a small increase in the equivalent pressure loading compared to the base pressure differential. The stress intensity for the combined loading condition is well below limits.

The fluid dynamic load on the tube support plate for the steam line break conditions has been considered. The analysis has determined that the tube supports will be restrained without deformation of the tubes.

In addition, the secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator is below 70 ^{o}F.

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 the Residual Heat Removal Systems. The Reactor Coolant System is dependent upon the steam generators and upon the steam, feedwater, and condensate system for decay heat removal under normal operating conditions until RHR can be placed in service (reactor coolant temperature between 250° F and 350° F). The layout of the system ensures the natural circulation capability to permit adequate core cooling following a loss of all main reactor coolant pumps.

The flow diagram of the Steam and Power Conversion System is shown on Figure 10.2-1. 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 Axiliary Feedwater System will supply water to the steam generators in the event that the main feedwater pumps are inoperative.

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, which reflected consideration of fall design loadings detailed in the design specification, was prepared by the manufacturer. The analysis showed that the reactor vessel, steam generators, reactor coolant pump casings and pressurizer complied with the stress limits of Section III of the ASME Code. A similar analysis of the piping showed that it complied with the stress limits of the applicable USAS Code.

As part of the design control on materials, Charpy V-notch toughness tests were run on all ferritic material used in fabricating pressure parts of the reactor vessel, steam generators and pressurizer to provide assurance that hydrotesting and operation will be in the ductile region at all times. In addition, dropweight 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. Replacement steam generators were shop tested on the primary side at 3107 psig, and hydrotested after installation in accordance with ASME Section XI requirements.

A summary of Charpy V-notch and dropweight test results for the reactor vessel plates and forgings is given in Section 4.4.

Furnace Sensitized Components

The following pressure or strength-bearing stainless steel component parts in the Reactor Coolant System have become furnace-sensitized during the fabrication sequence:

Reactor Vessel
Eight primary nozzle safe ends (forgings) which were overlaid in the field with stainless steel weld metal

Steam Generators
Two primary nozzle safe ends per generator – weld metal buttering

Pressurizer
All nozzle safe ends (forgings) in top and bottom heads.

Vibration and Cycle Loads

Vibration loads were considered in the design for the reactor internals, the steam generator tube bundles, and the reactor coolant pipe. Reactor coolant pump vibration is insignificant. Instrumentation is provided to check the vibration level of these pumps if an abnormal condition is suspected.

Reactor Internals

Model tests of the Indian Point 3 reactor internals were performed for normal operating and transient conditions. Results of the combined analytic and experimental work were factored into the design.

Predicted stresses and deflections were in agreement with tests on reactors having similar internals design. The results of the vibration tests performed on the Ginna rector (reported in WCAP-7408-L, Westinghouse proprietary report) [18] confirmed that the tests agree very closely with the predicted performance and margins. A more extensive testing program was performed during pre-operational testing for Indian Point 2.

Allowable stress amplitude for flow induced vibration was established on the basis of the material fatigue properties (endurance limit of 20,000 psi for 10^{10} cycles). Since infinite cycle fatigue was a criterion, no limits were then necessary for frequency. Displacement amplitudes for reactor internals vibration were not governing; stress limits were more restrictive.

An analysis of the dynamic response of the Indian Point 2 internals under seismic and blowdown loads was made. Allowable criteria were established and stresses and deflections were determined to assure that seismic and blowdown loads will not prevent core shutdown or will not interfere with the effectiveness of the emergency core cooling system (reported in detail in WCAP-7822, Westinghouse non-proprietary report).[15] This analysis applies directly to Indian Point 3.

Steam Generators

a) Tube Vibration Analysis

In the design of Westinghouse Model 44F steam generators used in Indian Point 3, the possibility of tube degradation due to either mechanical or flow-induced excitation was considered. This evaluation included detailed analysis of the tube support systems as well as an extensive research program with tube vibration model tests.

Consideration was given to potential sources of tube excitation including primary fluid flow within the U-tubes, mechanically-induced vibration, and secondary fluid flow on the outside of the tubes. The effects of primary fluid flow and mechanically-induced vibration are considered to be negligible during normal operation. The primary source of potential tube degradation due to vibration is the hydrodynamic excitation by the secondary fluid on the outside of the tubes, and this area has been emphasized in both analyses and tests, including evaluation of steam generator operating experience.

Three potential tube vibration mechanisms due to hydrodynamic excitation by the secondary fluid on the outside of the tubes have been identified and evaluated. These include potential flow-induced vibrations resulting from vortex shedding, turbulence, and fluidelastic vibration mechanisms.

Vortex shedding is possible, at most, only for the outer few rows in the wrapper inlet region of steam generators such as the Model 44F for which non-uniform, two-phase turbulent flow exists throughout most of the tube bundle. 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 wrapper inlet region. Bounding calculations consistent with laboratory tests parameters confirmed that vibration amplitudes would be less than .001 inches, 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, and these vibrations cause stresses which are two orders of magnitude below fatigue limits for the tubing material. Therefore, neither unacceptable tube wear nor fatigue degradation is anticipated due to secondary flow turbulence in the Model 44F design configuration.

Fluidelastic tube vibration is potentially more severe than either vortex shedding or turbulence because it is a self-excited mechanism: relatively large tube amplitudes can feedback proportional driving forces if an instability threshold is exceeded. Tube support spacing incorporated into design of 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. This approach provides large margins against initiation of fluidelastic vibration for tubes which are effectively supported by the Model 44F tube support configuration.

Small clearances between the tubes and 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 the location. Fluidelastic tube response within available support clearance is therefore theoretically possible if secondary flow conditions exceed the instability threshold assuming no support 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 two or more sequential 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 which are not expected. Corresponding tube bending stresses remain more than two orders of magnitude below fatigue limits as a consequence of vibration amplitudes constrained by available clearances. These analyses and tests for limiting postulated fit-up conditions include simultaneous contributions from flow turbulence.

Analyses and tests therefore demonstrate that unacceptable tube degradation resulting from tube vibration is not expected for the Model 44F steam generators at Indian Point 3. Operating experience with similar steam generators supports this conclusion.

b) Tubesheet Analysis

The evaluation of the Indian Point Unit 3 steam generator tubesheets was performed according to the rules of the ASME Boiler and Pressure Vessel Code. The design criteria encompassed consideration of both steady state, transient and emergency operations specified in the steam generator Design Specification for the plant.

The evaluation of the tubesheet involved the heat conduction and stress analysis of the tubesheet, channel head, and secondary shell structure for particular steady design conditions for which the Code stress limitations were to be satisfied, and for discrete points during transient operation for which the temperature/pressure conditions were to be known in order to evaluate stress minima and maxima for fatigue life usage.

The stress analysis of the tubesheet complex consisted of performing an interaction analysis between the tubesheet and the channel head and attached lower shell to determine the interaction forces and moments between parts of the structure. A finite element computer program is used to calculate the stress intensities. The tubesheet calculations were made with a ligament efficiency determined using guidelines from Article I-9 from the ASME Code, Section III (1965) Edition, Winter 1965 Addenda). The stress analysis considered stress due to symmetric temperature and pressure differential as well as asymmetric temperature distribution due to temperature drop across the tubesheet divider lane.

The fatigue analysis of the tubesheet was performed at potentially critical regions, such as the junction between the tubesheet and the channel head and secondary shell, as well as representative locations in the perforated region of the tubesheet. The nominal stress results from the primary chamber components interaction analysis are used as the basis for the fatigue evaluation. These stresses are modified by stress concentration factors which are a function of position around the circumference of the hole. The fatigue evaluation is performed at the hole evaluated for the nominal location. The Model 44F steam generators for Indian Point Unit 3 did not have any misdrilled holes to be evaluated. Fatigue usage is evaluated for several locations around the circumference of the hole.

In all cases evaluated, the Indian Point Unit 3 steam generator tubesheet complex met the stress limitations and fatigue criteria.

c) Tube-Tubesheet Analysis

The tubes in the Indian Point Unit 3 Model 44F steam generators are hydraulically expanded the full depth of the tubesheet. This expansion of the tubes into the tubesheet adds to the strength of the tubesheet for some loading conditions. However, the effect of the tubes is not accounted for in the analysis.

Due to the expansion of the tube into the tubesheet, the effects of fluid induced vibration of the tube terminate at or near the secondary surface of the tubesheet. The fatigue usage is calculated for the tube at the tube to tubesheet juncture at the secondary surface of the tubesheet. An appropriate fatigue strength reduction factor is used at that location. The calculated fatigue usage for the tube to tubesheet juncture on the secondary side is less than the Code limit.

d) Tube to Tubesheet Weld

At the primary surface of the tubesheet the tube is joined to the tubesheet with a weld of the tube to the tubesheet cladding. The weld is not subject to any effects of tube vibration on the secondary side. The stress analysis of the weld considers thermal stresses due to increasing and decreasing transient temperature conditions. A finite element model was used to determine the states of stress in the tube to tubesheet weld. The model included the effect of adjacent tube holes o n the stiffness of the tubesheet. The stress state of the tubesheet surrounding the tube was based on the tubesheet analysis. The weld was evaluated for fatigue usage using an appropriate fatigue strength reduction factor. The calculated fatigue usage for the weld is less than the Code limit.

Reactor Coolant Pumps

The RCP motors are equipped with a Bently-Nevada vibration monitoring system that provides continuous monitoring of both shaft and motor frame vibration. Continuous Control Room recording can be provided by this system via selector switches and chart recorders for trending and troubleshooting RCP vibration. Additionally, vibration alarms are provided for the RCP's. A Control Room alarm will be annunicated if an RCP vibration exceeds the predetermined level. A noise detector is also located adjacent to each motor and may be placed in service via a selector switch in the control room.

Each of the reactor coolant pumps is also equipped with two International Research and Development vibration pickups mounted at the top of the motor support stand. They are mounted in a horizontal plane and pick up radial vibrations of the pump. One is aligned parallel to the pump discharge; the other is aligned perpendicular to the pump discharge. Their signals are taken to a multi-point selector switch mounted outside the reactor containment. Both signals form each reactor coolant pump are taken to this selector switch. Also supplied is a vibration meter. This is a portable device that may be plugged into the selector switch, so the signal from any one pickup may be monitored at one time. The vibration levels of the reactor coolant pumps can thus be checked and the pumps were checked during full flow tests for flow induced vibration (Reference 19).

No analysis for vibration loading was performed for the reactor coolant pumps.

<u>Primary System Piping</u>

The reactor Coolant System piping was designed for normal and faulted conditions. For faulted conditions, the piping was 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 vibration from the reactor coolant pump impeller or the pistons of the charging pump.

The perturbating frequency of the reactor coolant pump is quite high when compared to the piping natural frequency. Frequency separation, therefore, insures a very small probability for self-excited or sympathetic vibration. This is borne out of satisfactory operation of several representative coolant loops.

4.3.4 <u>Overpressure Protection</u>

An overpressure Protection System (OPS) is required to prevent the reactor vessel pressure from exceeding the Technical Specifications (Appendix G) and ASME Code Case N-514 limits. The problem arises inasmuch as the reactor vessel steel has less ductility at low temperatures. As the reactor vessel is irradiated during its lifetime, the limitations become even more stringent.

The OPS is a three-channel analog curve tracking arrangement which can initiate an appropriate chain of coincidence logic for the purpose of automatically preventing a violation of the technical specification temperature/pressure limit curve for the reactor vessel.

Wide range RCS temperature signals will be used 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 curve which is the system pressure limit that must be adhered to. The basis of the OPS curve is the Appendix G isothermal cooldown curve increased by 10% (as allowed by ASME Code Case N-514) and then adjusted to allow for effects of the design basis heat input and mass input events.

   The arming function of the Overpressure Protection System is activated when the RCS temperature is below a predetermined value as specified in the Technical Specifications. Below this temperature, one half of a two-out-of-two (temperature-pressure) coincidence logic is satisfied to allow the air operated valves (PCV-455C, 456) to open automatically in the event of an impending overpressure condition. This automatic opening of the valves is initiated by the other half of the two-out-of-two coincidence logic which is described below.

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 RCS pressures allowed at those temperatures. The difference between the Appendix G/ASME Code Case N-514 curve pressures and the actual RCS pressure transmitted by three 0 to 1500 psig transmitters 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. The power operated relief valves are normally operated using the N2 system with accumulators at each valve.

However, the Overpressure Protection System requires an N2 supply of sufficient capacity which, in case of loss of the main N2 supply, can support the number of PORV cycles resulting from an overpressure event of 10 minute duration. This N2 supply is provided from one Safety Injection Accumulator having its associated N2 fill valve blocked open.

The system also includes appropriate accessory equipment to provide adequate testability, calibration facilities and operator surveillance instrumentation.

The main function of the power operated relief valves (PORVs) is to open automatically at 2335 psig to protect against pressure surges. Under the OPS, the PORVs will also open automatically when the reactor coolant temperature is within the temperature range specified in the Technical Specifications, to relieve the RCS from exceeding the Appendix G/ASME Code Case N-514 curves. Under this mode of operation, the PORVs will relieve solid water rather than steam-water mixture at 2335 psi.

Westinghouse conducted a generic reactor vessel overpressure study and the analytical results showed that only one PORV is necessary to turn around the design mass input overpressure incident and the design thermal input overpressure incident. The motor operated valves and power operated relief valves feed to the pressurizer relief tank. The tank has sufficient capacity to accept the expected short term flow from the OPS.

The electrical activation uses two-out-of-three logic for valve activation. The electrical supply is from the instrument buses with backup by the station emergency batteries. Thus, the system is single failure proof in addition to having a secured power supply.

The system allows for accurate control of system heatup and cooldown. Spurious opening and/or closing of the PORVs is essentially eliminated by the new two-out-of-three logic (if one channel were to fail the valve would not be able to malfunction).

4.3.5 System Incident Potential

The potential of the Reactor Coolant System as a cause of accidents was evaluated by investigating the consequences of certain credible types of components and control failures as discussed in Section 14.1 and 14.2. Reactor coolant pipe rupture is evaluated in Section 14.3.

4.3.6 Loose Parts Monitoring

The metal impact monitor was designed to enable early detection of loose metallic parts which may be in the steam generator or the reactor vessel. Upon the occurrence of an impact of loose metallic parts, a pressure wave is generated in the reactor system component causing minute displacements in the component material. The step excitation of the impact produces a broadband frequency response with peak amplitude response at resonant frequencies. Many of these resonant frequencies lie in the audible frequency range and are called “Bell” frequencies. Certain of these “Bell” frequencies are especially sensitive to impact excitation due to differing modes of preferred vibration response.

The displacements attendant with an impact wave are insignificant. However, the accelerations caused by the moderately high audio frequencies are very significant; for this reason, acceleration is the parameter chosen to indicate impact.

Acceleration Measurement

Acceleration is measured by the use of special transducers that convert accelerations to electrical signals. These transducers are mounted at specially selected monitoring points on the exterior of the Reactor Coolant System. Monitoring points normally in use during plant operation are at the top and bottom of the reactor vessel and above and below each steam generator tube sheet with transducers mounted on the generator shell. Additional monitoring points are available above and below each steam generator transition cone and above each feedwater nozzle.

Each transducer contains a piezoceramic material fabricated in a manner that provides for changes in compression of the material in response to accelerations. A small electric charge proportional to acceleration is generated by piezoelectric behavior. The charge is converted to a voltage signal by a charge preamplifier that operates in a feedback manner to continuously compensate for charge input changes to maintain a near-zero differential input voltage. A small current flows in the cable connecting the transducer to the charge pre-amplifier resulting in virtual elimination of cable effects on the acceleration signal. The charge preamplifier output voltage can then be treated as a normal instrument signal requiring only normal shielding and cable considerations.

The sensitivity of this measurement to impact is determined by the impact energy (determined by mass and impact velocity) and the distance from the point of impact to the measurement point (damping and geometry changes attenuate the traveling wave).

Impact Monitoring

The voltage signal from the charge preamplifier is further amplified and then fed to a monitoring device (Digital Metal Impact Monitor). This device excludes all frequency information except for the “Bell” frequencies from the impact. The selected signals are further processed to provide an output DC signal proportional to the impact energy (as seen by the transducer) and a DC signal indicating the rate of occurrence of impact repetition.

Records and Displays

Every 24 hours and whenever the alarm setpoint is exceeded the impact energy level and rate of occurrence can be displayed on a printer. These records serve to establish a history for establishing when and where impacts were observed. The rate at which impacts occur gives an indication of the amount of debris present in the monitored area while the impact energy is a measure of the weight of the debris.

A test circuit is provided for determining the continuity of the monitoring system. The test signal inputs a representative wave form similar to a metallic impact.

Cabling

The signal cables are an integral part of the accelerometer. They are constructed of high temperature and radiation resistant materials and transmit the accelerometer signal to the charge preamplifier.

4.3.7 Cold Shutdown RCS Level Indication

The original system used to monitor RCS level indication during cold shutdown conditions was replaced and upgraded with a new system in accordance with NRC Generic Letters 87-12 – “Loss of RHR While RCS Is Partially Filled” and 88-17 – “Loss of Decay heat Removal.” The new system provides two independent RCS water level indicators and the Westinghouse Ultrasonic Level Measuring System for use during reduced inventory conditions. Filling the RCS while under vacuum was a process introduced in RO11. In order to maintain compliance with Generic Letter 88-17, another RCS Level Monitoring system (Mansell Level Monitoring System, i.e. MLMS) was introduced during RO11 which was capable of providing RCS level indication while in reduced inventory and mid-loop conditions while the RCS is under vacuum as well as non-vacuum conditions. Hand held UT is required to be used with MLMS in mid-loop condition to increase the range of level indication in the hot leg. The four system includes:

1) Two redesigned, independent water level columns, with the supply and vent lines for the level gauges being constructed of stainless steel piping for strength and compatibility with the REC. The level gauges are constructed of 2 inch diameter transparent rigid polyvinyl chloride (PVC) tube and span a range of the RCS from the 78 foot elevation (=10% in the pressurizer) down to the 60.8 foot elevation (inside bottom of hot leg piping), well below the RCS “midloop" elevation of 62’ – 0”. The columns are independently supplied from the low point drain lines of the RCS intermediate loop piping, one level column from loop #32 and the other from loop #34. The level gauges can be viewed in the Control Room by remotely operated cameras.

2) A Westinghouse Ultrasonic Level measuring System (ULMS) that provides control room indications of water level in the reactor coolant piping during non-power operations. The ULMS consists of the following three basic components.

   a) The sensor, which transmits and receives the ultrasonic waves,

b) The preamp/pulsar module, which drives the sensor and preamplifies the received echo signal, and

c) The signal processing module, which provides an analog output signal proportional to the ultrasonic wave travel time in the water.

3) A MLMS that is capable of providing the Control Room indications of water level in the reactor coolant piping during non-power operation. The MLMS utilizes sensing locations above and below where the RCS water levels are expected to remain during outages. The MLMS consists of two independent indication loops, a wide range and a narrow range loop. The wide range loop has an indication range between 121 foot elevation down to the 62 foot elevation. The wide range high point connects to a vent on the pressurizer relief line. The wide range loop low point connects to a drain off of PT-413. The narrow range loop has an indication range between 77 foot elevation down to 62 foot elevation. The narrow range high point connects to a drain off of PT-433. The pressure transducer assembly (consisting of two redundant transducers per assembly) is connected to each of the four sensing locations. The pressure transducers transmit signal to two computers located in the Control Room, i.e. one narrow range and one wide range computer. The pressure difference between the high and low points is calculated by the computers and is displayed as water level on computer screens, as well as LED displays in the Control Room. The computers have an alarm feature.

4) Two hand held UT devices capable of providing indication of water level inside the reactor coolant piping during non-power operations. The UT devices are manually positioned at the 6:00 position of the RCS hot legs and level in the pipe is read locally and communicated back to the Control Room. This method of level indication is only used while the RCS is in mid-loop and may be used to comply with Generic Letter 88-17, but must be used in conjunction with the MLMS system in mid-loop operation.

In order to comply with the requirements of Generic Letter 88-17, two independent continuous RCS water level indications are necessary when the RCS is in a reduced inventory condition. Water level indication should be periodically checked and recorded by an operator or automatically and continuously monitored and alarmed. The ULMS and MLMS system provide an alarm function. The water level columns and UT devices do not. Any combination of two independent indicators may be used for level indication provided they are capable of performing within the RCS pressure (vacuum) ranges. Examples include: one level column and ULMS; one MLMS and one ULMS; two MLMS and two UT devices; two UT devices, etc. Operations procedures delineate specific actions required for the particular combination of level indicators in service.

4.3.8 <u>Saturation Alarm and Recorder (Analog System)</u>

The saturation recorder provides subcooling trend data to the operator which will note if saturation conditions in the Reactor Coolant System occur.

The system involves a calculating device to develop a saturation curve, an alarm duplex bistable (alarm in CCR disabled), an isolating amplifier, a high selector, a low selector and a summing unit to form a difference signal between reactor coolant temperature and saturation temperature.

The RCS temperature is determined by the use of four RCS temperature loops taken from all four hot leg instrument loops. The four signals from these loops are channeled into a device which selects the highest temperature signal and feeds it to a duplex bistable and a summing unit.

The programmed RCS saturation temperature is determined by the use of two RCS pressure loops. The two signals from these loops are channeled into a device which selects the lowest pressure signal and feeds it to a characterizer where a corresponding saturation temperature signal is derived. This temperature signal is fed into the duplex bistable and summing unit. The duplex bistable trips when the difference in temperature (RCS vs. saturation) approaches saturation conditions and when the difference in temperature is at saturation. These alarm functions are disabled from alarming in the CCR.

This system is operational during the natural circulation cooldown mode because the temperature readings in this mode are derived from the RCS loops.

References

1) Robinson, Ernest L., "Bursting Tests of Steam-Turbine Disk Wheels," Transactions of the ASME, July 1944.

2) Winne, D. H. and B. M. Wundt, "Application of the Griffith-Irwin Theory of Crack Propagation to the Bursting Behavior of Discs, Including Analytical and Experimental Studies," Transactions of ASME, Paper No. 57-A-249, December 1, 1957.

3) DELETED 6/2000

4) DELETED 6/2000

5) DELETED 6/2000

6) DELETED 6/2000

7) DELETED 6/2000

8) DELETED 6/2000

9) DELETED 6/2000

10) DELETED 6/2000

11) DELETED 6/2000

12) DELETED 6/2000

13) Shabbits, W.O., "Dynamic Fracture Toughness Properties of Heavy Section A-533 Grade B Class 1 Steel Plate," WCAP-7623.

14) Corten, H. T. and R. H. Sailors, “Relationship Between Material Fracture Toughness Using Fracture Mechanics and Transition Temperature Tests,” UILU-ENG 71-60010, August 1, 1971.

15) WCAP-7822, “Indian Point Unit No. 2 Reactor Internals Mechanical Analysis for Blowdown Excitation” (Non-Proprietary).

16) Housner, George W., “Vibration of Structures Induced by Seismic Waves,” <u>Shock and Vibration Handbook</u>, Volume 3, Chapter 50, ed. Harris and Crede.

17) DELETED 6/2000

18) WCAP-7408-L Topical Report, “Robert E. Ginna Unit #1 Internals Assurance Programs.”

19) Westinghouse NSD Technical Bulletin TB 75-3, “Reactor Coolant Pump (RCP) Vibration Limits for Type 93 & 93A Pumps”

20) Westinghouse Letter No. INT-98-227, “RCP Flywheel Integrity,” dated October 5, 1998.

21) Dominion Engineering Inc. Report R-4147-00-1, “Reactor Vessel Tensioning Optimization Stress Report – Indian Point Units 2 & 3.”

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	55,300	69,900
Head Flange	51,820	80,100
Vessel Flange	52,890	80,100
Closure Studs	109,400	110,300
Primary Nozzles – Inlet	45,500	80,100
Outlet	49,390	80,100
Core Support Pad	55,740	69,900
Bottom Head to Shell	37,700	80,100
Bottom Instrumentation	56,800	69,900
Nozzle Belt to Shell	42,630	80,100
CRDM Adapter Plug	27,630	48,600

TABLE 4.3-2

SUMMARY OF CUMULATIVE FATIGUE USAGE FACTORS FOR
COMPONENTS OF THE REACTOR VESSEL

Item	Usage Factor*
Control rod housing	0.124
Head Flange	0.024
Vessel Flange	0.023
Stud Bolts	0.944
Primary Nozzles – Inlet	0.049
Outlet	0.259
Core Support Pad (Lateral)	0.052
Bottom Head to Shell	0.020
Bottom Instrumentation	0.206
Nozzle Belt to Shell	0.002
CRDM Adapter Plug	0.0036

*As defined in Section III of the ASME Boiler and Pressure Vessel Code, Nuclear Vessel.

TABLE 4.3-3

STRESSES DUE TO MAXIMUM STEAM GENERATOR TUBE
SHEET PRESSURE DIFFERENTIAL (2485 PSI)

Stress	(668 F) Computed Value	Allowable Value
Primary Membrane Stress	6,960 psi	56,000 psi (0.70 Su)
Primary Membrane plus Primary Bending Stress	54,650 psi	84,000 psi (1.05 Su)

TABLE 4.3-4

RATIO OF ALLOWABLE STRESSES TO COMPUTED STRESSES
FOR A STEAM GENERATOR TUBE SHEET PRESSURE DIFFERENTIAL OF 2485 PSI

Component Part	Stress Ratio
Channel Head	1.90
Channel Head-Tube Sheet Joint	1.67
Tubes	2.08
Tube Sheet (Average Ligament)	1.54

TABLE 4.3-5

CHARPY V-NOTCH TEST RESULTS IP-3 FLYWHEELS AT +10°F
(CERTIFICATION DATA)

1. Flywheel No. 1

A. 5 In Thick Plate — Heat No. C4344 Slab No. 3

	1	2	3
"Weak" direction (ft-lbs)	30	32	33
90° to "weak" direction (ft-lbs)	60	75	80

B. 8 In Thick Plate — Heat No. C4908 Slab No. 1A

	1	2	3
"Weak" direction	57	57	53
90° to "weak" direction	38	44	49

2. Flywheel No. 2

A. 5 In Thick Plate — Heat No. C4679 Slab No. 3

	1	2	3
"Weak" direction	96	101	104
90° to "weak" direction	50	47	53

B. 8 In Thick Plate — Heat No. C4909 Slab No. 1

	1	2	3
"Weak" direction	58	52	57
90° to "weak" direction	95	78	77

3. Flywheel No. 3

A. 5 In Thick Plate — Heat No. C4679 Slab No. 3

Results as for Flywheel No. 2 (Case 2A)

TABLE 4.3-5
(Cont.)

CHARPY V-NOTCH TEST RESULTS IP-3 FLYWHEELS AT +10°F
(CERTIFICATION DATA)

B. 8 In Thick Plate — Heat No. C4909 Slab No. 1

Results as for Flywheel No. 2 (Case 2B)

4. Flywheel No. 4

A. 5 In Thick Plate — Heat No. C4679 Slab No. 3

Results as for Flywheel No. 2 (Case 2A)

B. 8 In Thick Plate — Heat No. C4909 Slab No. 1

Results as for Flywheel No. 2 (Case 2B)

4.4 SAFETY LIMITS AND CONDITIONS

4.4.1 System Heatup and Cooldown Rates

The operating limits for the Reactor Coolant System heatup and cooldown rates are defined in the Technical Specifications. These limits are recalculated periodically using methods derived from Appendix G, Protection Against Non-Ductile Failure, of Section III of the ASME Boiler and Pressure Vessel Code and ASME Code case N-514. The ASME approach utilizes fracture mechanics concepts and is based on the reference nil-ductility transition temperature, RT_{NDT}. The method calculation for the Indian Point 3 operating limits for heat up and cooldown rates is described in detail in Reference 10.

RT_{NDT} is defined as the greater of either the drop weight nil-ductility transition temperature or the temperature 60°F less than the 50 ft-lb (and 35-mil lateral expansion) temperature as determined from Charpy specimens oriented normal (transverse) to the major working direction of the material. The RT_{NDT} of a given material is used to index that material to a reference stress intensity factor curve (K_{IR} curve) which appears in Appendix G of the ASME Code. The K_{IR} curve is a lower bound of dynamic, crack arrest, and static fracture toughness results obtained from several heats of pressure vessel steel. When a given material is indexed to the K_{IR} curve, allowable stress intensity factors can be obtained for this material as a function of temperature. Allowable operating limits can then be determined utilizing these allowable stress intensity factors.

RT_{NDT} and, turn, the operating limits of the nuclear power plant can be adjusted to account for the effects of radiation on the reactor vessel material properties. The radiation embrittlement or changes in mechanical properties of the pressure vessel steel are monitored by the "Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program" (WCAP-8475, Reference 2). Surveillance capsules are periodically removed from the reactor, one at a time, and the encapsulated specimens are tested. The increase in the average Charpy V-notch 30 ft-lb temperature (ΔRT_{NDT}) due to irradiation is added to the original RT_{NDT} as an adjustment for radiation embrittlement. The radiation induced ΔRT_{NDT} can be estimated from Figure 4.4-1. The adjusted RT_{NDT} is used to index the material to the K_{IR} curve and, in turn, to reset the operating limits of the plant to take into account the effects of irradiation on the reactor vessel materials.

The first reactor vessel material surveillance capsule was removed during the 1978 refueling outage. The analysis of this capsule (T) (Reference 3) forms the basis of the original technical specification operating limits for RCS heatup and cooldown rates for up to 9.26 Effective Full Power Years (EFPY's). The second and third reactor vessel material surveillance capsules were removed in 1982 and 1987, respectively. These capsules have been tested and the results have been evaluated and reported (References 4,5,6,7). No change to the technical specification heatup and cooldown limit were required as a result of the Capsule Y and Capsule Z analyses.

In May 1991, 10 CFR 50.61 (Reference 9) was amended to include the provisions of Reg. Guide 1.99, Rev. 2 (Reference 8), which provided a slightly different methodology for calculation of reactor vessel embrittlement.

The heatup-cooldown curves were accordingly regenerated using this new rule for plant operating periods of 9, 11, 13.3, 16.2, and 20 EFPY. Future Technical Specifications Amendment proposals for greater operating periods will be submitted, using this new rule, as appropriate (References 10 and 12). The fourth surveillance capsule (Capsule X) was removed in 2003. No changes to the Technical Specifications were required as a result of Capsule X analysis (Ref. 13).

Transition temperature shifts occurring in the pressure vessel materials due to radiation exposure will continue to be obtained form ongoing and future capsule analyses in accordance with the Indian Point 3 reactor vessel radiation surveillance program.

The use of an RT_{NDT} that includes a ΔRT_{NDT} to account for radiation effects on the core region material, automatically provides additional conservatism for the non-irradiated regions. Therefore, the flanges, nozzles, and other regions not affected by radiation will be favored by additional conservatism approximately equal to the assumed ΔRT_{NDT}.

Table 4.4-1 provides the material toughness test requirements and data that were specified and reported for plates, forgings, piping, and weld material prior to plant operation. Specifically, the following data were provided:

1) The maximum NDT temperature as obtained from DWT results.
2) The maximum temperature corresponding to the 50 ft-lb value of the Cv fracture energy.
3) The minimum upper shelf Cv energy value for the weak direction (WR direction in plates) of the material.

For the limiting reactor vessel beltline materials, the end-of-life RT_{PTS} was estimated to be within the screening criteria of 10 CFR 50.61 for plate metal and welds (Reference 11). (Reg. Guide 1.99, Rev. 2 defines the property RT_{PTS} as an indicator of vessel embrittlement.)

The analysis of the reactor vessel material contained in surveillance Capsule T showed that the irradiated properties of the limiting reactor vessel beltline materials exceeded those predicted. However, due to the effects of low-leakage core loading strategy, measured beltline properties were within predicted values by the time Capsule Z was analyzed.

4.4.2 Reactor Coolant Activity Limits

Release of activity into the reactor coolant in itself does not constitute a hazard. Activity in the coolant could constitute a hazard only if the reactor coolant system barrier is breached, and then only if the coolant contains excessive amounts of activity which could be released to the environment. The plant systems were designed for operation with activity in the Reactor Coolant System corresponding to 1 percent fuel defect. The waste gas system is designed such that rupture of a gas decay tank, following a refueling shutdown wherein the gaseous activity is removed from the reactor coolant to the waste gas tanks for decay, will not result in an offset whole body exposure in excess of 0.5 rem. In the event of a steam generator tube rupture a high activity level signal at the condenser air ejector exhaust will divert the radioactive discharge back into Containment.

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% 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 and Technical Requirements Manual.

References

1) Hazeton, W. S., S. L. Anderson and S. E. Yanichko, "Basis for Heatup and Cooldown Limit Curves," WCAP-7924, July 1972.

2) Yanichko, S. E. and J. A. Davidson, "Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program," WCAP-8475, January 1975.

3) Davidson, J. A., S. L. Anderson and W. T. Kaiser, "Analysis of Capsule T from the Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program," WCAP-9491, April 1979.

4) Yanichko, S. E. and Anderson, "Analysis of Capsule Y From the Power Authority of the State of New York Indian Point Unit 3 Reactor Vessel Radiation Surveillance Program, Volume 1," WCAP-10300-1, March 1983.

5) Kaiser, W. T., "Analysis of Capsule Y From the Power Authority of the State of New York Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program, Volume 2," WCAP-10300-2, March 1983: Heatup and Cooldown Curves for Normal Operation.

6) Yanichko, S. E., "Analysis of Capsule Y From the Power Authority of the State of New York Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program, Volume 3," WCAP-10300-3, March 1983: J-Integral Testing of IX-WOL Fracture Mechanics Specimens.

7) Yanichko, S. E., "Analysis of Capsule Z From the New York Power Authority Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program," WCAP-11815, March 1988.

8) Regulatory Guide 1.99, Rev. 2

9) 10 CFR 50.61 "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," Federal Register, Vol. 56, No. 94, Effective May 15, 1991.

10) "Final Report on Pressure Temperature Limits for Indian Point 3 Nuclear Power Plant," ABB Combustion Engineering, July 1990.

11) Memorandum REC:92-003, dated February 10, 1992, F. Gumble to P. Kokolakis

12) "Indian Point Unit 3 Section XI Enable Temperatures for 13 and 15 EFPY," ABB Combustion Engineering, August 14, 1997.

13) Laubham, T., "Analysis of Capsule X from Entergy's Indian Point Unit 3 Reactor Vessel Radiation Surveillance Program," WCAP-16251-NP, Rev. 0, July 2004.

TABLE 4.4-1

MATERIAL TOUGHNESS TEST REQUIREMENTS AND DATA

Component	Material Grade	Cu (%)	(Drop Wt.) NDTT (F)	50 Ft-lb/35 Mil Temp. Long. (F)	50 Ft-lb/35 Mil Temp. Trans. (F)	RT_{NDT} (F)	Minimum Upper Shelf Long. (Ft-lb)	Minimum Upper Shelf Trans. (Ft-1b)
I. Reactor Vessel								
a. Closure Hd. Dome	A533,B,C1 1	0.13	+10	93	148[a]	88	85	55[b]
b. Clos. Hd. Peel Seg.	A533,B,C1 1	0.14	0	83	112[a]	52	108	70[b]
c. Clos. Hd Peel Seg.	A533,B,C1 1	0.14	+10	51	80[a]	20	128	83[b]
d. Clos. Hd. Peel Seg.	A533,B,C1 1	0.13	+10	78	99[a]	39	117	76[b]
e. Head Flange	A508, C1 2	NA	+ 3*	-22	- 8[a]	3	117	76[b]
f. Vessel Flange	A508, C1 2	NA	+38*	-5	16[a]	38	141	92[b]
g. Inlet Nozzle	A508, C1 2	NA	+20*	-28	-2[a]	20	154	100[b]
h. Inlet Nozzle	A508, C1 2	NA	+45*	-8	40[a]	45	120	78[b]
i. Inlet Nozzle	A508, C1 2	NA	+40*	-7	20[a]	40	158	103[b]
j. Inlet Nozzle	A508, C1 2	NA	+12*	-14	0[a]	12	155	101[b]
k. Outlet Nozzle	A508, C1 2	NA	+60*	60	150[a]	90	72.5	47[b]
l. Outlet nozzle	A508, C1 2	NA	+60*	4	44[a]	60	105.5	69[b]
m. Outlet Nozzle	A508, C1 2	NA	+60*	4	44[a]	60	96	62[b]
n. Outlet Nozzle	A508, C1 2	NA	+60*	-2	43[a]	60	123.5	80[b]
o. Upper Shell	A533, B,C1 1	NA	-50	74	128[a]	68	90 (95% shear)	58[b]
p. Upper Shell	A533, B,C1 1	0.20	-40	84	130[a]	70	100	65[b]
q. Upper Shell	A533, B,C1 1	NA	-40	43	82[a]	22	127	82.5[b]
r. Inter. Shell	A533, B,C1 1	0.20	-50	24	65**	5	134	97**

TABLE 4.4-1
(Cont.)

MATERIAL TOUGHNESS TEST REQUIREMENTS AND DATA

Component		Material Grade	Cu (%)	(Drop Wt.) NDTT (F)	50 Ft-lb/35 Mil Temp. Long. (F)	Trans. (F)	RT_{NDT} (F)	Minimum Upper Shelf Long. (Ft-lb)	Trans. (Ft-1b)
I.	Reactor Vessel								
s.	Inter. Shell	A533, B,C1 1	0.22	-50	23	56**	-4	113	86**
t.	Inter. Shell	A533, B,C1 1	0.20	-40	40	77**	17	113	85**
u.	Lower Shell	A533, B,C1 1	0.19	0	74	109**	49	90	65**
v.	Lower Shell	A533, B,C1 1	0.22	-20	6	55**	-5	134	89**
w.	Lower Shell	A533, B,C1 1	0.24	-10	68**	134**	74	105**	62**
x.	Bot. Hd.								
	Peel Seg.	A533, B,C1 1	0.13	-40	23	62[a]	2	103	67[b]
y.	Bot. Hd.								
	Peel Seg.	A533, B,C1 1	0.16	-40	33	56[a]	-4	108	70[b]
z.	Bot. Hd.								
	Peel Seg.	A533, B,C1 1	0.13	-40	38	69[a]	9	106	69[b]
aa.	Bottom Hd. Dome	A533, B,C1 1	0.13	-30	60	107[a]	47	80	52[b]
ab.	WELD	-	0.15**	0*	-	5**	0	-	112**
ac.	HAZ	-	NA	NA	-	10**	-	-	111**

NA – Not Available

* Estimated (60F or 100 Ft-lb temp., whichever is less for forgings; OF or 30 Ft-1b temp, whichever is higher for welds).
a) Estimated when no transverse data are available. (77 Ft-lb/54 mil longitudinal temp.)
b) Estimated when no transverse data are available. (65% of longitudinal shelf)
** Westinghouse data (all other data provided by the vessel fabricator).

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 non-destructive 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 were 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 were equivalent to those used for the reactor vessel.

Westinghouse required, as part of its reactor vessel specification, that certain special tests not specified by the applicable codes be performed. These tests are listed below:

1) Ultrasonic Testing – Westinghouse required that a 100% volumetric ultrasonic test (both shear wave and longitudinal wave) of reactor vessel plate be performed. Section III Class A vessel plates were required by code to receive only a longitudinal wave ultrasonic test. The 100% volumetric ultrasonic test by both techniques was a severe requirement, but it assured that the plates were of the highest quality.

2) Radiation Surveillance Program – In the surveillance program, the evaluation of the radiation damage was based on pre-irradiation testing of Charpy V-notch and tensile specimens and post-irradiation testing of Charpy V-notch, tensile and wedge opening loading (WOL) fracture mechanics test specimens.

4.5.2 Reactor Vessel Surveillance

The reactor vessel surveillance program is directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the transition temperature and the fracture mechanics approach. The reactor vessel surveillance program includes specimens from the most limiting plate used in the core region of the reactor vessel. Three capsules meet the requirements of ASTM-E-185-70 except that the limiting plate specimen orientation is transverse (weak direction) rather than longitudinal (strong direction). Five additional capsules do not meet E-185-70 since they do not include HAZ specimens. The program is essentially in accordance with ASTM-E-185-70, and ASTM-E-185-79 “Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels,” except that three capsules contain, in addition to the specified Charpy specimens taken from the weld metal, the heat affected zone and the ASTM reference plate, core region base metal specimens oriented normal (transverse) to the principal rolling direction of the plate rather than parallel to the principal rolling direction.

Five additional capsules, not required by ASTM-E-185 but included in the program, contain both longitudinal and transverse Charpy specimens taken from the limiting core region material, and longitudinal tensile and Charpy specimens of one of the other core region plates; however, they do not contain weld heat affected zone specimens of ASTM reference correlation monitor specimens. The surveillance program does not include thermal control specimens. These specimens were not required since the surveillance specimens are exposed to the combined neutron irradiation and temperature effects and the test results provide the maximum transition temperature shift. Thermal control specimens as considered in ASTM-E-185 would not provide any additional information on which the operational limits for the reactor are set.

Test stock from four plates (three intermediate shell plates and the most limiting lower shell plate) used in the core region of the reactor vessel, and a weldment containing representative (as deposited) weld metal (but no HAZ representing the limiting plate) were retained as per Section 3.1.2 of ASTM E-185-70.

Chemical analyses (excluding nitrogen and iron) as per Section 3.1.3 of ASTM #-185-70 were obtained for the limiting core region plate and weld metal.

The reactor vessel surveillance program uses eight specimen capsules. The capsules are located about 3 inches from the vessel wall directly opposite the center of the core and are retained in guide baskets welded to the outside of the thermal shield. Sketches of an elevation and plan view showing the location and dimensional spacing of the capsules with relation to the core, thermal shield and vessel are shown in Figures 4.5.1 and 4.5.2, respectively. The capsules can be removed when the vessel head is removed and can be replaced when the internals are removed. The capsules contain reactor vessel steel specimens from shell plates located in the core region of the reactor and from associated weld metal and heat affected zone metal. In addition, three capsules contain correlation monitors made from fully documented specimens of SA-533 Grade B, Class 1 material obtained through Subcommittee II of ASTM Committee E10, “Radioisotopes and Radiation Effects”. The capsules contain tensile specimens, Charpy V-notch specimens (which include weld metal and heat affected zone material) and WOL specimens.

Sixty-four Charpy V-notch specimens (oriented with respect to the weak direction) for one of the lower shell plates were included in the reactor vessel surveillance program. This lower shell plate is the limiting material in the core region as defined by E-185.

Dosimeters, including Ni, Cu, Co-Al, Ed shielded Co-Al, Cd shielded Np-237 and Cd shielded U-238, for capsules V, Y & S were placed in 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 were included to monitor temperature of the specimens. The specimens are enclosed in a tight fitting stainless steel sheath to prevent corrosion and to ensure good thermal conductivity.

The complete capsule was helium leak tested. Vessel material sufficient for at least 2 capsules will be kept in storage. This material represents four plates (three intermediate shell plates and the most limiting lower shell course plate) used in the core region of the reactor vessel and a representative weldment. As part of the surveillance program, a report of the residual elements in weight percent to the nearest 0.01% is made for surveillance material and as deposited weld metal.

Each of three capsules contain the following specimens:

CAPSULES V. Y***** and S

Material	No. of Charpys	No. of Tensiles	No. of WOL’s

Plate B2803-3*	8	2	-
Weld Metal	8	2	4
Heat affected zone metal	8	-	-
ASTM Reference	8	-	-

The following dosimeters and thermal monitors are included in each of the three capsules:

DOSIMETERS

Copper

Nickel

Cobalt – Aluminum (0.15% Co)

Cobalt – Aluminum (Cadmium shielded)

U-238 (Cadmium shielded)

Np-237 (Cadmium shielded)

THERMAL MONITORS

97.5% Pb, 2.5% Ag (579 F Melting Point)

97.5% Pb, 1.75% Ag, 0.75% Sn (590^{o} F Melting Point)

Five additional capsules contain the following specimens:

Capsules	Material	No. of Charpys	No. of Tensiles	No. of WOL's
W and T ****	Plate B2803-3*	8	-	-
	Plate B2803-3**	8	2	-
	Plate B2802-1***	8	1	6
	Weld Metal	8	-	-
X and U	Plate B2803-3*	8	-	-
	Plate B2803-3**	8	2	-
	Plate B2802-2***	8	2	6
	Weld Metal	8	-	-

Z*****	Plate B2803-3*	8	-	-
	Plate B2803-3**	8	2	-
	Plate B2802-3***	8	2	6
	Weld Metal	8	-	-

* Lower shell plate specimens oriented normal (transverse) to the principal rolling direction of the plate.

** Lower shell plate specimens oriented parallel (longitudinal) to the principal rolling direction of the plate.

*** Intermediate shell plate specimens oriented parallel (longitudinal) to the principal rolling direction of the plate.

**** Capsule T has been removed and analyzed.

***** Capsule Z has been removed and analyzed.

The following dosimeters and thermal monitors are included in each of the five capsules:

<u>Dosimeters</u>

Copper
Nickel
Cobalt-Aluminum (0.15% Co)
Cobalt-Aluminum (Cadmium shielded)
Iron

<u>Thermal Monitors</u>

97.5% Pb, 2.5% Ag (579°F Melting Point)
97.5% Pb, 1.75% Ag, 0.75% Sn (590°F Melting Point)

The fast neutron exposure of the specimens occurs at a rate equal to or faster than the maximum exposure experienced by the vessel wall with the specimens being located between the core and the vessel. Since some of these specimens experience accelerated exposure and are actual samples from the materials used in the vessel, the RT_{NDT} determinations of these specimens are representative of the vessel at a later time in service.

Data from the WOL fracture toughness specimens provide additional information for use in determining allowable stresses for irradiated material.
The calculated average fast neutron exposure at the vessel clad-base metal interface was 5.86 x 10^8 n/cm^2 (E greater than 1 MeV at the end of Cycle 12). The reactor vessel surveillance capsules are located at 4°, 40°, and 220° as shown in Figure 4.5-2. The design basis lead factor and the plant specific lead factor are listed below.

Capsules at	Design Basis Lead Factor	Plant Specific Lead Factor
4°	1.07	1.30
40°	3.74	3.74
220°	3.46	3.44

Correlations between the calculations and the measurements on the irradiated samples in the capsules, assuming the same neutron spectrum at the samples and the vessel inner wall, are described in Appendix 4A. The analysis of the reactor vessel material contained in Surveillance Capsule T for Indian Point 3 was reported in WCAP-9491, April 1979. The analysis of the reactor vessel material contained in Surveillance Capsule Y for Indian Point 3 was reported in WCAP-10300, March 1983. The analysis of the reactor vessel material contained in Surveillance Capsule Z for Indian Point 3 was reported in WCAP-11815, March 1988. The analysis of reactor vessel material contained in Surveillance Capsule X for Indian Point Unit 3 was reported in WCAP – 16251-NP, July 2004. The Capsule T report indicated that the damage rate of the plate and weld metal due to irradiation is in excess of that predicted by the Westinghouse trend curves and that the calculated lead factors were slightly higher than originally estimated. However, due to the effects of low-leakage core loading strategy, measured belting plate and weld metal properties were found to be slightly better than design by the time Capsule Z was analyzed.

The anticipated degree to which the specimens will perturb the fast neutron flux and energy distribution is considered in the evaluation of the surveillance specimen data. Verification and readjustment of the calculated wall exposure are made by use of data on all the capsules withdrawn as was done for Capsule T, Capsule Y, and Capsule Z.

The tentative schedule for removal of the capsules is as follows:

CAPSULE	REMOVAL TIME
T	Removed (1978 Refueling Outage, At the Replacement of the First Region of the Core, 1.34 EFPY*)
Y	Removed (1982 Refueling Outage, 3.13 EFPY)
Z	Removed (1987 Refueling Outage, 5.55 EFPY)
S	**
X	Removed (2203 Refueling Outage, 15.6 EFPY)
U	30 Years or 25.5 EFPY, assuming an 85% capacity)
V	Standby
W	Standby

*NOTE: Effective full power years from plant startup.

**Capsule S, scheduled for removal in the 2001 outage, was found to be inaccessible due to equipment interference and has therefore been removed from the program. The schedule for specimen retrieval beyond Capsule Z was revised in 2003 in order to optimize the benefits gained from specimen analysis in the latter half of plant live.

This plan was developed assuming an 85% capacity factor over plant life. Accordingly, the times for removal may be extended to allow for historical capacity factor below 85%.

4.5.3 Primary System Quality Assurance Program

Table 4.5-1 summarizes the quality assurance program with regard to inspections performed on primary system components, including the replacement steam generators installed during the cycle 6/7 refueling outage. In addition to the inspections shown in Table 4.5-1, there were those performed by the equipment supplier to confirm the adequacy of material received and those performed by the material manufacturer in producing the basic material. The inspections of reactor vessel, pressurizer, and steam generator were governed by ASME Code requirements. The inspection procedures and acceptance standards required on original pipe materials and piping fabrication were governed by USAS B31.1 and Westinghouse requirements, and were equivalent to those performed on ASME coded vessels. The loop 32 RCS hot leg elbow replaced in conjunction with the steam generator was fabricated and inspected to ASME Code Section III requirements.

Procedures for performing the examinations were consistent with those established in the ASME Code Section III and were 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 possible flaws, but equally as important, how the material was fabricated, the orientation and type of possible flaws, and the areas of most severe conditions. In addition, the surfaces most subject to damage as a result of the heat treating, rolling, forging, forming and fabrication processes received a 100% surface inspection by magnetic particle or liquid penetrant testing after all these operations were completed.

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% 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 and witnessed key inspections not only in the supplier’s 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, review of radiographs, performance of required test and qualification of supplier personnel. An independent surveillance of the conformance to the fabrication and installation specifications and the quality control requirements of, amongst other things, the original Reactor Coolant System components was carried out by the United States Testing Company for Consolidated Edison. Comparable independent surveillance was also carried out during fabrication and installation of the replacement steam generators.

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 assure that installation welds were of equal quality.

Consolidated Edison engineers witnessed the hydrostatic test of the reactor vessel.

Field erection and field welding of the Reactor Coolant System during original plant construction were performed so as to permit exact fitting of the 31" ID closure pipe subassemblies between the steam generator and the reactor coolant pump. After installation of the pump casing and the steam generator, measurements were taken of the pipe length required to close the loop. Based on these measurements, the 31" ID closure pipe subassembly was properly machined and then erected and field welded to the pump suction nozzle and to the steam generator exit nozzle. During replacement steam generator fabrication and installation, customized steam generator primary nozzle coordinates, temporary pipe restraints, mechanical and optical templating methods, and precision machining were all employed in order to ensure restoration of the RCS to its original configuration.

Cleaning of RCS 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 B&PV Code required that nozzles carrying significant external loads be attached to the shell by full penetration welds. This requirement was satisfied for 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, non-mandatory under Code rules, were performed on all weldments, including "P1" and "P3" materials which were the materials of construction in the reactor vessel, pressurizer and steam generators. Both preheat and postheat of weldments 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 post-heating achieves this by tempering any hard zones which may have formed due to rapid cooling.

Quality control techniques used in the fabrication of the Reactor Coolant System were equivalent to those used in the manufacture of the reactor vessel which conformed to Section III of the ASME Boiler and Pressure Vessel Code.

The piping was designed to the USAS B31.1 (1955) Code for Power Piping using the allowable stresses found in Nuclear Code Cases N-7 and N-10 for pipe and fittings, respectively. Results of piping reanalysis for seismic performance are presented in Section 16.3.5.

While the governing code for design, fabrication, inspection and testing of original RCS piping was USAS B31.1 (1955), the quality assurance requirements imposed by Westinghouse in the purchase and examination of the reactor coolant piping assured that the quality level of the plant is comparable to that delineated by USAS B31.7, Class I, Code for Nuclear Piping. This is demonstrated by the following comparison of original RCS quality assurance measures to selected provisions of USAS B31.7. The RCS reconnection activities associated with the replacement of steam generators during the cycle 6/7 refueling outage were governed by ANSI B31.1-1986 requirements but also met or exceeded the original RCS quality assurance measures, including the measures described below where applicable.

a) All materials conformed to ASTM specifications listed for B31.7 Class I, Nuclear Piping. In addition, all materials were certified, identified, and marked to facilitate traceability thus complying with the requirements of USAS B31.7, Class I, Code for Nuclear Piping

b) Piping base materials were examined by quality assurance methods having acceptance criteria which met the requirements set forth in USAS B31.7, Class I, Code for Nuclear Piping

c) All welding procedures, welding, and welding operators were qualified to the requirements of ASME Section IX, Welding Qualifications, which was in compliance with the requirements of USAS B31.7, Class I, Code for Nuclear Piping

d) All welds were examined by NDT methods and to the extent prescribed in USAS B31.7 for Class I, Nuclear Piping

e) All branch connection nozzle welds of nominal sizes of 3" and larger were 100% radiographed. This exceeded the requirements of USAS B31.7, Class I piping since it included nominal sizes of 6" and larger for 100% radiography

f) All finished welds were 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" and smaller were progressively examined after each ¼ inch increment of weld deposit in lieu of radiography

g) Hydrostatic testing was performed on the erected and installed piping. This requirement was the same as in USAS B31.7, Class I.

Hence, the Westinghouse quality assurance requirements implemented in the procurement of Indian Point 3 piping and fittings were equal to and in some instances exceeded the requirements of USAS B31.7.

The design and stress criteria specified in USAS B31.7 are not directly comparable to that of USAS B31.1 (1955 for piping design and 1967 for piping stress qualification). The following described how USAS B31.1 (1955 and 1967) were used in the primary coolant piping and the ASME B&PV Code Section III, Subsection NB, 1986 Edition for the pressurizer surge line including the effects of thermal stratification in the Indian Point Unit 3 design. A thermal expansion flexibility 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 1967) 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 and number of thermal cycles (usage factor) are safely 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 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 analyses were performed on the composite piping, which included the combined effects of all the sustained (pressure and weight) loading plus seismic vertical/horizontal loading components. The resultant reactions of the piping due to separate and combined effects of thermal, sustained and seismic loading were factored into the checking of the final design of the equipment nozzles with which the piping is interconnected. In turn, the equipment supporting structures were checked for adequate design including the added effects of these same loadings. Thus the total design analysis including pipe, equipment and structures considered the effects of thermal expansion, sustained and seismic loadings with a normal usage factor.

For considering and protecting against the dynamic effects of postulated ruptures, the Reactor Coolant Loop (RCL) LOCA analysis is performed for postulated breaks in the following branch lines:

- The Surge and the Residual Heat Removal (RHR) lines ion the hot leg,
- and the Accumulator line in the cold leg.

The RCL is also evaluated for the secondary side breaks at the main steam line and feedwater line terminal end nozzle locations at the steam generator.

Thermally induced stresses arising from temperature gradients were limited to a safe and low order of magnitude in assigning a maximum permissible time rate of temperature change on plant heat up, cool down, and incremental loadings in the plant operation procedure.

An added margin of conservatism was 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, and residual heat return nozzle connections to the primary coolant loop piping.

The use of thermal sleeves was not a specific requirement in B31.7. The seismic reanalysis effort for the Reactor Coolant System piping is described in Section 16.3.5.

Shop and field fabrication requirements, documentation, and quality assurance examinations all complied with those found in USAS B31.7 for Class I Nuclear Piping.

Electroslag Welding

The 90° elbows were electroslag welded. The following 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 B&PV Code, Section IX, and Code Case 1355 plus supplementary evaluations as requested by Westinghouse. The following test specimens were removed from a 5 inch thick weldment and successfully tested:
    a) 6 Transverse Tensile Bars – as welded
    b) 6 transverse Tensile Bars – 2050 F, H_20 Quench
    c) 6 Transverse Tensile Bars – 2050 F, H_20 Quench + 750 F stress relief heat treatment
    d) 6 Transverse Tensile Bars – 2050 F, H_20 Quench, tested at 650 F
    e) 12 Guided Side Bend Test Bars

2) The casting segments were surface conditioned for 100% 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 preparation 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. The, the electroslag weld and adjacent base material were 100% radiographed in accordance with ASME Code Case 1355. Also, the electroslag weld surfaces and adjacent base material were penetrant inspected in accordance with ASME Section III, Paragraph N-627.

5) Weld metal and base metal chemical and physical analysis were determined and certified.

6) Heat treatment furnace charts were recorded and certified.

Two of the Indian Point 3 reactor coolant pump casings were electroslag welded. The efforts discussed below were performed for quality assurance of the components:

1) The electroslag welding procedure employing “two- and three-wire” techniques was qualified in accordance with the requirements of the ASME B&PV Code Section IX and Code Case 1355 plus supplementary evaluations as required by Westinghouse. The following test specimens were removed from an 8 inch thick and from a 12 inch thick weldment and successfully tested for both the “2-wire” and “3-wire” techniques, respectively:

    a) Two wire electroslag process – 8” thick weldment
        1. 6 Transverse Tensile Bars – 750 F postweld stress relief
        2. 12 Guide Side Bend Test Bars

    b) Three wire electroslag process – 12” thick weldment
        1. 6 Transverse Tensile Bars – 750 F postweld stress relief
        2. 17 Guided Side Bend Test Bars
        3. 21 Charpy Vee 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.

2) A separate weld test was made using the “2-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 inch thick weldment in the stop-restart-repaired region and successfully tested:

a) 2 Transverse Tensile Bars – as welded
b) 4 Guided Side Bend Test Bars
c) Full section macroexamination of weld and heat affected zone.

3) All of the weld test blocks in 1) and 2) above were radiographed using a 24 MeV Betatron. The radiographic quality level obtained was between one-half of 1% to 1%. There were no discontinuities evident in any of the electroslag welds.

a) The casting segments were surface conditioned for 100% 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 4½ inches and ASTM E-280 severity level 2 for section thicknesses greater than 4½ inches. The penetrant acceptance standards were ASME B&PV Code Section III, Paragraph N-627.

b) The edges of the electroslag weld preparations were machined. These surfaces were penetrant inspected prior to welding. The acceptance standards were ASME B&PV Code Section III, Paragraph N-627.

c) The completed electroslag weld surfaces were ground flush with the casing surface. Then the electroslag weld and adjacent base material were 100% radiographed in accordance with ASME Code Case 1355. Also, the electroslag weld surfaces and adjacent base material were penetrant inspected in accordance with ASME B&PV Code Section III, Paragraph N-627.

d) Weld metal and base metal chemical and physical analyses were determined and certified.

e) Heat Treatment furnace charts were recorded and certified.

The two remaining Indian Point 3 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.

4.5.4 Non-Destructive Testing

Section XI of the ASME Boiler and Pressure Vessel Code sets the requirements for both the pre-operational and operational non-destructive testing of nuclear reactor coolant system.

The plant was examined to the fullest extent practical in accordance with Section XI, IS-141 and IS-142, even though this plant was ordered and designed before the code was effective. Examinations nonetheless followed coed requirements wherever the design of the plant allowed.

Non-destructive testing was performed by one of several methods, as specified in Section XI and its applicable reference:

1) Visual Examination

a) Direct Visual
b) Remote Visual
c) Indirect Visual

2) Surface Examination
a) Magnetic Particle
b) Liquid Penetrant

3) Volumetric Examination
a) Radiographic
b) Ultrasonic

Test personnel were qualified in accordance with all code requirements.

Pre-Service Inspection

Section XI, IS-232 required pre-operational examination of essentially 100% of the pressure containing welds within the reactor coolant system boundary.

The plant components were examined in accordance with the requirements wherever it was possible and practical to do so in order to provide base line date for subsequent inservice inspections.

The pre-service examination for the original plant components was performed at the plant site after the components had been installed. With the exception of the reactor coolant pipe to channel head weld which received a pre-service examination after the replacement steam generators were installed, pre-service examination of replacement steam generator pressure boundary welds was performed at the manufacturer's shop. Primary and secondary side hydrostatic tests were performed after installation. Personnel qualifications, equipment and records met the requirements of applicable codes. Onsite examinations were necessarily limited by the design and accessibility restrictions of the plant.

In-Service Inspection

Operational examinations as set forth in ASME Section XI are performed to the fullest extent practical at the required intervals.

The structural integrity of the Reactor Coolant System is maintained at the level required by the original acceptance standards throughout the life of the plant. Any evidence resulting from the inspections required by the ISI Program and indicating that potential defect implications have initiated or enlarged are investigated, including evaluation of comparable areas of the Reactor Coolant System.

Non-destructive test methods, personnel, equipment and records conform to the requirements of ASME Section XI.

The use of conventional non-destructive, direct visual and remote visual test techniques can be applied to the inspection of all primary loop components except for the reactor vessel. The reactor vessel presents special problems because of the radiation levels and remote underwater accessibility to this component. Because of these limitations on access to the reactor vessel, several features were incorporated into the design and manufacturing procedures in preparation for non-destructive test techniques as they become available:

1) Shop ultrasonic examinations were performed on all thermally clad surfaces to an acceptance and repair standard which assures an adequate cladding bond to allow later ultrasonic testing of the base metal. Size of cladding bonding defect allowed was ¼" x ¾"

2) The design of the reactor vessel shell in the core area is a clean, uncluttered cylindrical surface to permit future positioning of test equipment without obstruction

3) To establish baselines for Post-Operational Ultrasonic Testing of the Reactor Vessel, during the manufacturing stage selected areas of the reactor vessel were ultrasonic tested and mapped to facilitate the in-service inspection program.

   The areas selected for ultrasonic testing mapping included:
   a) Vessel flange radius, including the vessel flange to upper shell weld
   b) Middle shell course
   c) Lower shell course above the radial core supports
   d) Exterior surface of the closure head form the flange knuckle to the cooling shroud
   e) Nozzle to upper shell weld
   f) Middle shell to lower shell weld
   g) Upper shell to middle shell weld.

The pre-operational ultrasonic testing of these areas was performed after hydrostatic testing of the reactor vessel.

A qualified inspector employed by an insurance company authorized to write boiler and pressure vessel insurance certified all examinations.

Means of access to the Reactor Coolant Pressure Boundary were provided as necessary for the surveillance programs as detailed in the ISI Program. This inspection program is in compliance with Section XI of the ASME Code for in-service inspection of nuclear reactor coolant systems.

During the design phase, careful consideration was given to provide access for both visual and non-destructive in-service inspections of the reactor coolant primary and associated auxiliary systems and components within the boundaries established in accordance with the Section XI Code.

Specific provisions made for inspection access in the design of the reactor vessel, system layout and other major primary coolant components were:

1) All reactor internals are completely removable. The tools and storage space required to permit reactor internals removal for these inspections are provided

2) The reactor vessel shell in the core areas was designed with a clean, uncluttered cylindrical inside surface to permit future positioning of test equipment without obstruction

3) The reactor vessel cladding was improved in finish by grinding to the extent necessary to permit meaningful examination of the vessel welds and adjacent base metal in accordance with the Code

4) The cladding to base metal interface was ultrasonically examined to assure satisfactory bonding to allow the volumetric inspection of the vessel welds and base metal from the vessel inside surface

5) The reactor closure head is stored in a dry condition on the operating deck during refueling, allowing direct access for inspection

6) [Deleted]

7) Access holes were provided in the core barrel flange, allowing access for the remote visual examination of the clad surface of the vessel without removal of the lower internals assembly

8) Removable plugs were provided in the primary shield, providing limited access for inspection of the primary nozzle safe-end welds

9) Manways were provided in the steam generator channel head to provide access for internal inspection

10) A manway was provided in the pressurizer top head to allow access for internal inspection

11) The insulation covering all component and piping welds 6 inches in diameter and larger and covering the adjacent base metal was designed for ease of removal and replacement in areas where external inspection is planned

12) Removable plugs were provided in the primary shield concrete above the main coolant pumps to permit removal of the pump motor and to provide internal inspection access to the pumps.

The Indian Point 3 reactor vessel was built to the 1965 edition of the ASME Code Section III and all addenda through the Summer 1965 issue. ASME Section XI in-service inspection was not a requirement at that time. However, accessibility and techniques are available for inspecting all welds requiring inspection by ASME Section XI on the vessel, except for the closure head dome and bottom head dome circumferential welds and the control rod mechanism housing and bottom instrumentation tube attachment welds which were completed prior to the issuance of Section XI. Particular design improvements applied to the reactor vessel to facilitate in-service inspection include Items 1) through 4) and 7) above.

The data and results of the pre-operational examination serve as baseline data for the in-service inspection program.

In-service inspection of seismic Class I pressure retaining components, such as vessels, piping, etc. within the Reactor Coolant Pressure Boundary is performed in accordance with Section XI of the ASME Code, as described in the IP3 Inservice Inspection Program Plan for the applicable interval, with certain exceptions whenever specific relief is granted by the NRC.

The engineered safety features, the reactor shutdown systems, the cooling water systems, and the radioactive waste treatment systems which are necessary for plant operation are provided as redundant systems. This redundancy provides the capability for system and/or component outage (per Technical Specification requirements) to perform operability tests/checks or repair/maintenance. Periodic testing is in accordance with Technical Specification requirements and the IP3 Pump and Valve Testing Plan. The Pump and Valve Testing Plan is in accordance with ASME Section XI with certain exceptions whenever specific relief is granted by the NRC.

[Deleted]

Pre-operational Vibration Test Program

During hot functional testing, the piping was observed and any vibration problems were eliminated. Also, any other piping vibrations that were deemed excessive were eliminated. Observation for piping vibration was made by persons experienced in piping design when systems were operated in normal modes during hot functional testing. When piping vibrations were observed, and evaluation was made to determine corrective action.

Class I (seismic) systems were checked out and run prior to hot functional testing in accordance with Section 13.1.

During the normal course of the pre-operational test program, specific attention was directed to evaluating possible vibration problems during the performance of the following transients:

PRE-OPERATIONAL TEST	SPECIFIC TRANSIENTS
1. Reactor Coolant System Heatup	Operational Test of Charging Pumps (Step Changes) Reactor Coolant Pump Start Operation of Pressurizer Power-Operated Relief Valves Operation of Pressurizer Spray Valves Operation of Letdown Isolation Valves
2. Reactor Coolant System at Temperature	Operation of Pressurizer Power-Operated Relief Valve Reactor Coolant Pumps (Stopping and Starting)
3. Reactor Coolant System Cooldown	Initiation of Residual Heat Removal
4. Emergency Core Cooling Full Flow Test	Initiation and Termination of the Following:

		A. Safety Injection Pumps B. Residual Heat Removal Pumps
5.	Chemical and Volume Control System Test	Operational Test of Positive Displacement Charging Pumps (Stop and Start)

Amplitudes of vibration will theoretically cause the pipe to reach its elastic limit. Charts or monographs were provided during pre-operational testing to define these amplitudes as a function of pipe size, span and schedule as an aide for the operator and cognizant engineer to determine acceptability.

The acceptance of an observed vibration was based on operator and cognizant engineer experience. In addition, systems and components were physically examined (visually) for the following types of deficiencies which are indicative of a possible vibration problem:

1) Cracks in the grouting of equipment foundations
2) Leaking gaskets in piping systems and pump seals
3) Leaks from flanged connections in piping systems
4) Metal to metal contact indications on piping systems restraints.

If the above types of indications were observed, further investigation was performed to establish and correct any adverse conditions.

TABLE 4.5-1

REACTOR COOLANT SYSTEM
QUALITY ASSURANCE PROGRAM

Component	RT*	UT*	PT*	MT*	ET*	LT*
1. Steam Generator						
1.1 Tube Sheet						
1.1.1 Forging		yes		yes		
1.1.2 Cladding		yes(1)	yes			
1.2 Channel Head						
1.2.1 Forging		yes		yes		
1.2.2 Cladding		yes(1)	yes			
1.3 Secondary Sheet & Head						
1.3.1 Plates		yes				
1.3.2 Shell Transition Cone (forging)		yes		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)	yes			
1.6.4 Steam and Feedwater Nozzle to shell	yes			yes		
1.6.5 Support brackets				yes		
1.6.6 Tube to tube sheet			yes			yes
1.6.7 Instrument connections (primary and secondary)				yes		
1.6.8 Temporary attachments after removal				yes		
1.6.9 After hydrostatic test (all shell welds and Tube-sheet to channel head)				yes		
1.6.10 Nozzle safe ends (weld deposit)	yes		yes			
2. Pressurizer						
2.1 Heads						
2.1.1 Casting	yes			yes		
2.1.2 Cladding			yes			
2.2 Shell						
2.2.1 Plates		yes		yes		
2.2.2 Cladding			yes			

TABLE 4.5-1
(Cont.)

REACTOR COOLANT SYSTEM
QUALITY ASSURANCE PROGRAM

Component	RT*	UT*	PT*	MT*	ET*	LT*
2.3 Heaters						
2.3.1 Tubing(++++)		yes	yes			
2.3.2 Centering of element	yes					
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 surfaces after hydrostatic test				yes		
3. Piping						
3.1 Fittings (Castings)	yes		yes			
3.2 Fitting (Forgings)		yes	yes			
3.3 Pipe		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 inches)	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			

TABLE 4.5-1
(Cont.)

REACTOR COOLANT SYSTEM
QUALITY ASSURANCE PROGRAM

Component	RT*	UT*	PT*	MT*	ET*	LT*
5. Reactor Vessel						
5.1 Forgings						
5.1.1 Flanges		yes		yes		
5.1.2 Studs		yes		yes		
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(+++)	yes			
5.3.6 Nozzle Safe-Ends (If forging)		yes				
5.3.7 Nozzle Safe-Ends (If weld deposit)	yes		yes			
5.3.8 Head adapter forging to head adapter tube	yes		yes			
5.3.9 All welds after hydrotest				yes		
6. Valves						
6.1 Castings	yes		yes			
6.2 Forgings (No UT for valves two inch and smaller)		yes	yes			

* RT - Radiographic
UT - Ultrasonic
PT - Dye Penetrant
MT - Magnetic Particle
ET - Eddy Current
LT - Leak Testing (Helium)

(+) Flat Surfaces Only
(++) Weld Deposit Areas Only
(+++) UT of Clad Bond-to-Base Metal
(++++) Or a UT and ET
(1) For clad defects and for bond to base metal

APPENDIX 4A

NEUTRON DOSIMETRY

1. MEASUREMENT OF INTEGRATED FAST NEUTRON FLUX

In order to obtain a correlation between fast neutron (E greater than 1.0 MeV) exposure and the changes observed in radiation induced properties in the test specimens, a number of fast neutron monitors are included as an integral part of the Reactor Vessel Surveillance Program. In particular, the surveillance capsules contain detectors employing the following reactions:

Fe^{54} (n,P) Mn 54 (Fe from test specimens)
Ni^{58} (n,P) Co^{58}
Cu^{63} (n, ~) Co^{60}
Np^{237} (n,f) CS^{137}
U^{238} (n,f) Cs^{137}

In addition, thermal neutron flux monitors, in the form of bare and cadmium shielded Co-Al wire, are included within the capsules to enable an assessment of the effects of isotopic burnup on the response of the fast neutron detectors.

The use of activation detectors such as those listed above does not yield a direct measure of the energy dependent neutron flux level at the point of interest. Rather, the activation process is a measure of the integrated effect that the time and energy dependent neutron flux on the target material. An accurate estimate of the average neutron flux level incident on the various detectors may be derived from the activation measurements only if the parameters of the irradiation are well known. In particular, the following variables are of interest:

1. The operating history of the reactor
2. The energy response of the given detector
3. The neutron energy spectrum at the detector location
4. The physical characteristics of the detector

The procedure for the derivation of the fast neutron flux from the results of the Fe ^{54}(n,P)Mn^{54} reaction is described below. The measurement technique for the other dosimeters, which are sensitive to different portions of the neutron energy spectrum, is similar.

The Mn^{54} product of the Fe^{54}(n,P)Mn^{54} reaction has a half-life of 314 days and emits gamma rays of 0.84 MeV energy which are easily detected using gamma spectrometry. In irradiated steel samples, chemical separation of the Mn^{54} may be performed to ensure freedom from interfering activities. This separation is simple and very effective, yielding sources of very pure Mn^{54} activity. In some samples, all of the interferences may be corrected for by the gamma spectrometric methods without any chemical separation.

The analysis of the sample requires that two procedures be completed. First, the Mn^{54} disintegration rate per unit mass of sample and the iron content of the sample must be measured as described above. Second, the neutron energy spectrum at the detector location must be calculated.

For this analysis, the DOT[1] two-dimensional multigroup discrete ordinates transport code is employed to calculate the spectral data at the location of interest. Briefly, the DOT calculations utilize a 21 group energy scheme, an S_8 order of angular quadrature, and P_3 (Reference 2) expansion of the scattering matrix to compute neutron radiation levels within the geometry of interest. The reactor geometry employed here includes a description of the radial regions internal to the primary concrete (core barrel, neutron pad, pressure vessel and water annuli) as well as the surveillance capsule and an appropriate reactor and core baffle description. Thus, distortions in the fission spectrum due to the attenuation of the reactor internals are accounted for in the analytical approach.

Having the measured activity, sample weight, and neutron energy spectrum at the location of interest, the calculation of the threshold flux is as follows:

The induced Mn^{54} activity in the iron flux monitors may be expressed as

$$D = (No/A) f_i y \int_E \sigma(E)\phi(E) \, dE \sum_{J=1}^{N} F_J (1 - e^{-\lambda T_J}) e^{-\lambda (T - T_J)}$$

where:	D	=	Induced Mn^{54} activity	(dps/ gmFe)
	No	=	Avogadro's number	(atoms/ gm-atom)
	A	=	Atomic weight of iron	(gm/ gm-atom)
	f_i	=	Weight fraction of Fe^{54} in the detector	
	y	=	Number of product atoms produced per reaction	
	$\sigma(E)$	=	Energy dependent activation cross-section for the $Fe^{54}(n,P)Mn^{54}$ reaction	(barns)
	$\phi(E)$	=	Energy dependent neutron flux at the detector at full reactor power	(n/ cm^2-sec)
	λ |	=	Decay constant of Mn^{54}	(sec-1)
	F_J	=	Fraction of full reactor power during the Jth time interval, T_J	
	t_J	=	Length of the J^{th} irradiation period	(sec)
	T	=	Elapsed time between initial reactor startup and sample counting	(sec)

The parameters F_J, t_J, and T depend on the operating history of the reactor and the delay between capsule removal and sample counting.

The integral term in the above equation may be replaced by the following relation:

$$\int_E \sigma(E)\phi(E)\,dE = \bar{\sigma}\,\bar{\phi}E_{TH} = \overline{\phi}\,E_{TH}\sum_{E_{TH}}^{\infty}\sigma_S(E)\phi_S(E)\Big/\sum_{E_{TH}}^{\infty}\phi_S(E)$$

where: $\bar{\sigma}$ = Effective spectrum average reaction cross-section for neutrons above energy, E_{TH}

$\bar{\phi}E_{TH}$ = Average neutron flux above energy, E_{TH}

$\sigma_S(E)$ = Multigroup Fe^{54}(n,P)Mn^{54} reaction cross-section

compatible with the DOT energy group structure

$\phi_S(E)$ = Multigroup energy spectra at the detector location

obtained

from the DOT analysis

Thus,

$$D = (No/A)\, f_i\, y\, \bar{\sigma}\, \bar{\phi}E_{TH} \sum_{J=1}^{n} F_J \left(1 - e^{-\lambda T_J}\right) e^{-\lambda(T - T_J)}$$

or, solving for the threshold flux

$$\bar{\phi}E_{TH} = D/(No/A)\, f_i\, y\, \bar{\sigma} \sum_{J=1}^{n} F_J \left(1 - e^{-\lambda T_J}\right) e^{-\lambda(T - T_J)}$$

The total fluence above energy E_{TH} is then given by

$$\phi_{E_{TH}} = \bar{\phi}E_{TH} \sum_{J=1}^{n} F_J T_J$$

where $\sum_{J=1}^{n} F_J T_J$ represents the total effective full power seconds of reactor

operation up to the time of capsule removal.

Because of the relatively long half-life of Mn^{54}, the fluence may be accurately calculated in this manner for irradiation periods up to about two years. Beyond this time, the calculated average flux begins to be weighted toward the later stages of irradiation and some inaccuracies may be introduced. At these longer irradiation times, therefore, more reliance must be placed on the Np^{237} and U^{238} fission detectors with their 30 year half-life product (Cs^{137}).

No burnup correction was made to the measured activities, since burnout of the Mn^{54} product is not significant until the thermal flux level is about 10^{14} n/cm^2-sec.

The error involved in the measurement of the specific activity of the detector after irradiation is estimated to be 6 5 percent.

2. CALCULATION OF INTEGRATED FAST FLUX

The energy and spatial distribution of neutron flux within the reactor geometry is obtained from the DOT(1) two dimensional Sn transport code. The radial and azimuthal distributions are obtained from an R, u computation wherein the reactor core as well as the water and steel annuli surrounding the core are modeled explicity. The axial variations are then obtained from an R, Z DOT calculation using the equivalent cylindrical core concept. The neutron flux at any point in the geometry is then given by

$$\phi(E,R,\theta,Z) = \phi(E,R,\theta) \quad F(Z)$$

Where $\phi(E,R,\theta)$ is obtained directly from the R,θ calculation and $F(Z)$ is a normalized function obtained directly from the R,Z analysis. The core power distributions used in both the R,θ and R,Z computations represent the expected average over the life of the station.

Having the calculated neutron flux distributions within the reactor geometry, the exposure of the capsule as well as the lead factor between the capsule and the vessel may be determined as follows:

The neutron flux at the surveillance capsule is given by

$$\phi c \quad = \quad \phi(E, R_c, \theta_c, Z_c)$$

and the flux at the location of peak exposure on the pressure vessel inner diameter is

$$\phi_{v-\max} \quad = \quad \phi(E, R_v, \theta_{v-\max}, Z_{v-\max})$$

The lead factor then becomes

$$LF = \phi_c / \phi_{v-\max}$$

Similar expressions may be developed for points within the pressure vessel wall; and, thus, together with the surveillance program dosimetry, serve to correlate the radiation induced damage to test specimens with that of reactor vessel.

The specific activity of each of the activation monitors is determined by using established ASTM procedures.

References

1. R. G. Soltesz, et al., “Nuclear Rocket Shielding Methods, Modification, Updating, and Input Data Preparation, Volume 5 – Two-Dimensional Discrete Ordinates Technique, “ WANL-PR-(LL)-034, Aug. 1970

2. WCAP-11057, “Indian Point Unit 3 Reactor Vessel Fluence and RT-PTS Evaluations for Consideration of Life Extension,” Westinghouse Electric Corp, June 1989 Rev. 1

APPENDIX 4B

EVALUATION OF REACTOR COOLANT SYSTEM AND SUPPORTS UNDER COMBINED SEISMIC AND BLOWDOWN LOADS

1. Description of reactor coolant system component support structures
2. Analysis of reactor coolant system and supports under combined loads

APPENDIX 4C

PROCEDURE FOR PLUGGING A TUBE IN A STEAM GENERATOR
(DELETED)

APPENDIX 4D

SENSITIZED STAINLESS STEEL

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 (Westinghouse proprietary) which covers 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 which will cause cracking. Chlorides and flourides 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 3, 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 overlayed with type 312L and Inconel weld metal to eliminate the exposure of sensitized stainless steel in areas where there is limited accessibility for inservice inspection and plant maintenance. There is complete accessibility to the remaining RCS components. The pre-operational inspection of the RCS components provided assurance that there was no stress corrosion cracking of sensitized stainless steel.

All core structural load bearing members were made from annealed type 304 stainless steel, so there is no possibility of sensitization, with the exception of the core barrel itself, which required stress relief during manufacture at temperatures over 750 F. The stress relieving operation was conducted in a manner to minimize the possibility of severe sensitization, while maintaining the necessary conditions for relieving residual fabrication stresses. This consisted of heating to 1650 F to 1750 F, holding at this temperature for several hours, then cooling very slowly in the furnace. This treatment results in massive carbide precipitation at the grain boundaries, and agglomeration of the carbides, instead of the formation of detrimental continuous carbide films. Further, the long times at high temperatures cause diffusion of chromium into the grain boundary areas that were depleted in chromium by the precipitation of chromium carbides. This combination of formation of massive carbides, plus diffusion of chromium back into the depleted zone is referred to as "desensitization", and is commonly used to prevent severe sensitization of parts requiring heat treatments that otherwise would cause severe sensitization of the material. Stress tests run according to ASTM A393 were performed on core barrel material given this heat treatment, and results verified that severe sensitization is prevented. Material that does not or would not be expected to pass ASTM A393 is considered to be severely sensitized.

Severe sensitization of component parts of the Reactor Coolant Pressure Boundary was avoided by the following methods:

- Reactor Vessel - The type 316 safe ends were overlaid with Inconel and type 308L weld metal on the ID and OD after post welding heat treatment.
- Steam Generators - The safe ends were made of weld metal, type 309 and 308L, containing enough ferrite to preclude severe sensitization.
- Pressurizer - The nozzles were made of type 316 stainless steel, but the very short post welding heat treatment time used (less than 10 hours) is not expected to cause severe sensitization of type 316 which is more resistant to sensitization than type 304.

The welding processes employed for field use were shielded metal arc welding (SMAW) and gas tungsten arc welding (GTAW). Both welding processes were used individually or in combination to qualify procedures per the requirements of ASME Code Section IX. Quality controls employed included the use of qualified weld and inspection procedures as well as verification of the maximum interpass temperatures by use of Tempil-Stiks or contact pyrometers.

The effect of nitrogen addition on the corrosion resistance of stainless steel in the PWR environment is discussed in WCAP-7735, "Topical Report - Sensitized Stainless Steel in W PWR NSSS", August 1971.

Further justification that nitrogen addition does not adversely affect the corrosion resistance of sensitized austenitic stainless steel can be obtained from the following literature:

1) C. J. Smithells, Metal Reference Book, Vol. II, p. 621, Plenum Press (1967).

2) L. R. Scharistein, "Effects of Residual Elements on the General Corrosion Resistance of Austenitic Stainless Steels", Effects of Residual Elements on Properties of Austenitic Stainless Steels, ASTM, STP 418 (1967).

3) R. B. Gunia, G. R. Woodrow, "Nitrogen Improves Engineering Properties of Chromium-Nickel Stainless Steels", Journal of Metals , Volume 5, No. 2, p. 413, June (1970).

4) Jones and Laughlin data, sheet-type 304-N Stainless Steel, Jones and Laughlin Steel Corp., Stainless and Strip Division, Warren, Michigan.

All piping in Indian Point 3 was fabricated in a manner to assure that it will not be sensitized. All pipes and fittings were purchased in the sensitized (carbide solution treated) condition. Heat treatment after bending was also a carbide solution treatment at 1900 F or above. No heat treatment was permitted after welding. Welding was done with closely controlled interpass temperature, both in the shop and in the field, to assure freedom from sensitization.

Reactor Coolant System Nozzle Safe Ends

1. Reactor Vessel Primary Nozzle Safe Ends

   A. Method of Fabrication (See Figure 4D-1)

1) Wrought Stainless Steel - A-508, class 2 nozzle forging clad with 308 stainless steel welded to type 316 forging with Inconel weld metal. Attached prior to final post weld heat treatment.

2) Forging was overlayed on ID and OD with type 312L stainless and Inconel weld metal. This was performed in the shop after the final post weld heat treatment.

B. Inspection

1) Forging Safe Ends were examined by UT and PT at Combustion Engineering using Section III acceptance standards.

2) Weld overlay of the ID and OD surfaces was examined by UT and PT. The acceptance standards are shown below:

Ultrasonic Acceptance Standards

Rejectable Defect Indications:

a) Those exceeding 90% screen height and exceeding 1/2" length
b) Those exceeding 90% screen height and 1/2" or less in length if not separated by 2" from a similar indication
c) Those with range of 50% to 90% screen height and exceeding 1-1/2" in length
d) Those with range of 50% to 90% screen height and 1" to1-1/2" in length if not separated by 2" from a similar indication.

Penetrant Inspection Acceptance Standards

The following relevant indications were unacceptable:
a) Any cracks and linear indications
b) Rounded indications with dimensions greater than 3/16"
c) Four or more rounded indications in a line separated by 1/16 in or less edge-to-edge
d) Ten or more rounded indications in any six square inches of surface with the major dimension of this area not to exceed six inches with the area taken in the most unfavorable location relative to the indications being evaluated.

2. Steam Generator Primary Nozzle Safe Ends (See Figure 4D-2)

A. Method of Fabrication

308 stainless steel weld metal buttering applied to low alloy steel (SA508 Class 3 forging) nozzles prior to final post weld heat treatment. Stainless weld metal for the first layer is type 309L (modified) and for the balance is type 308L.

B. Inspection

Buttered safe ends were examined by PT and RT using ASME B&PV Code Section III acceptance standards.

3. Pressurizer (See 4D-3)

A. Method of Fabrication

Wrought stainless steel pipe or Type 316 forgings welded to carbon steel (A-216 Grade WCC Casting with 308 stainless steel cladding) nozzles with type 309 (modified) and 308L weld metal before PWHT. The surge nozzle safe end is fabricated from SA- 312 pipe, type 316 and the spray, relief, and safety nozzle safe ends from SA-182 forgings, type 316.

B. Inspection

Wrought material was examined by UT and PT using Section III acceptance standards.
Reactor Coolant System Construction

All primary piping and fittings were given a solution annealing treatment consisting of heating to 1900 - 1950 F, holding 1 hour per inch of thickness and water quenching. This assured that the material would not be sensitized.

Main coolant pipe welds are of type 308L or 316 stainless steel. Welding was performed during original plant construction 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"), and the interpass temperature control of 350 F maximum, there was no sensitization of the solution treated pipe during welding.

Comparable welding controls to avoid primary piping sensitization were also employed during steam generator replacement, however, automatic gas metal arc welding processes were used after the root and hot passes were manually completed. The use of inserts was not required during the reconnection of primary piping to the replacement steam generators.

Venting provisions were made at high points throughout the Reactor Coolant System to relieve entrapped air when the system is filled and pressurized. Principally, vents were installed on the reactor vessel head, the pressurizer, and the reactor coolant pumps. Additional vents are 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 utilized. 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 Requirements Manual.

<u>Reactor Coolant System Operational Stresses</u>

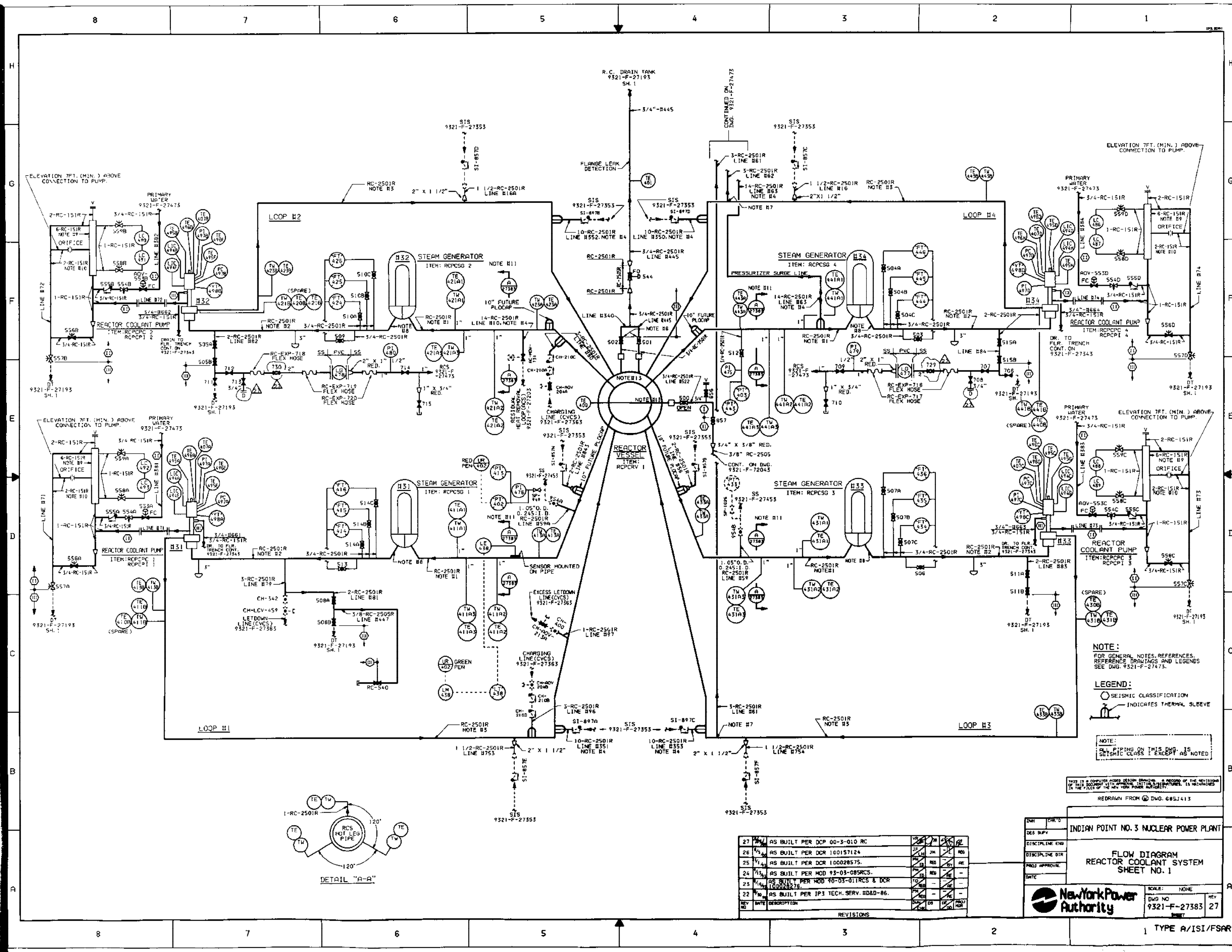
To avoid unusual stresses in areas where nozzle safe ends are joined to the piping, precautions were taken to eliminate unnecessary stresses due to erection of the various components of the Reactor Coolant System. The primary coolant system piping closure pieces are two pipe fitting subassemblies located 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 was free from cold spring. During steam generator replacement, customized steam generator primary nozzle coordinates, temporary pipe restraints, mechanical and optical templating methods, and precision machining were all employed to ensure restoration of the RCS to its original configuration.

As a precaution that the behavior of the Reactor Coolant System during operating conditions was as predicted, measurements were made during incremental temperature increases during the hot functional test. The measurements were made to check the movement of the components at temperature and pressure to insure interferences were not present. The data taken during the test were compared with the flexibility analysis predictions and evaluated.

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 post-operational inspection plan was developed for the nozzle safe ends within the Reactor Coolant System Boundary. The pressurizer and steam generator stainless steel safe ends which 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 of the safe end, approximately 120 degrees of the top segment of the safe ends are accessible for direct visual and surface examination.

A specially designed in-vessel, remote, ultrasonic, inservice inspection tool was developed which can be affixed to the upper vessel flange after removal of the head. This tool is intended for ultrasonic examination of the vessel circumferential and longitudinal welds, nozzle-to-vessel welds and nozzle-to-safe-end welds. Some of these examinations utilizing this invessel tool were performed in the 1979 refueling outage. No indications were revealed by this testing.



REV NO	DATE	DESCRIPTION
27		AS BUILT PER DCP 00-3-010 RC
26		AS BUILT PER DCR 100157124
25		AS BUILT PER DCR 100028575.
24		AS BUILT PER MOD 93-03-085RCS.
23		AS BUILT PER MOD 90-03-011RCS & DCR 100028276.
22		AS BUILT PER IP3 TECH. SERV. #D&D-86.

REVISIONS

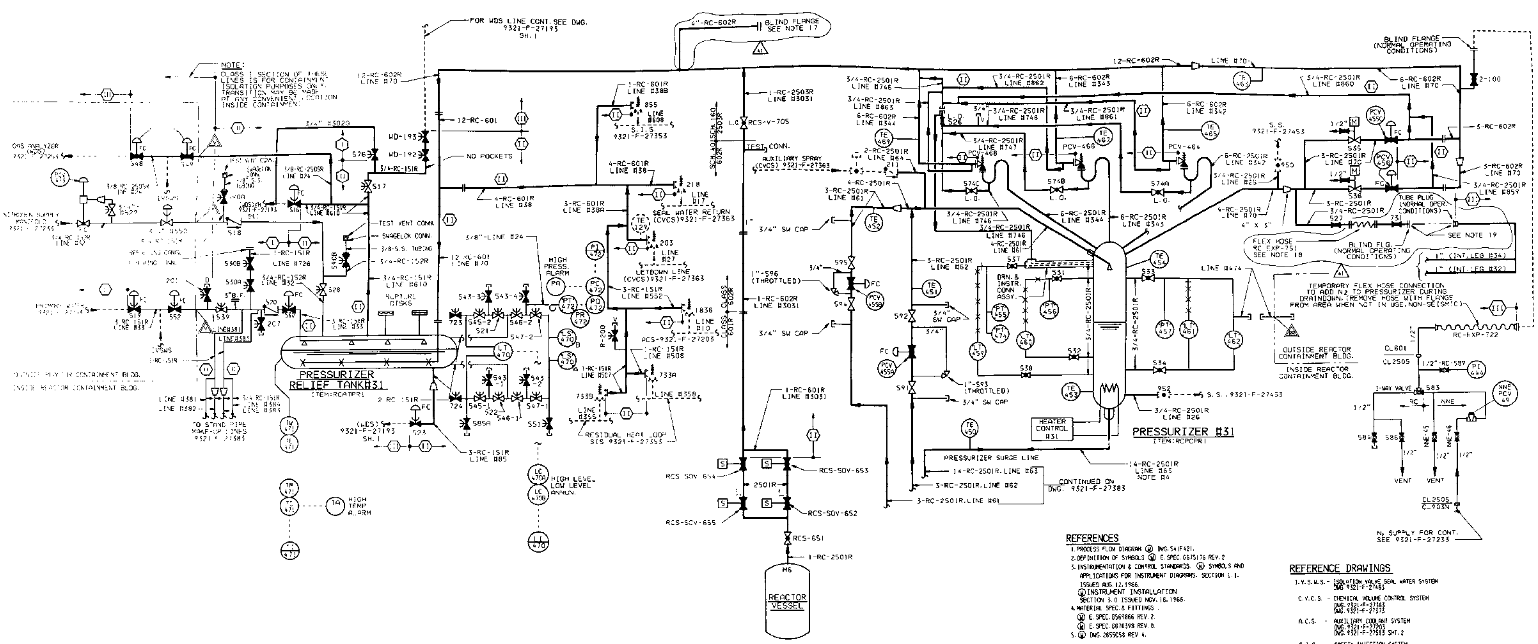
INDIAN POINT NO. 3 NUCLEAR POWER PLANT

FLOW DIAGRAM
REACTOR COOLANT SYSTEM
SHEET NO. 1

New York Power Authority

SCALE: NONE
DWG NO 9321-F-27383
REV 27

TYPE A/ISI/FSAR



REFERENCES

1. PROCESS FLOW DIAGRAM Ⓦ DWG. 541F421.
2. DEFINITION OF SYMBOLS Ⓦ E. SPEC. G675176 REV. 2
3. INSTRUMENTATION & CONTROL STANDARDS. Ⓦ SYMBOLS AND APPLICATIONS FOR INSTRUMENT DIAGRAMS. SECTION 1.1. ISSUED AUG. 12, 1966.
   Ⓦ INSTRUMENT INSTALLATION SECTION 3.0 ISSUED NOV. 16, 1966.
4. MATERIAL SPEC. & FITTINGS.
   Ⓦ E. SPEC. G569866 REV. 2.
   Ⓦ E. SPEC. G676398 REV. 0.
5. Ⓦ DWG. 2655C58 REV 4.

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-LOCATED 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.
9-7FT. LG., 6" DIA. SCHEDULE 10S STAND PIPE. BOTH ENDS CAPPED.
10-PIPE SLOPED DOWNHILL TO DRAIN TANK.
11-ALL SCOOPS LOCATED IN THE SAME CROSS-SECTIONAL PLANE.
12-ROOT VALVE IS LOCATED ABOVE ELEVATION OF REACTOR VESSEL NOZZLES.
13-VESSEL VENT LINE AND GASKET MONITORING LINES FURNISHED WITH VESSEL.
14-VALVES 539, 540, 541 & 542 ARE LOCATED BELOW HOT LEG PIPING.
15-FOR SEAL LEAKOFF SEE DWG. 9321-F-26473 FOR VALVES 535, 536, 455A & B.
16-ALL VALVE NO'S. ARE PRECEDED BY "RC" UNLESS OTHERWISE NOTED.
17-THIS CONNECTION SHALL BE USED FOR THE RCS VACUUM REFILL CONNECTION. SEE OPERATING PROCEDURE NO. SOP-RCS-017
18-FLEX HOSE RC-EXP-751 IS A SPECIALTY ITEM NOT GOVERNED BY PIPE CLASS 2501 AND IS RATED FOR THE PRESSURIZER DESIGN CONDITIONS.
19-THESE CONNECTIONS ARE USED WITH LT-390 DURING REDUCED INVENTORY OPERATIONS.

41

REFERENCE DRAWINGS

I.V.S.W.S. - ISOLATION VALVE SEAL WATER SYSTEM DWG. 9321-F-27463
C.V.C.S. - CHEMICAL VOLUME CONTROL SYSTEM DWG. 9321-F-27363 DWG. 9321-F-27373
A.C.S. - AUXILIARY COOLANT SYSTEM DWG. 9321-F-27203 DWG. 9321-F-27513 SHT. 2
S.I.S. - SAFETY INJECTION SYSTEM DWG. 9321-F-27353 DWG. 9321-F-27503
S.S. - SAMPLING SYSTEM DWG. 9321-F-27453
W.D.S. - WASTE DISPOSAL SYSTEM DWG. 9321-F-27193 SHT. 1 DWG. 9321-F-27505
R.C.S. - REACTOR COOLANT SYSTEM DWG. 9321-F-27383
NUCLEAR LINE LIST DWG. 9321-C-27413
REACTOR HEAD VENT SYSTEM DWG. 6604-171-F-1

LEGEND

D T - DRAIN TANK (WDS)
V - VENT TO ATMOSPHERE
F O - FAIL OPEN
F C - FAIL CLOSED
T - TRIP ON CONTAINMENT ISOLATION SIGN
V H - VENT HEADER
D - LOCAL DRAIN
△ CONTAINMENT PENETRATION
⬡ SEISMIC CLASSIFICATION

NOTE:
ALL PIPING ON THIS DWG. IS SEISMIC CLASS I EXCEPT AS NOTED.

REDRAWN FROM Ⓦ DWG. 685J413

REV NO	DATE	DESCRIPTION	DWN/CHK	DS	DE/DD	PROJ MGR
41	[illegible]	AS BUILT PER DCP 00-3-010 RC	VR/JF	JM	-/SP	[illegible]
40	2/5/01	AS BUILT PER DCR 100159871	JF/VR	JN	LC/SP	REG
39	1/?/00	AS BUILT PER DCR 100159105	JF/VR	JN	-/-	REG
38	3/1/00	AS BUILT PER DCR 100155808	VR/LH	JH	-/-	REG
37	1/24/??	AS BUILT PER DCR 100149154	VR/LH	JH	-/-	REG

REVISIONS

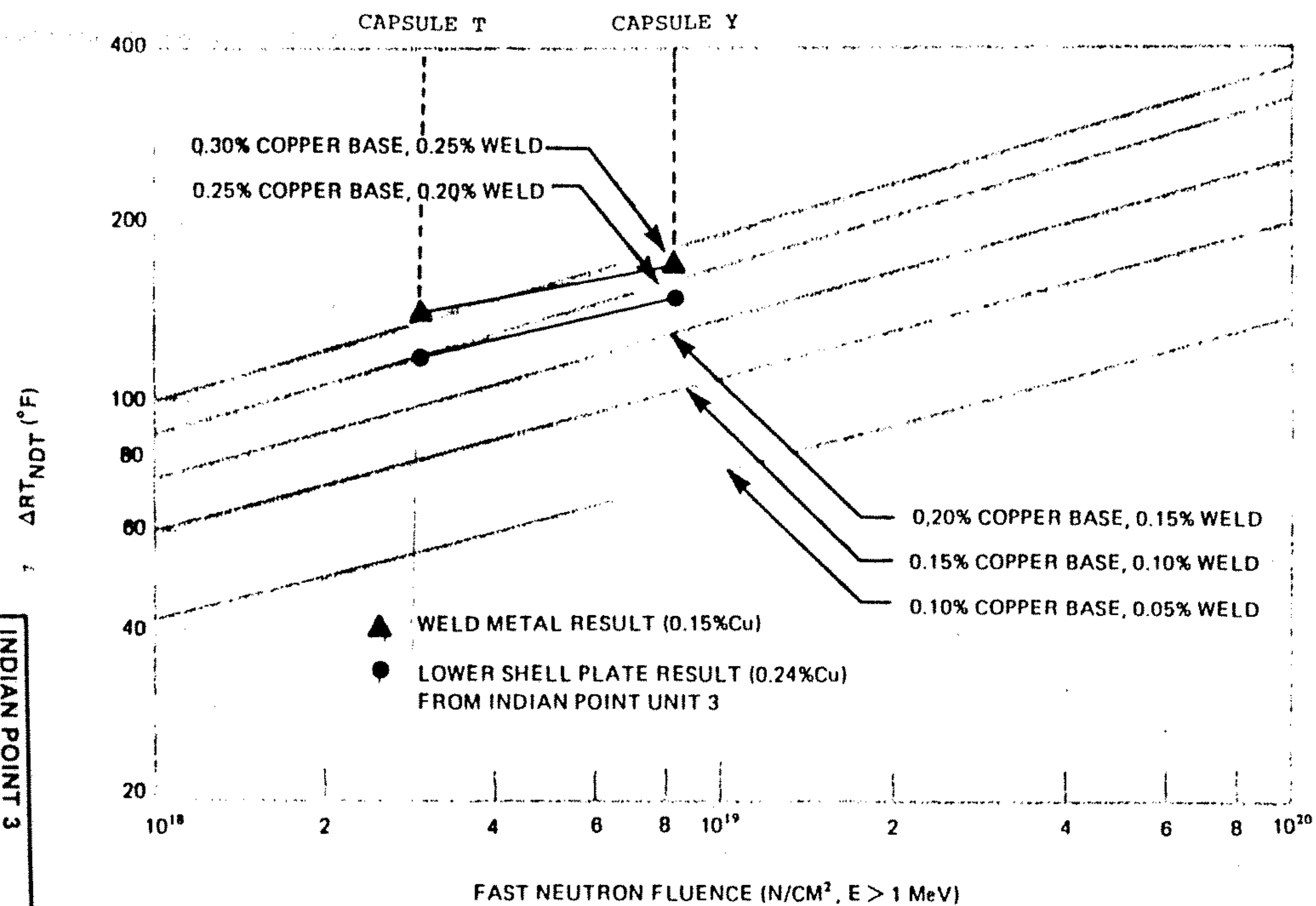
DWN/CHK'D	INDIAN POINT NO. 3 NUCLEAR POWER PLANT
DES SUPV	
DISCIPLINE ENG	FLOW DIAGRAM REACTOR COOLANT SYSTEM SHEET NO. 2
DISCIPLINE DIR	
PROJ APPROVAL	
DATE	

NewYorkPower Authority

SCALE NONE
DWG NO 9321-F-27473 SHEET
REV 41

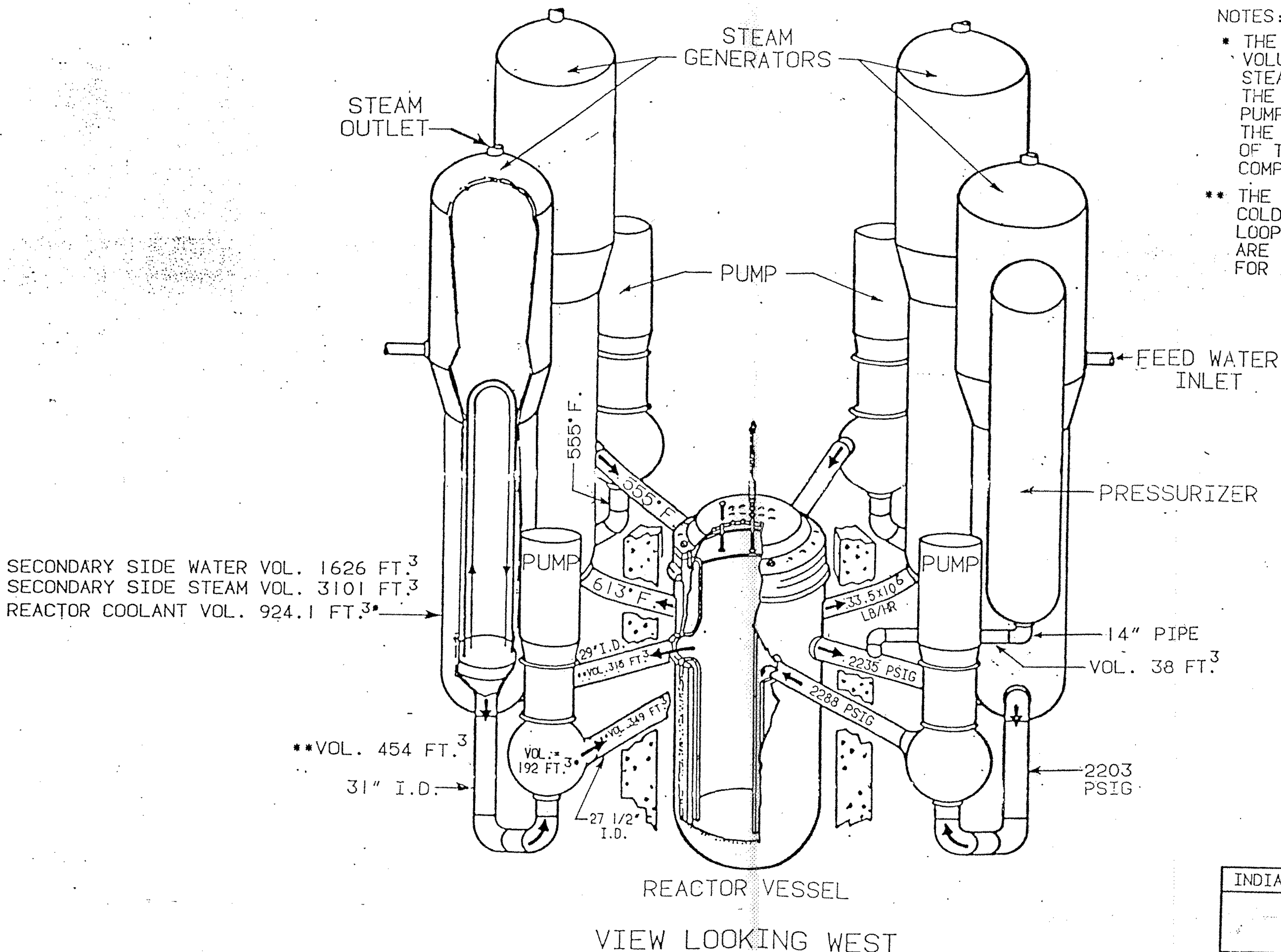
TYPE A/ISI/FSAR

THIS IS A COMPUTER AIDED DESIGN DRAWING. A RECORD OF THE REVISIONS OF THIS DOCUMENT WITH APPROVAL INITIALS/SIGNATURES, IS MAINTAINED IN THE FILES OF THE NEW YORK POWER AUTHORITY.



Effect of Fluence and Copper on Shift of RT_{NDT} for Reactor Vessel Steels Exposed to Irradiation at 550° F

INDIAN POINT 3		FSAR UPDATE
EFFECT OF FLUENCE AND COPPER CONTENT ON SHIFT OF RT_{NDT} FOR REACTOR VESSEL STEELS EXPOSED TO 550°F TEMPERATURE		
REV. 1	JULY, 1984	FIGURE NO. 4.4-1

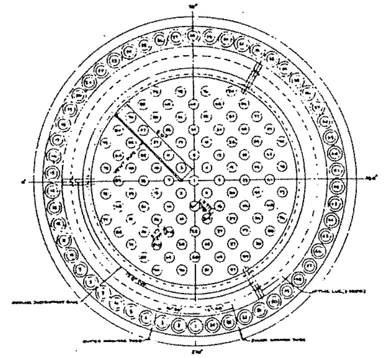


NOTES:

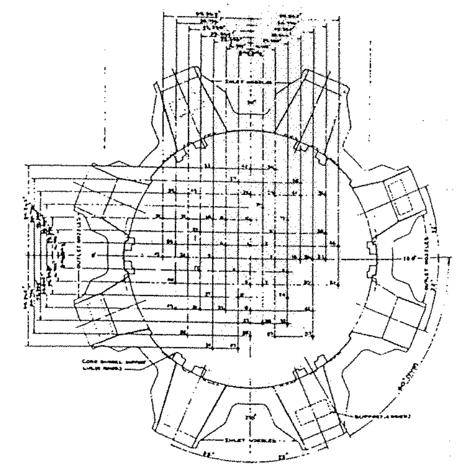
* THE REACTOR COOLANT VOLUME SHOWN FOR THE STEAM GENERATOR AND THE REACTOR COOLANT PUMP VOLUME SHOWN ARE THE VOLUMES FOR EACH OF THE FOUR RESPECTIVE COMPONENTS.

** THE HOT LEG VOLUME, COLD LEG VOLUME AND LOOP SEAL VOLUME SHOWN ARE THE COMBINED VOLUMES FOR ALL FOUR LOOPS.

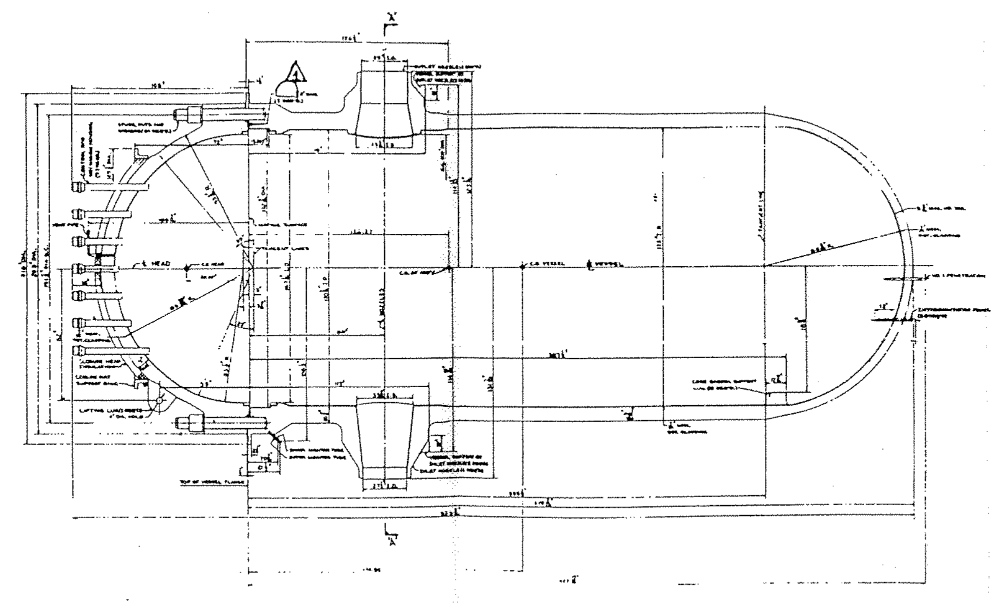
INDIAN POINT 3	FSAR UPDATE
FLOW DIAGRAM REACTOR COOLANT SYSTEM SCHEMATIC	
REV. 2, JULY, 1991	FIGURE NO. 4.2-2



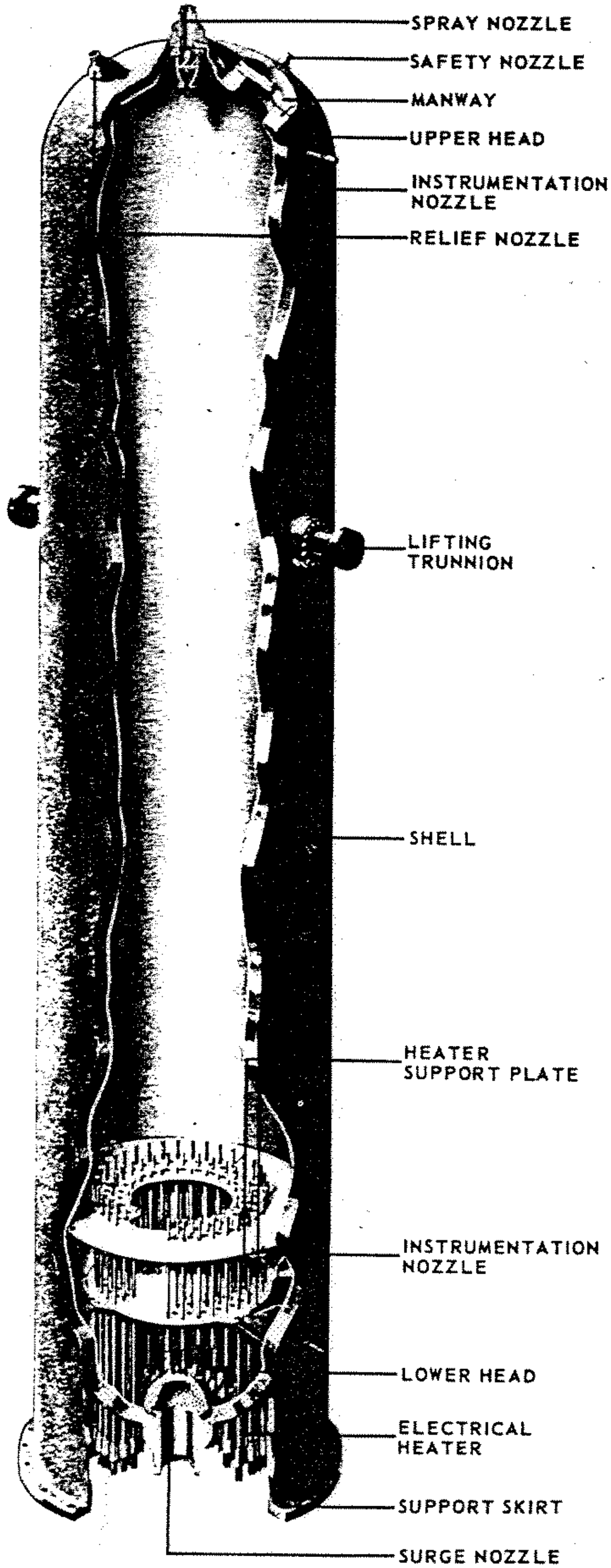
PLAN VIEW



SECTION 'A-A'

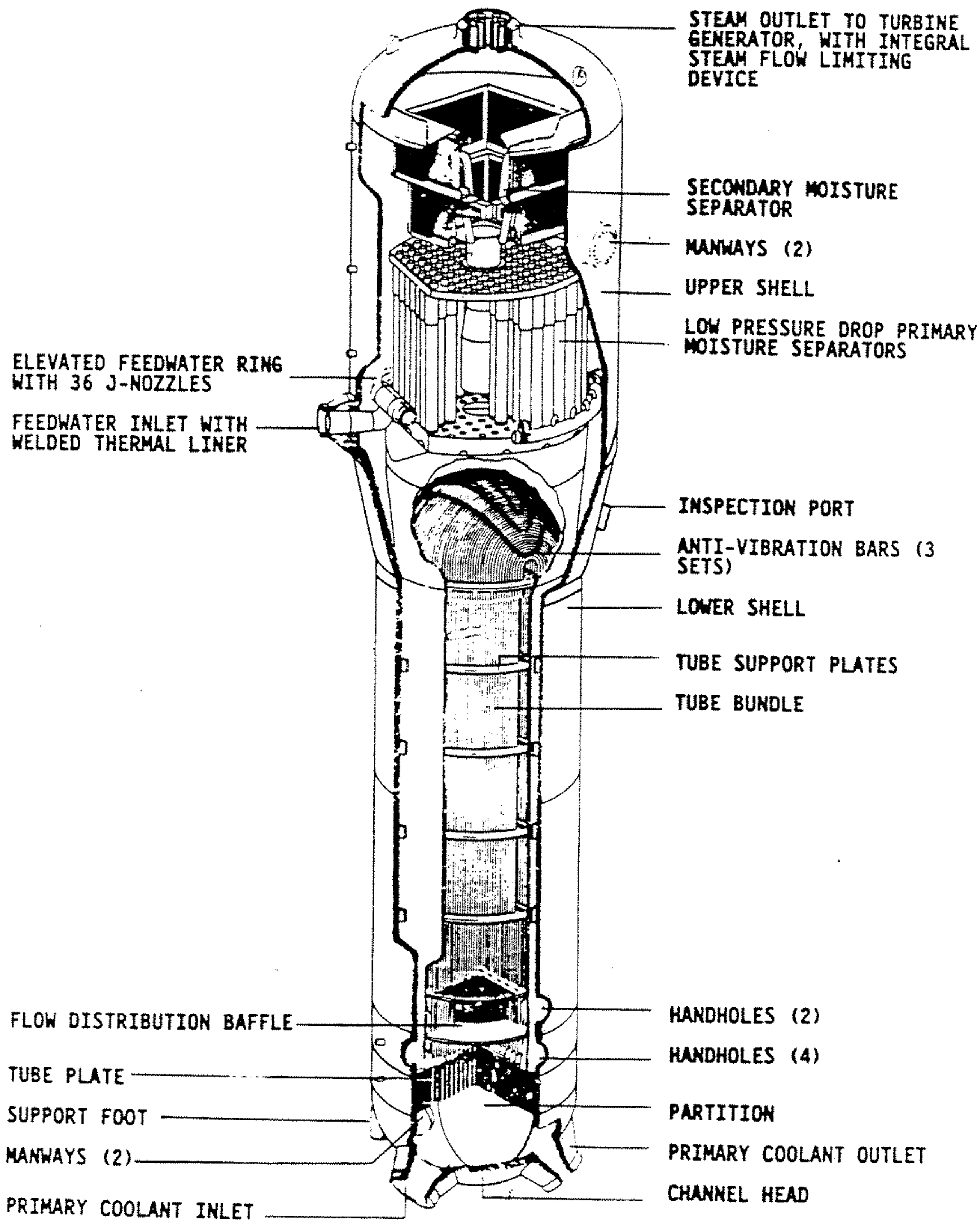


INDIAN POINT 3		FSAR UPDATE
REACTOR VESSEL		
REV. 1	DEC 1995	FIGURE NO. 4.2-3

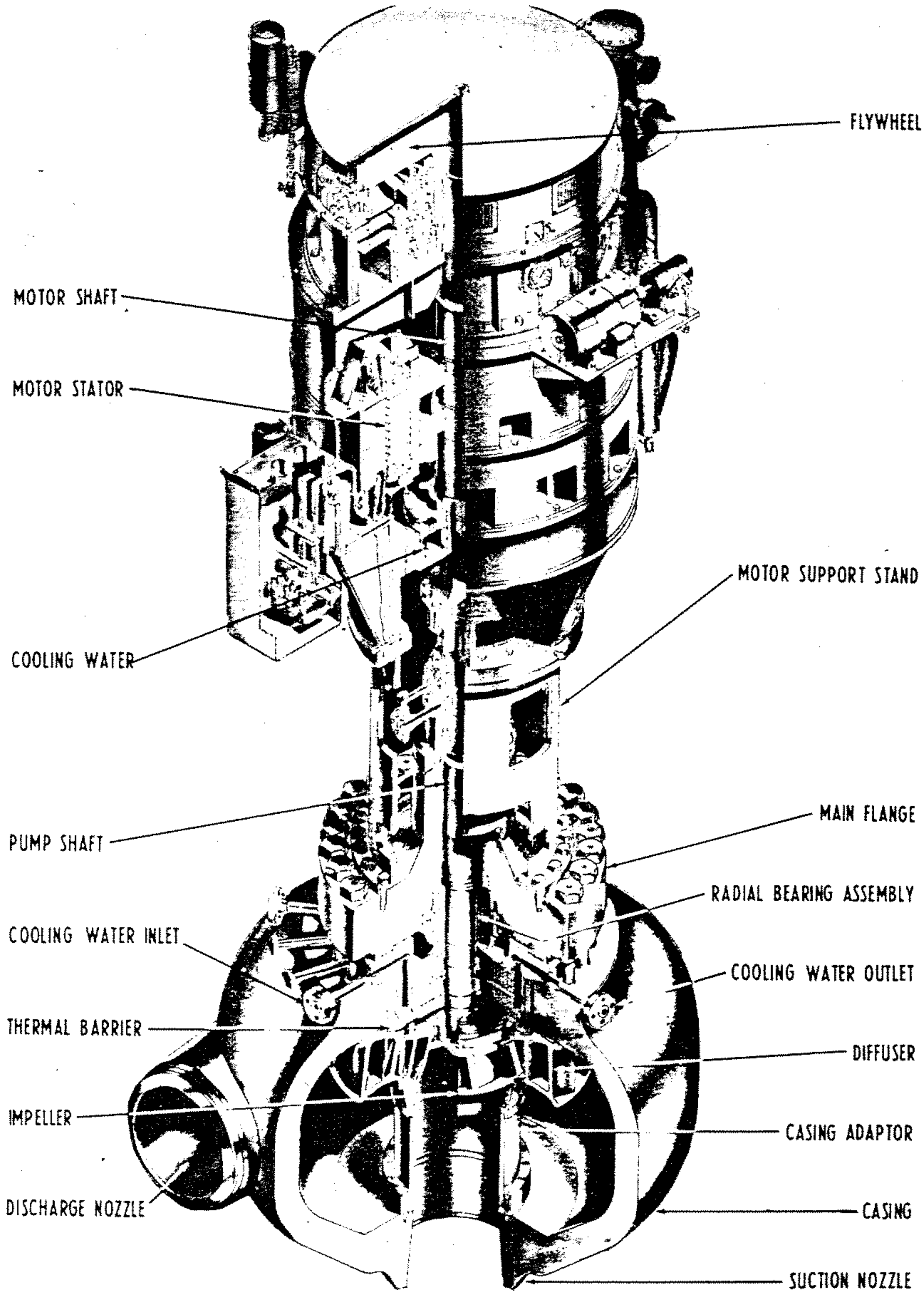


INDIAN POINT 3		FSAR UPDATE
PRESSURIZER		
REV. 0	JULY, 1982	FIGURE NO. 4.2-4

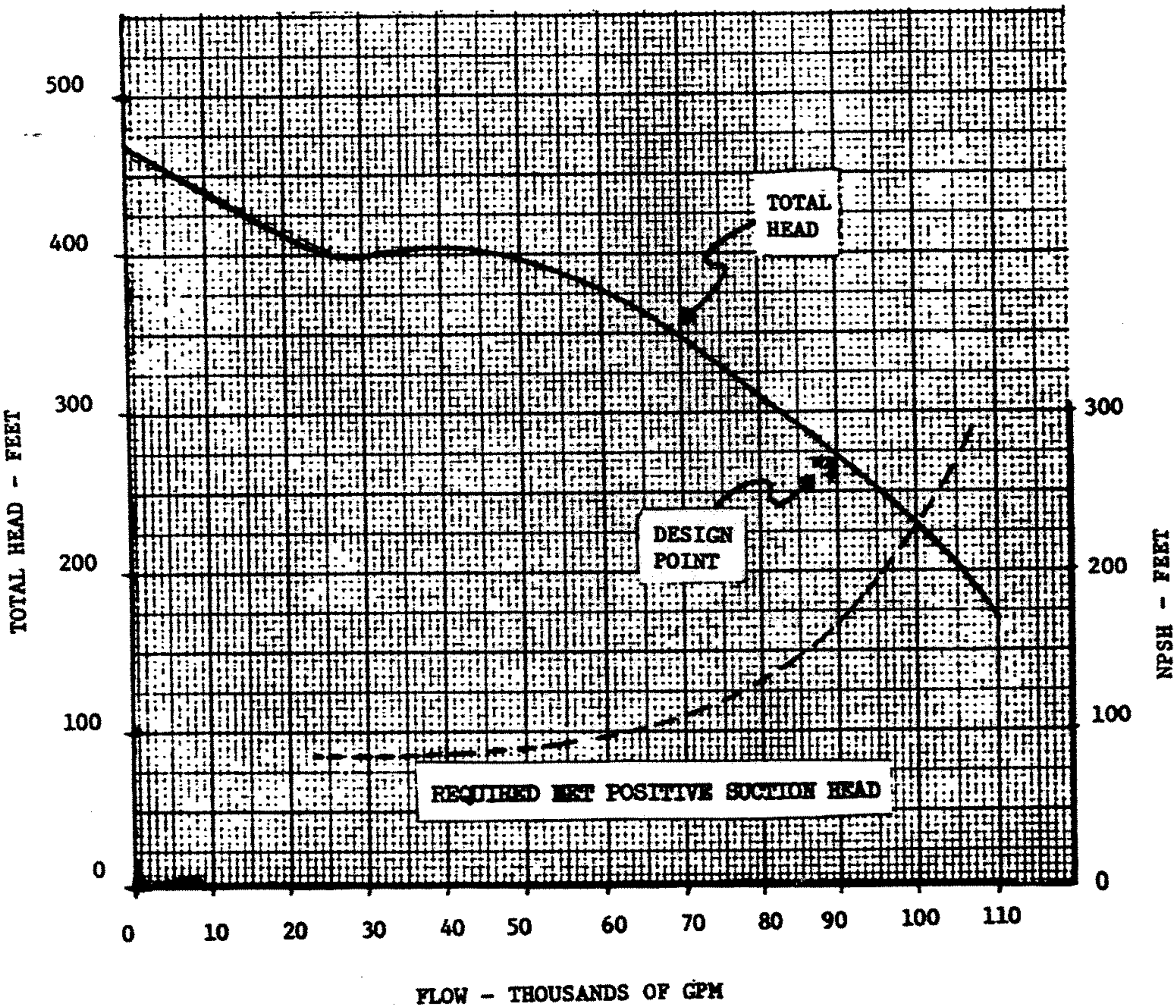
MODEL 44F



INDIAN POINT 3	FSAR UPDATE
STEAM GENERATOR	
REV. 1 JULY 1990	FIGURE NO 4.2-5



INDIAN POINT 3		FSAR UPDATE
REACTOR COOLANT PUMP		
REV 0	JULY, 1982	FIGURE NO 4.2-6

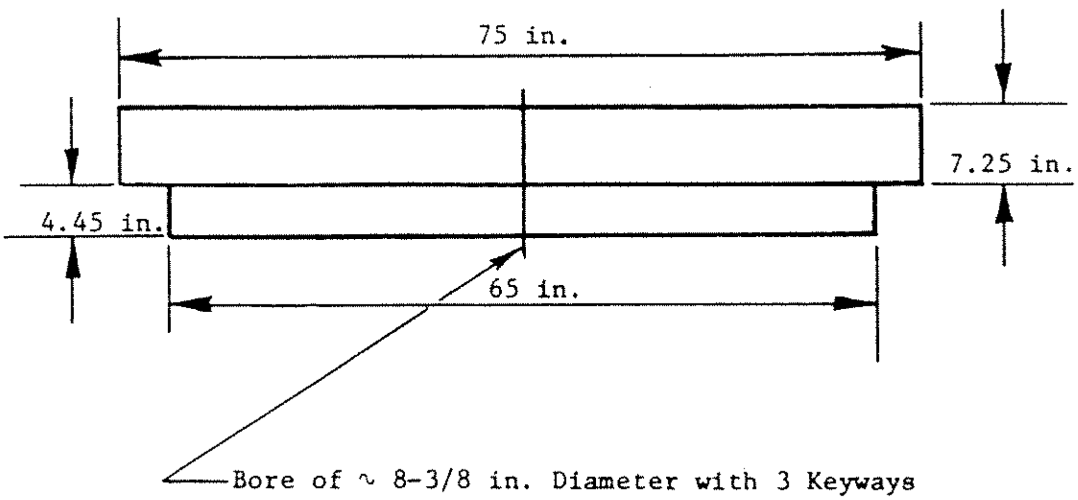


INDIAN POINT 3 FSAR UPDATE

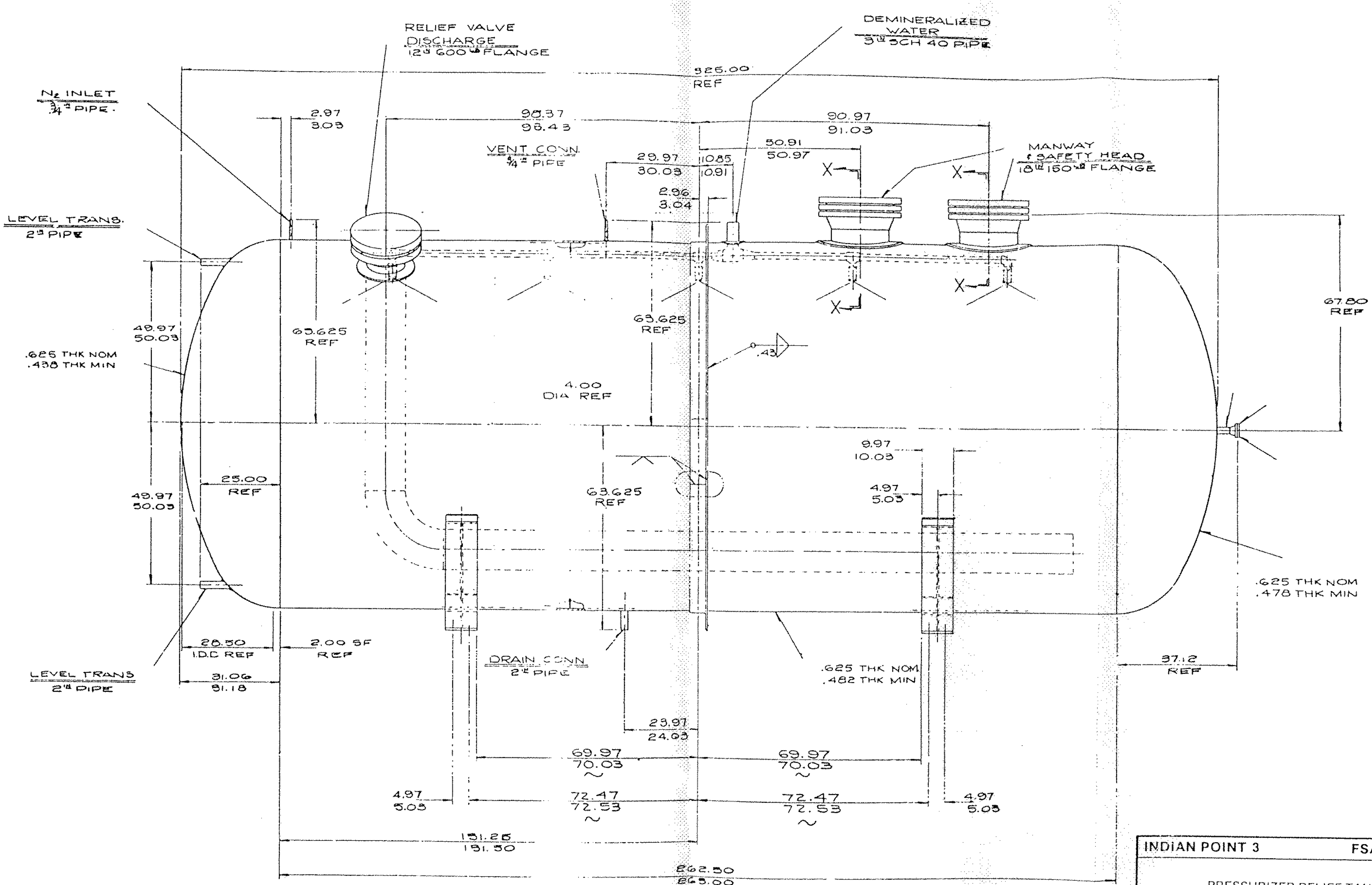
REACTOR COOLANT PUMP
ESTIMATED PERFORMANCE CHARACTERISTIC

REV. 1 JUN 2000 | FIGURE 4.2-7

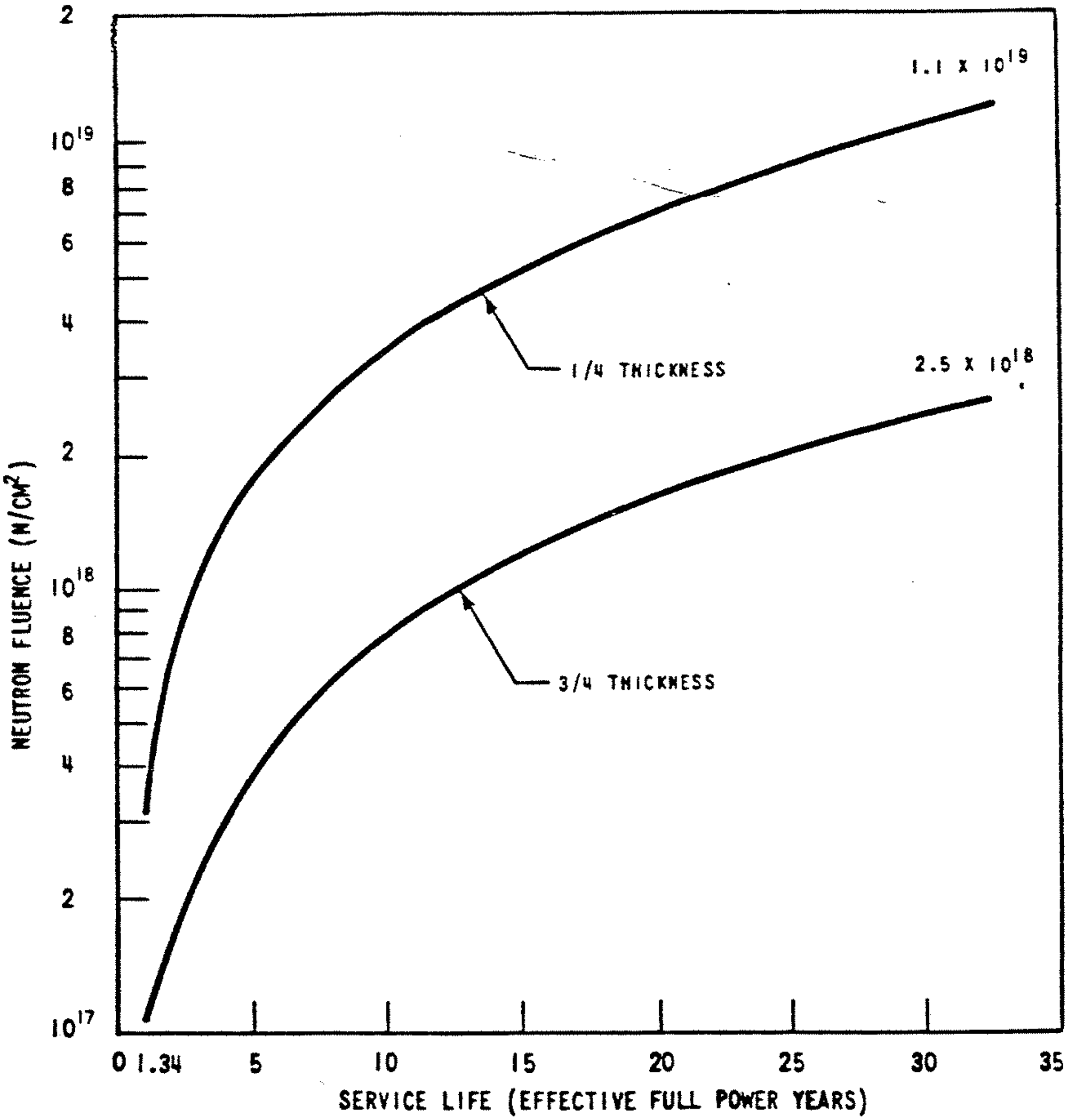
FLYWHEEL



NOTE: The plates are bolted together with the bolts aligned perpendicular to the planes of the plates

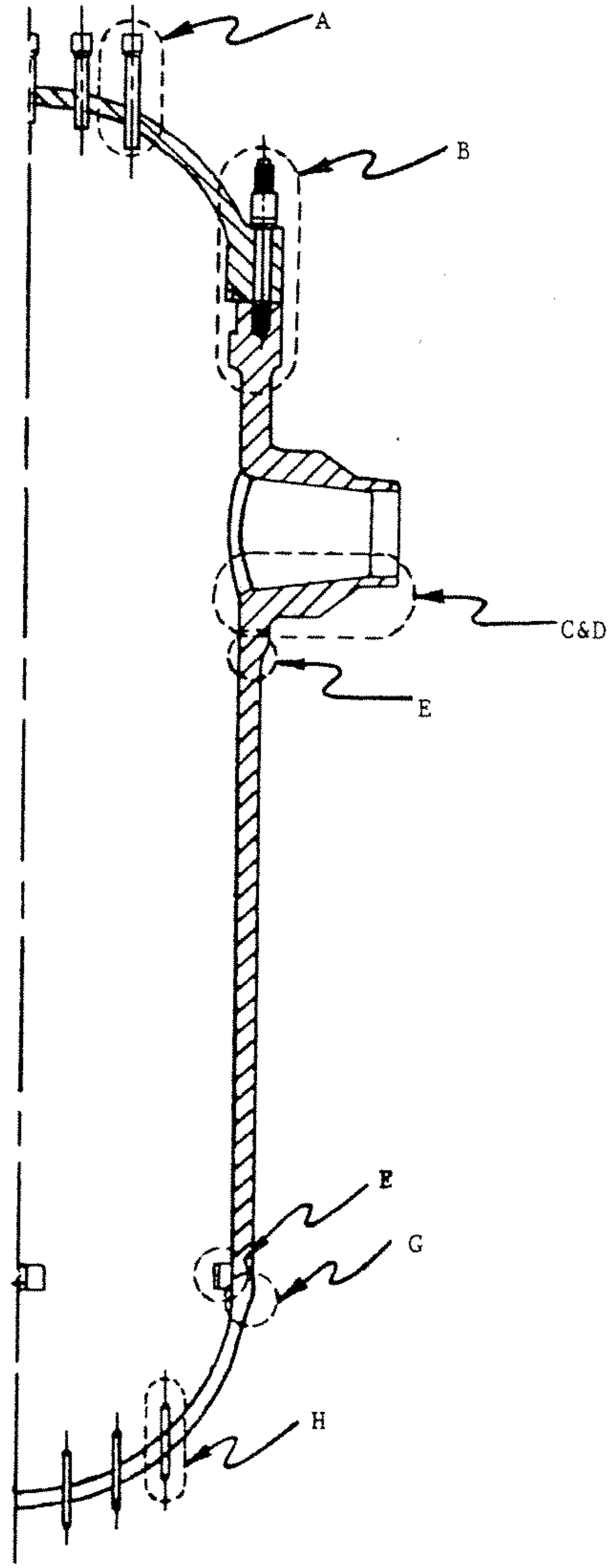


INDIAN POINT 3		FSAR UPDATE
PRESSURIZER RELIEF TANK		
REV. 0	JULY, 1982	FIGURE NO. 4.2-9

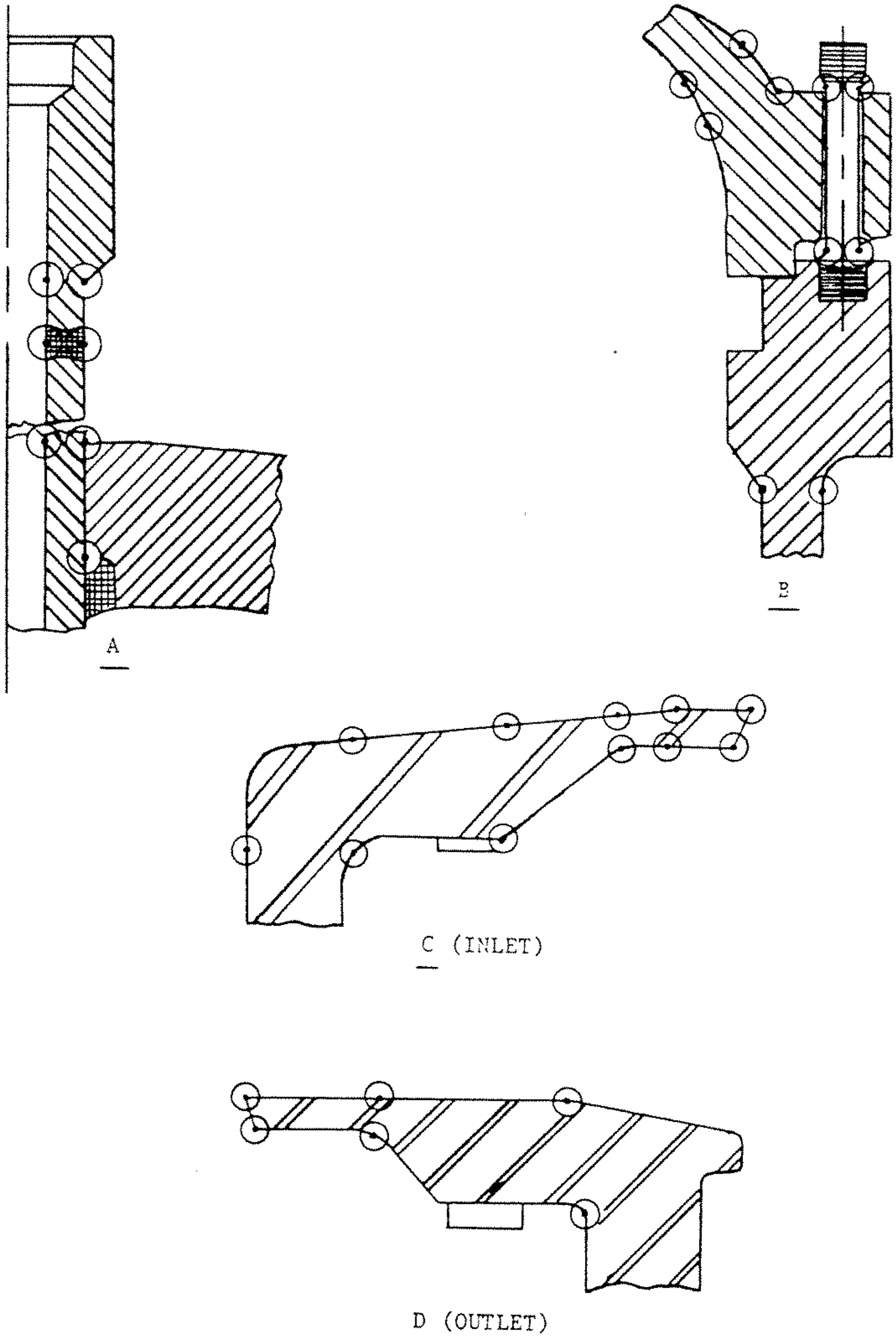


NOTE:
THIS FIGURE DOES NOT TAKE INTO ACCOUNT THE CONSERVATIVE EFFECTS OF LOW-LEAKAGE CORE DESIGN THAT WERE IMPLEMENTED IN AND BEYOND CYCLE 6.

INDIAN POINT 3 FSAR UPDATE	
FAST NEUTRON FLUENCE (E > 1MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE	
REV. 1 JUN. 1999	FIG. 4.2-10

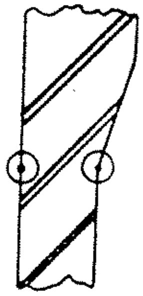


INDIAN POINT 3		FSAR UPDATE
REACTOR VESSEL LONGITUDINAL SECTION LOCATION RV ANALYSIS		
REV. 0	JULY, 1982	FIGURE NO. 4.3-1

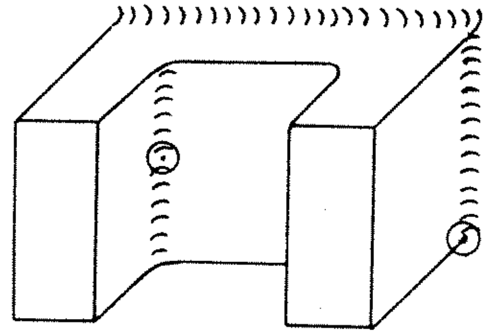


NOT TO SCALE

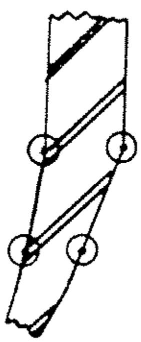
INDIAN POINT 3		FSAR UPDATE
LOCATION OF RV ANALYSIS UPPER VIEW		
REV. 0	JULY, 1982	FIGURE NO. 4.3-2



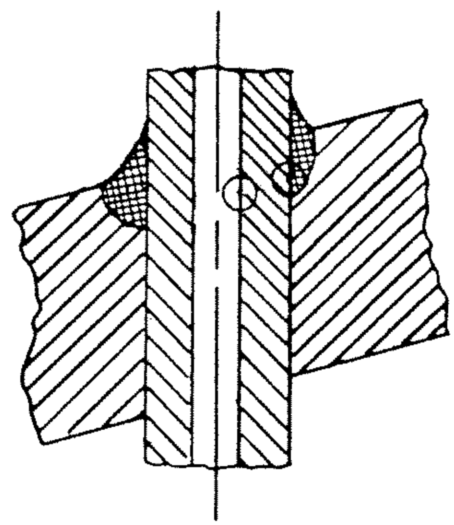
E



F



G

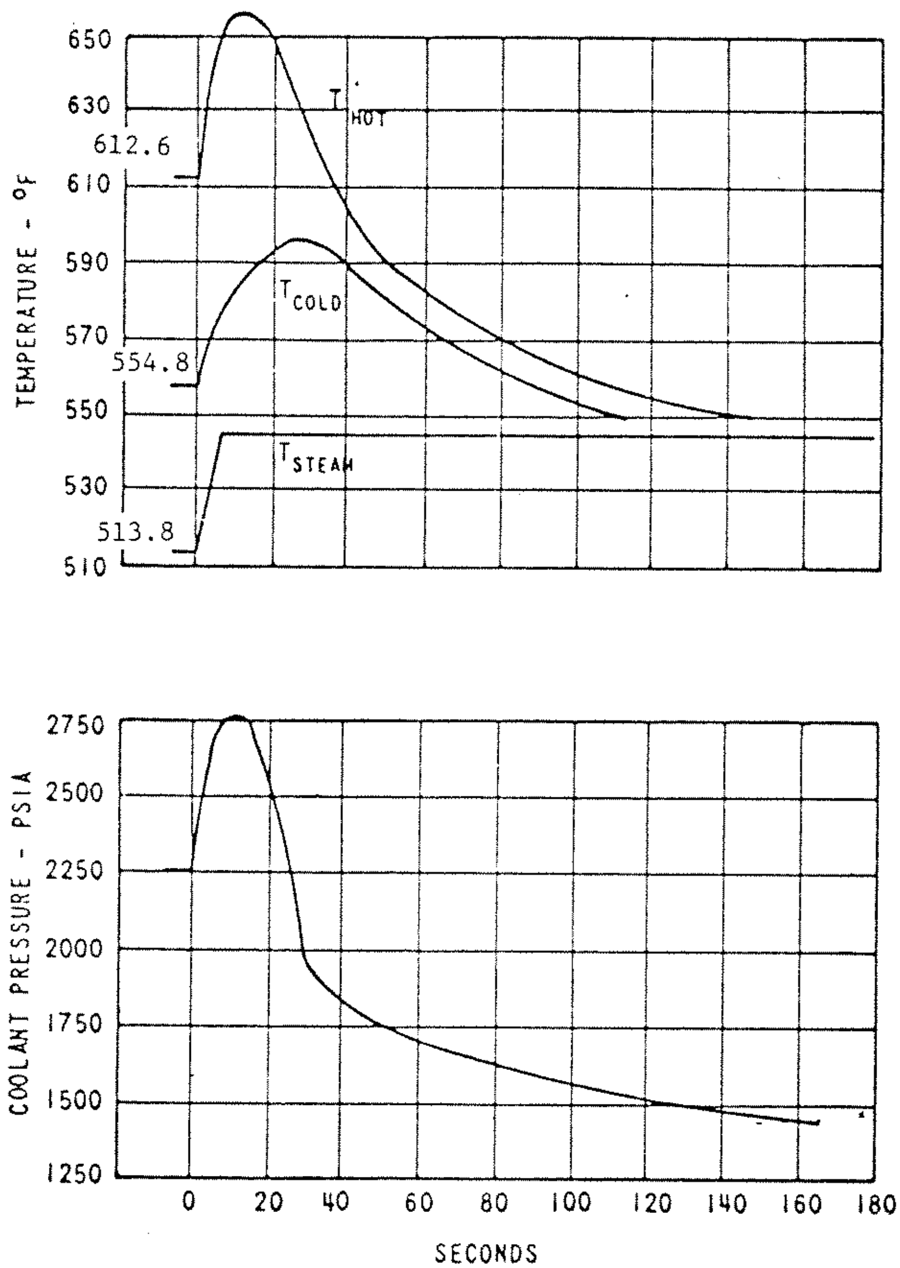


H

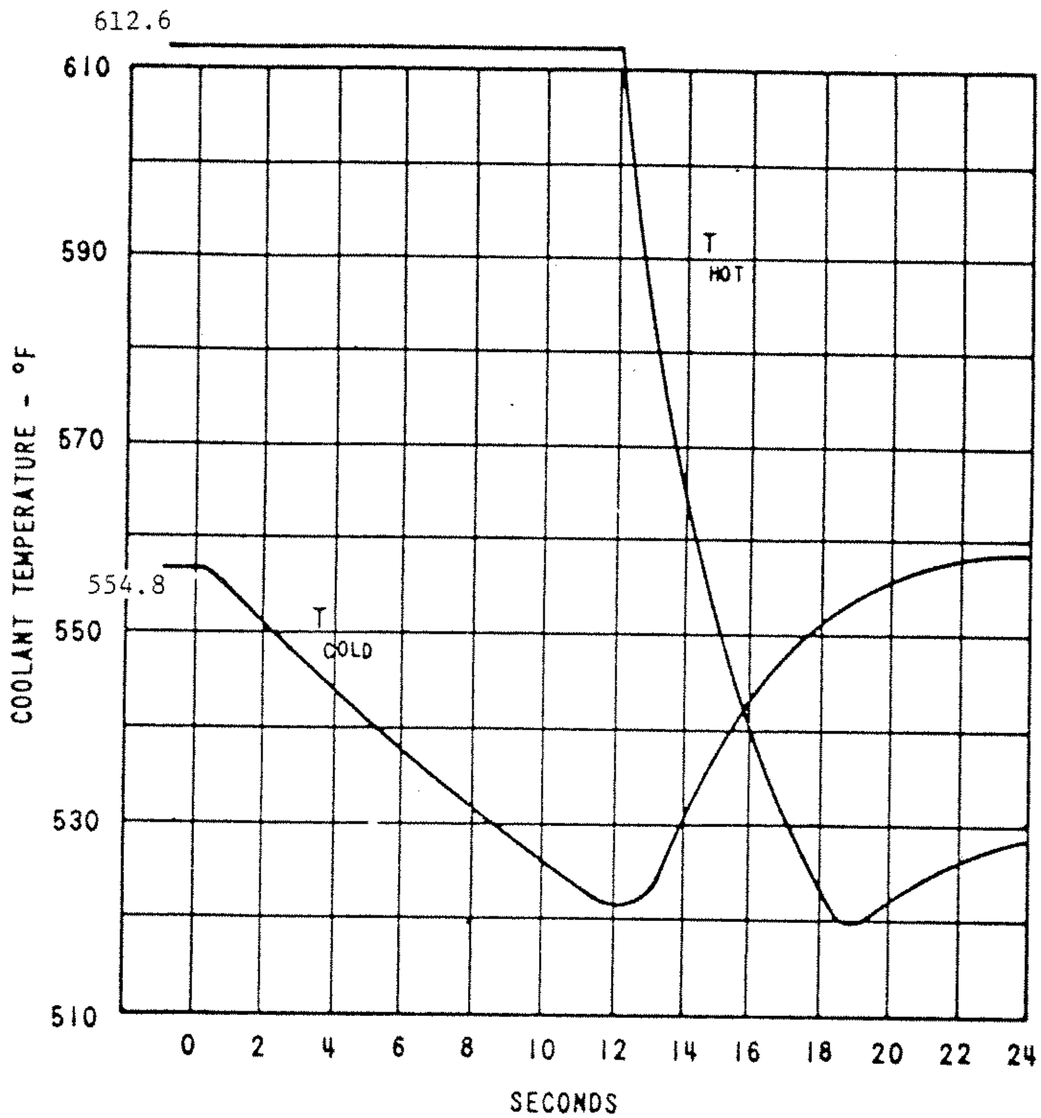
NOTE:

THE POINTS CIRCLED IN THE SKETCHES REPRESENT THE GENERAL LOCATION AND GEOMETRY OF THE AREAS OF DISCONTINUITY AND/OR STRESS CONCENTRATION.

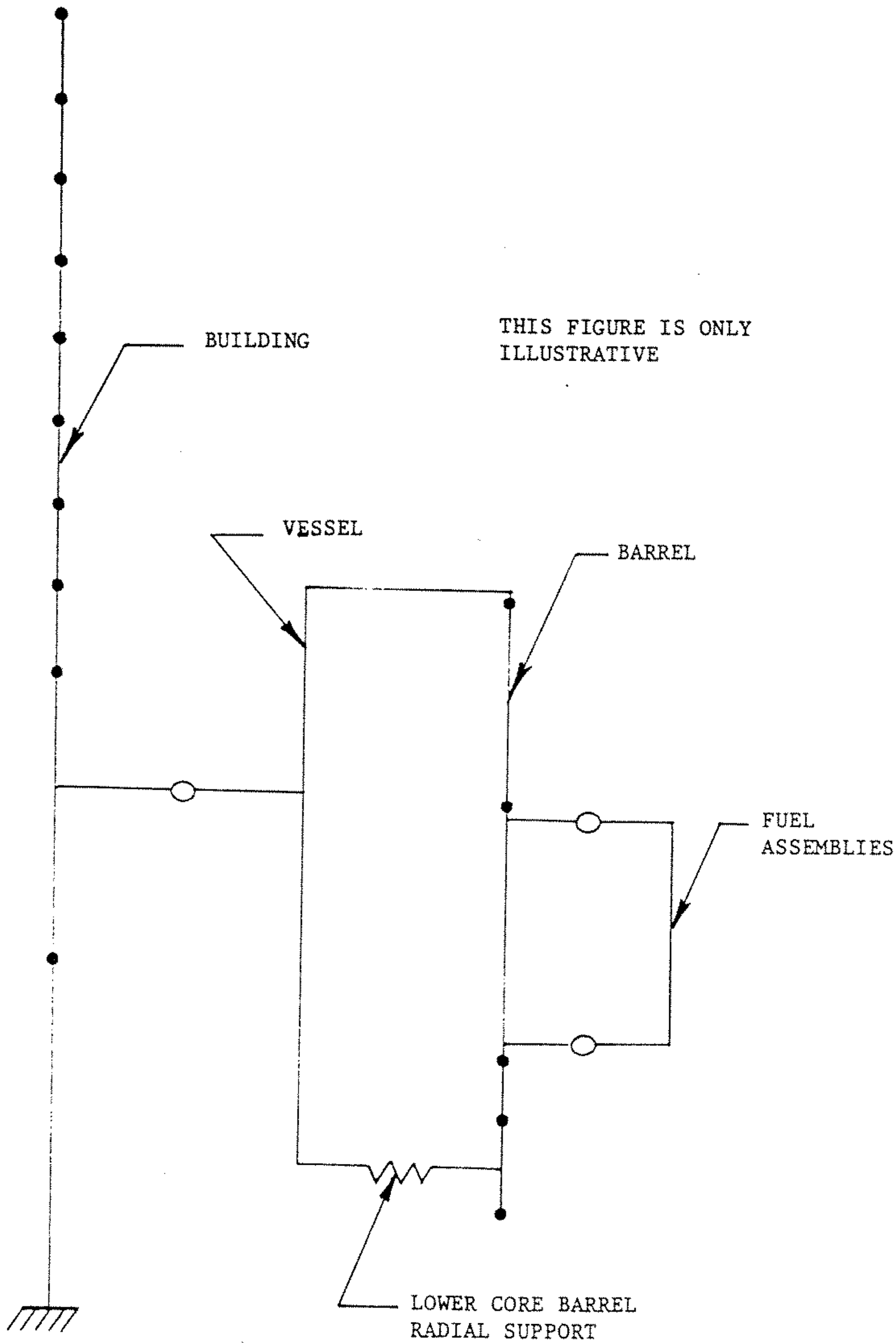
INDIAN POINT 3		FSAR UPDATE
LOCATION OF RV ANALYSIS LOWER VIEW		
REV. 0	JULY, 1982	FIGURE NO. 4.3-3



INDIAN POINT 3		FSAR UPDATE
LOSS OF LOAD TRANSIENT		
REV. 0	JULY, 1982	FIGURE NO. 4.3-4

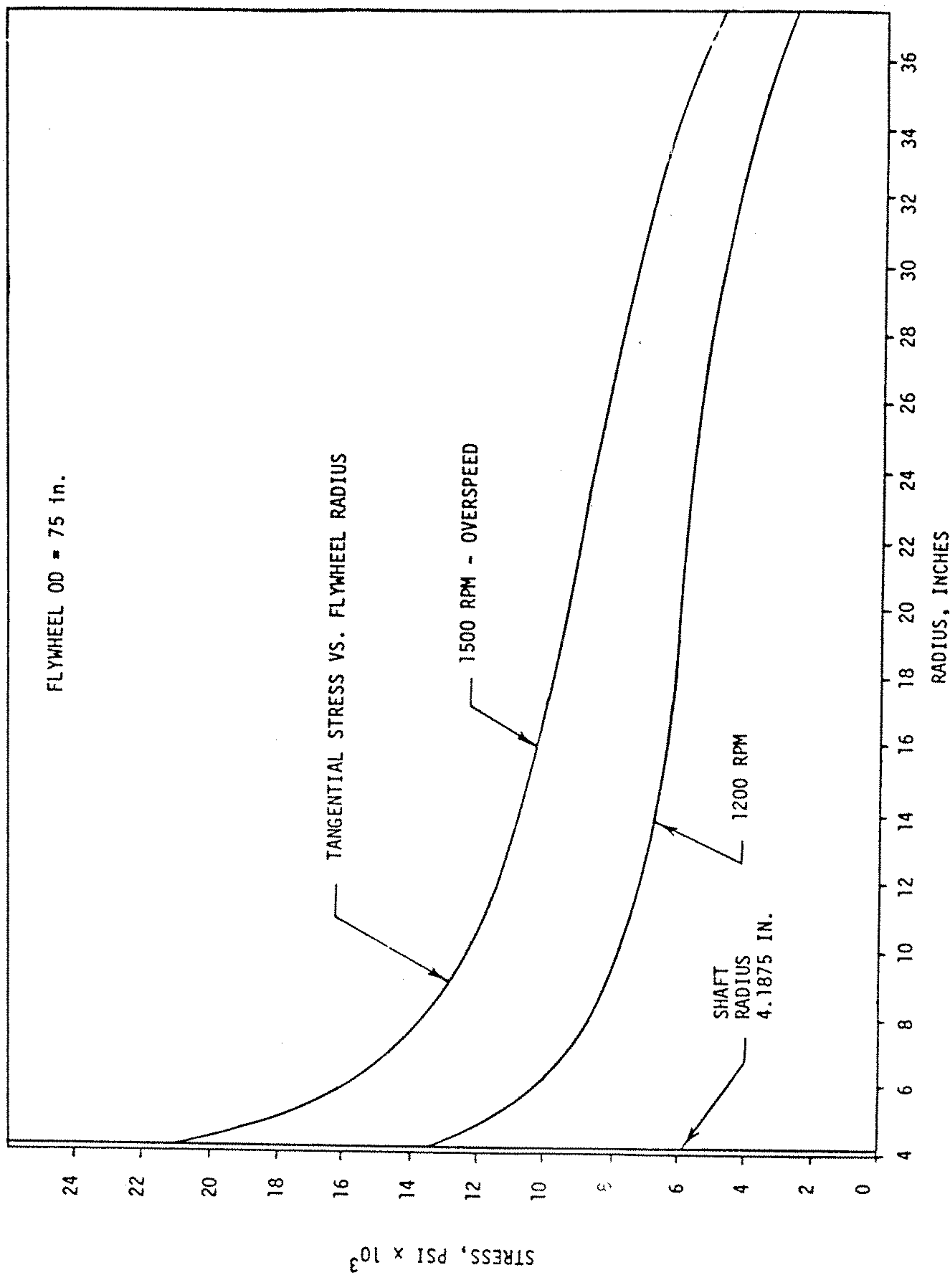


INDIAN POINT 3		FSAR UPDATE
LOSS OF FLOW TRANSIENT		
REV. 0	JULY, 1982	FIGURE NO. 4.3-5



NOT TO SCALE

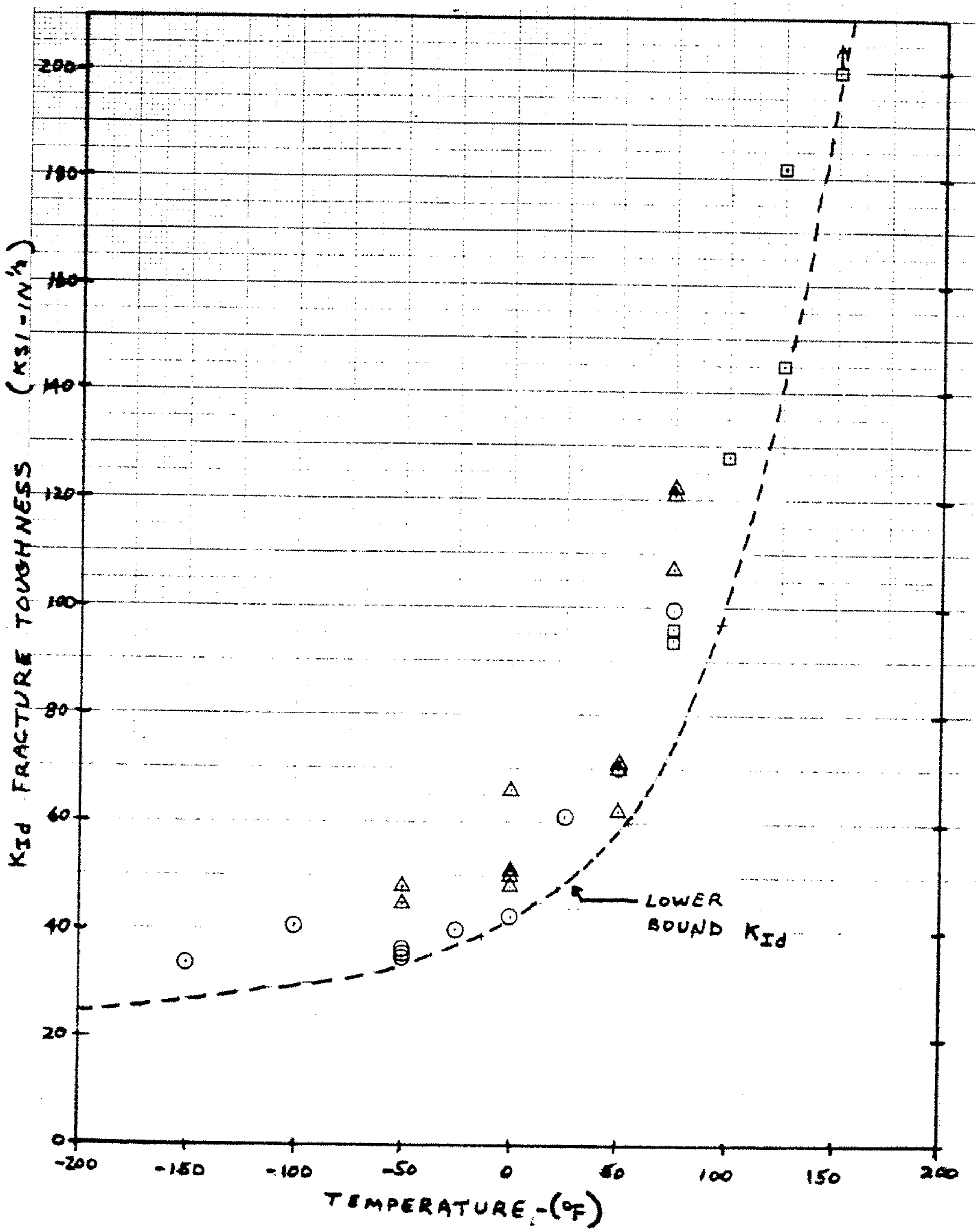
INDIAN POINT 3		FSAR UPDATE
MATHEMATICAL MODEL FOR REACTOR VESSEL INTERNALS ANALYSIS- HORIZONTAL EXCITATION		
REV. 0	JULY, 1982	FIGURE NO. 4.3-6



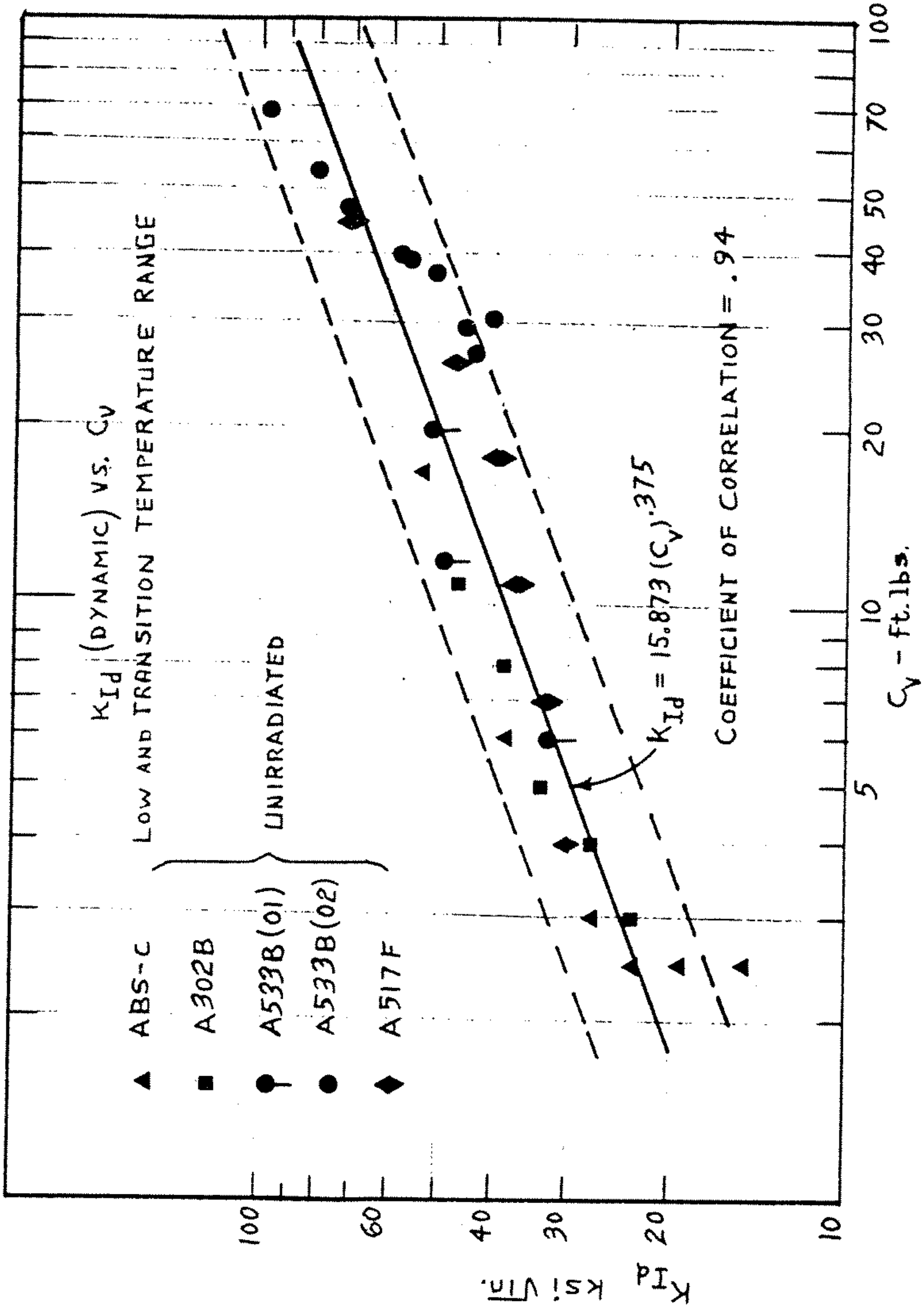
INDIAN POINT 3 **FSAR UPDATE**

FLYWHEEL CALCULATED STRESSES
AT OPERATING SPEED

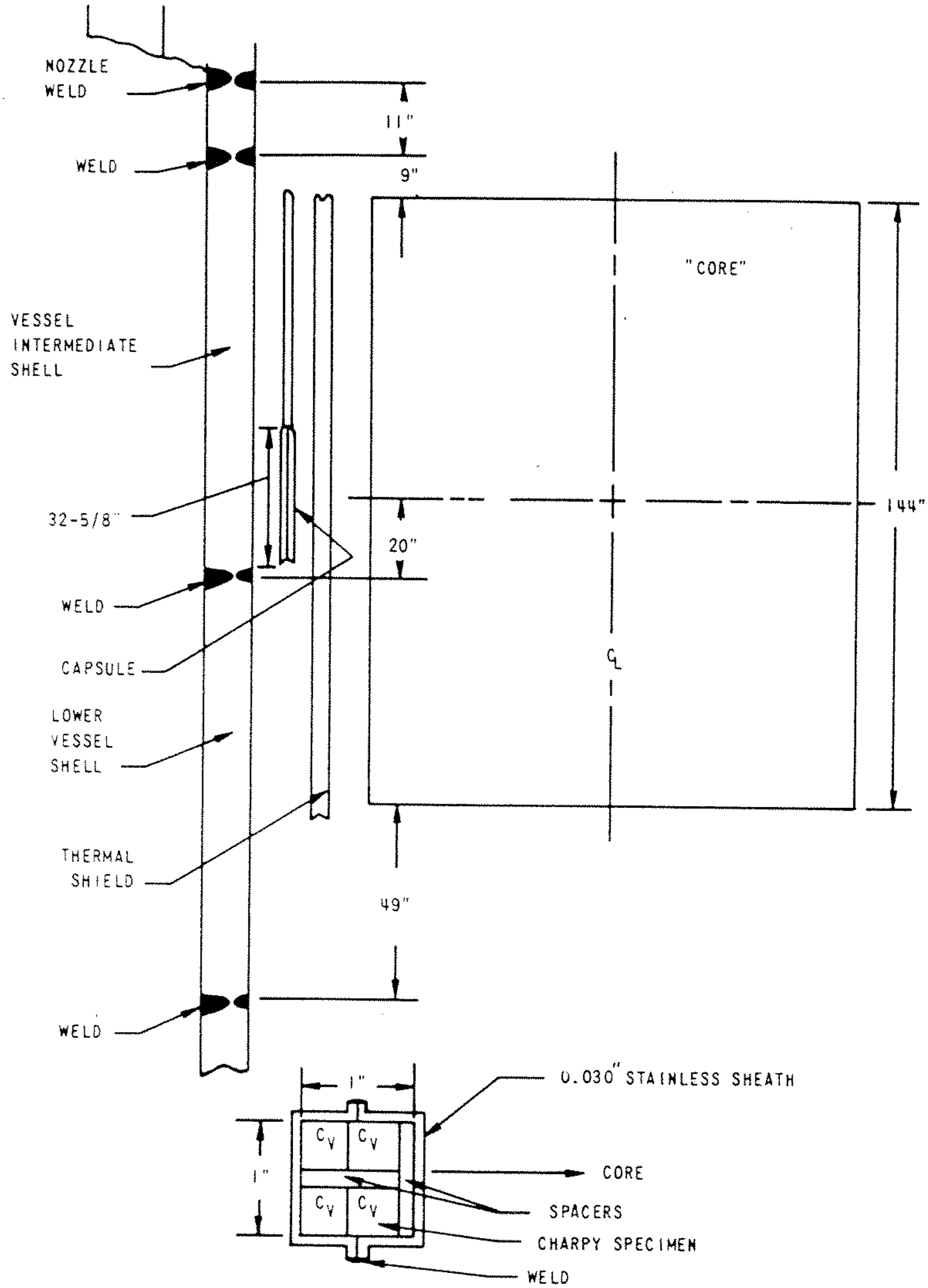
REV. 0 JULY, 1982 FIGURE NO 4.3-7



INDIAN POINT 3		FSAR UPDATE
K_{ID} LOWER BOUND FRACTURE TOUGHNESS A533 GRADE B CLASS 1		
REV. 0	JULY, 1982	FIGURE NO. 4.3-8



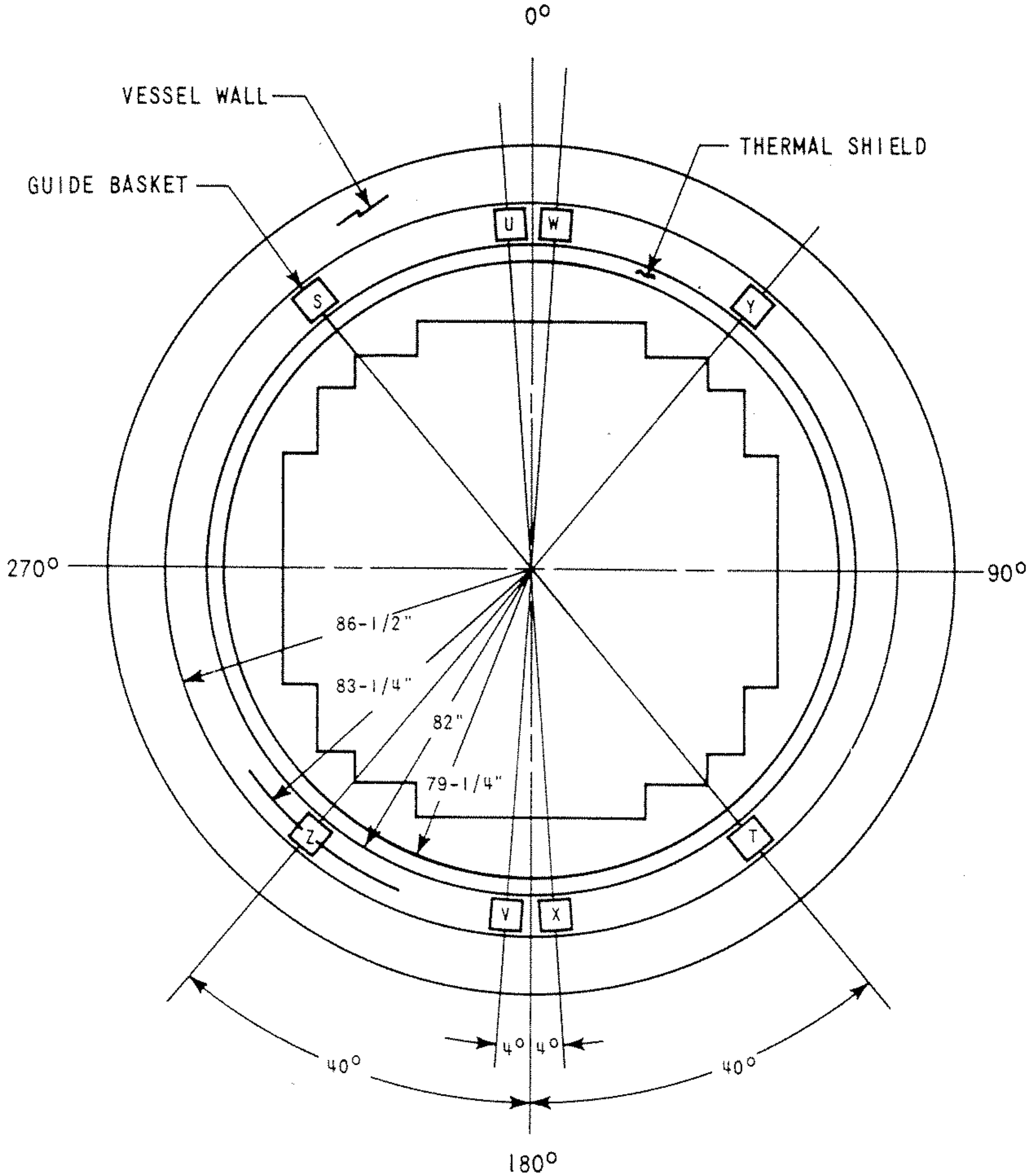
INDIAN POINT 3		FSAR UPDATE
CORTEN & SAILO CORRELATION[2]		
REV. 0	JULY, 1982	FIGURE NO. 4.3-9



INDIAN POINT 3 — FSAR UPDATE

TYPICAL SURVEILLANCE CAPSULE
ELEVATION VIEW

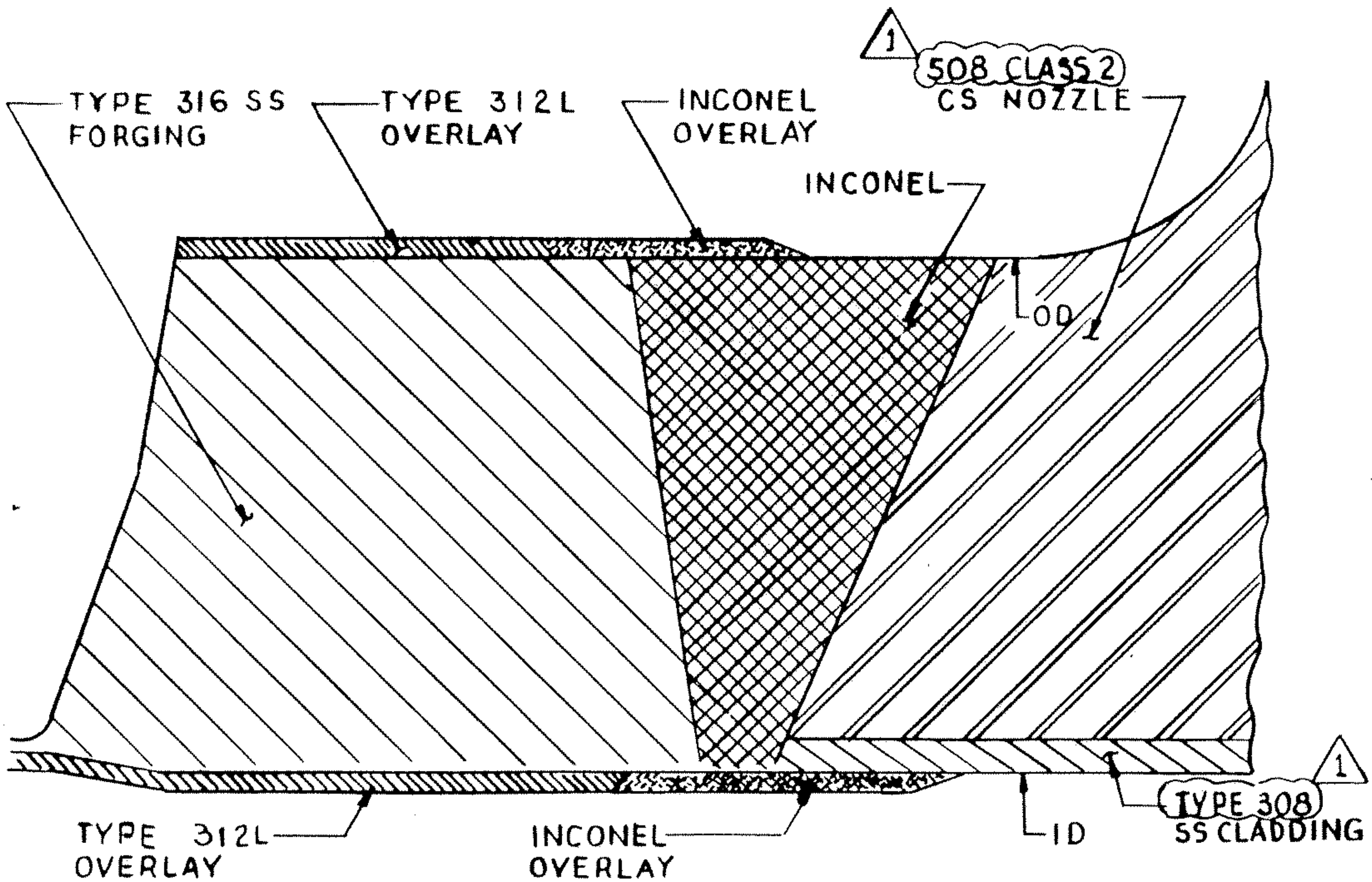
REV. 0 — JULY, 1982 — FIGURE NO. 4.5-1



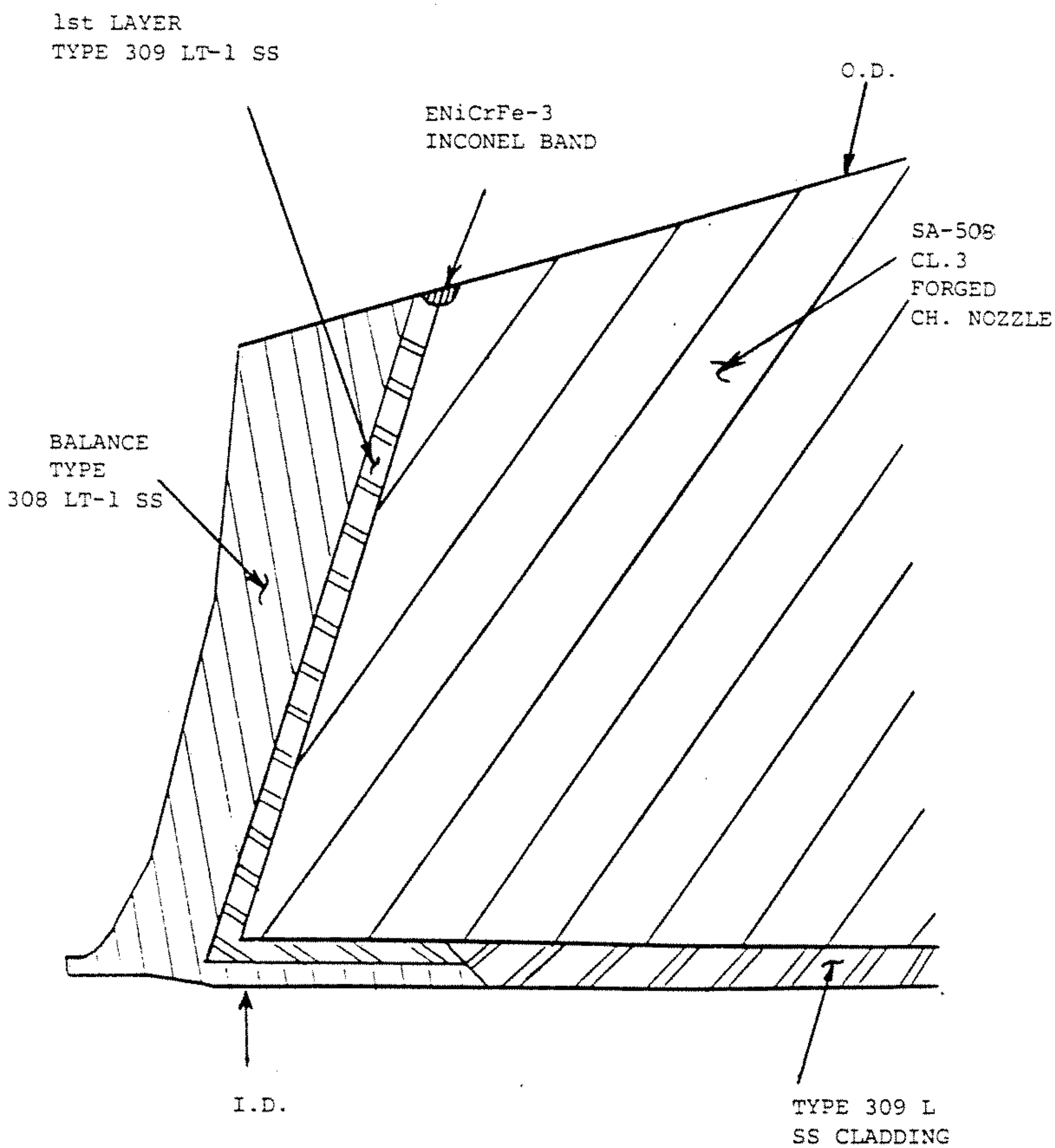
INDIAN POINT 3 FSAR UPDATE

SURVEILLANCE CAPSULE
PLAN VIEW

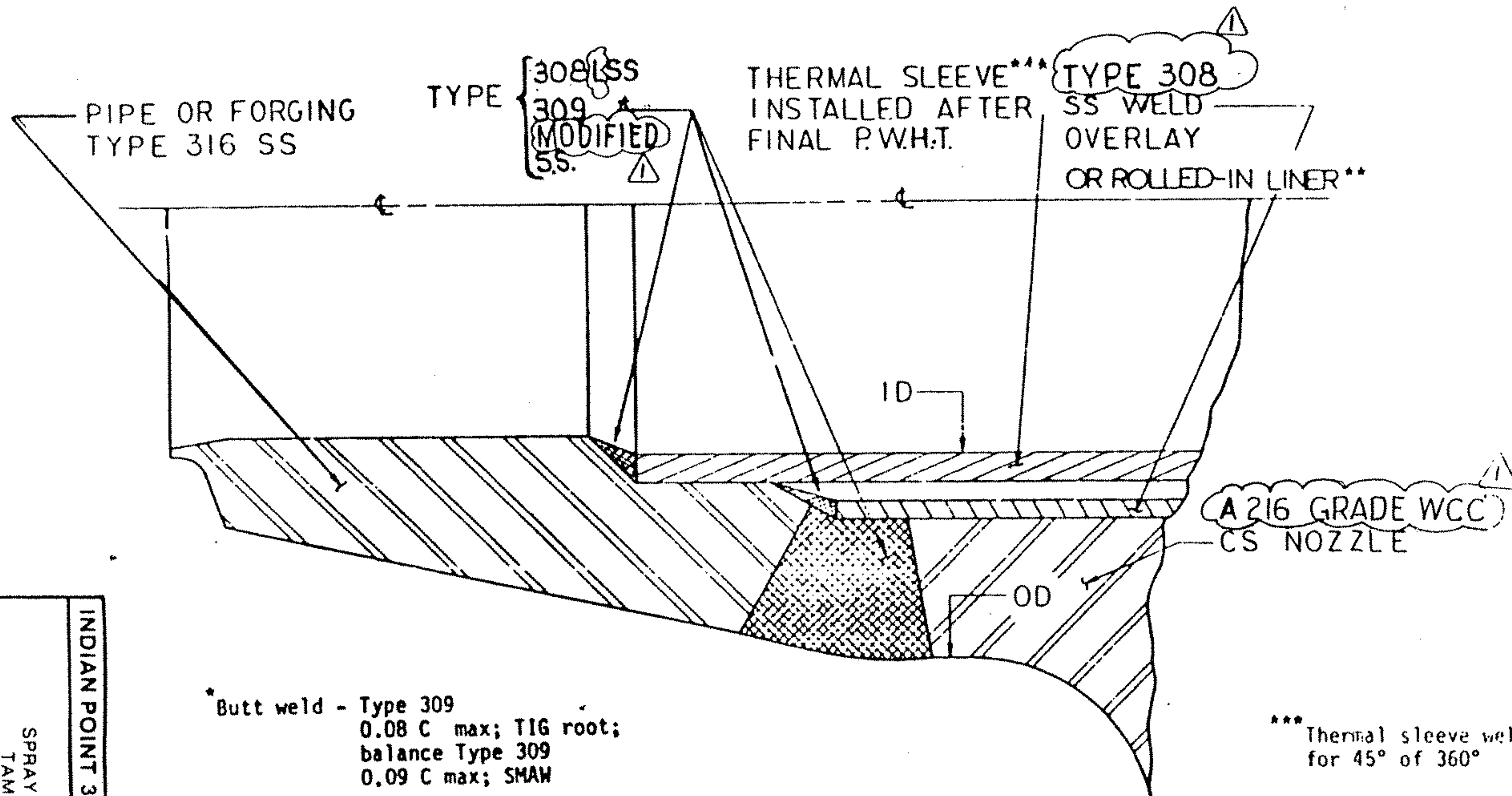
REV. 0 JULY, 1982 FIGURE NO. 4.5-2



INDIAN POINT 3	FSAR UPDATE
PRIMARY NOZZLE COMBUSTION ENGINEERING REACTOR VESSEL	
REV. 1 JULY 1988	FIGURE No. 4D-1



INDIAN POINT 3	FSAR UPDATE
PRIMARY NOZZLE STEAM GENERATOR	
REV. 2 JULY 1990	FIGURE NO. 4D-2



*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 3	FSAR UPDATE
SPRAY OR SURGE NOZZLE TAMPA PRESSURIZER	
REV 1 JULY 1988	FIGURE NO. 4D-3

CHAPTER 5

CONTAINMENT

5.1 CONTAINMENT SYSTEM 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 were designed to prevent any leakage through the containment. The structure provides biological shielding for both normal and accident situations.

The Reactor Containment was designed to safely withstand several conditions of loading and their credible combinations. The major loading conditions were:

a) Occurrence of a gross failure of the Reactor Coolant System which creates a high pressure and temperature condition within the containment.
b) Coincident failure of the Reactor Coolant System with an earthquake or wind.

5.1.1.1 Principal Design Criteria

The General Design Criteria presented and discussed in this section are those which were in effect at the time when Indian Point 3 was designed and constructed. These general design criteria, which formed the basis for the Indian Point 3 design, were published by the Atomic Energy Commission in the Federal Register of July 11, 1967, and subsequently made a part of 10 CFR 50.

The Authority has completed a study of compliance with 10 CFR Parts 20 and 50 in accordance with some of the provisions of the Commission's Confirmatory Order of February 11, 1980. The detailed results of the evaluation of compliance of Indian Point 3 with the General Design Criteria presently established by the Nuclear Regulatory Commission (NRC) in 10 CFR 50 Appendix A, were submitted to NRC on August 11, 1980, and approved by the Commission on January 19, 1982. These results are presented in Section 1.3.

Quality Standards

Criterion: Those systems and components of reactor facilities which are essential to the prevention, or the mitigation of 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 of 7/11/67)

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 conformed to the applicable provisions of recognized codes and good nuclear practice. The concrete structure of the reactor containment conformed to the applicable portions of ACI-318-63. Further elaboration on quality standards of the reactor containment is given in Section 5.1.1.5.

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 could 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 conditions, 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 of their suitability as a basis for design. (GDC 2 of 7/11/67)

All components and supporting structures of the Reactor Containment were designed so that there is no loss of function of such equipment in the event of design basis 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 system damping, was included in the design analysis.

The Reactor containment is defined as a seismic Class I structure for purposes of seismic design (Chapter 16). 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 design basis earthquake.

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. Non-combustible 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 of 7/11/67)

Fire protection in all areas of the nuclear electric plant is provided by structure and component design which optimizes the containment of combustible materials and maintains exposed combustible material below the ignition temperature. The station was designed on the basis of limiting the use of combustible materials in construction by using fire resistant materials to the greatest extent practical. The Reactor Containment System was designed to maintain its capability in case of fire to safely shut down and isolate the reactor. Since containment recirculation ventilation charcoal filters are required, special manually-actuated sprays are installed which are operable from the Control Room. Containment liner thermal insulation does not support combustion. Fire headers are provided inside Containment and the Reactor Coolant Pumps are provided with an oil collection system. For additional details on fire protection features, see Section 9.6.2.

Records Requirements

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 of 7/11/67)

Records of the design, fabrication, construction and testing of the reactor containment are maintained throughout the life of the reactor.

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 along 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. (GCD 10 to 7/11/67)

The design pressure and temperature of the Containment exceed 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 reactor coolant pipe. Energy contribution from the steam system was 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 were 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.

The containment structure and all penetrations were designed to withstand, within design limits, the combined loadings of the Design Basis Accident and design basis seismic conditions.

All piping systems which penetrate the vapor barrier are anchored at the liner. The penetrations for the main steam, feedwater, blowdown and sample lines were designed so that the penetration is stronger than the piping system and that the vapor barrier is not breached due to a hypothesized pipe rupture combined, for the case of the steam line, with the coincident internal pressure. The pipe capacity in the flexure was assumed to be limited to the plastic moment capacity based upon the ultimate strength of the pipe material. All lines, with the exception of small bore lines, 2" and smaller connected to the Primary Coolant System that penetrate the vapor barrier were also anchored at or within 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 anchor and the Reactor Coolant System. These anchors were 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 basis seismic conditions.

Appendix 4B 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: the latter utilizing only the emergency electric power supply.

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 of 7/11/67)

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

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

b) In considering post-accident pressure effects, various malfunctions of the emergency systems were evaluated. Contingent mechanical or electrical failures were 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.

c) 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 below:

Discharge of reactor coolant through a double-ended rupture of the main loop piping, followed by operation of only these engineered safety features which can run simultaneously with power from two of the three on-site diesel generators (two high head safety injection pumps, one recirculation pump, three fan cooler units, one spray pump), results in a sufficiently low radioactive materials leakage from the containment structure that there is not undue risk to the health and safety of the public.

NDTT Requirement for Containment Material

Criterion: The selection and use of containment materials shall be in accordance with applicable engineering codes. (GDC 50 of 7/11/67)

The selection and use of containment materials compiled 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 normal operation is between 50°F and 130°F. This includes both hot operating and cold shutdown conditions. The minimum service metal temperature of the containment liner is well above the NDT temperature of +30°F for the liner material. The Equipment Hatch, penetration sleeves and Personnel Lock meet the Charpy V-notch impact values for a minimum of 15 ft-lbs at -50°F. Penetration "SS" end plates were replaced in 1997 and the Charpy V-notch impact values were determined.

5.1.1.2 Supplementary Accident Criteria

Systems relied upon to operate under post-accident conditions, which are located external to the containment and were considered to be extensions of the leakage boundary.

The pressure retaining components of the containment structure were 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 was designed to ensure that no average strains greater than the strain at the guaranteed yield point occur at the factored loads.

2) The mild steel reinforcement was generally 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 pressure retaining components of containment subject to deterioration or corrosion in service were 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 design pressure and temperature on the containment structure are those created by the hypothetical Loss-of-Coolant Accident. The Reactor Coolant System contains approximately 512,000 lb of coolant at a weighted average enthalpy of 595 Btu/lb for a total energy of 304,000,000 Btu. In a hypothetical accident, this water is released through a double-ended break in the largest reactor coolant pipe, causing a rapid pressure rise in the containment. The reactor coolant pipe used in the accident is the 29 inch ID section because rupture of the 31 inch ID section requires that the blowdown go through both the 29 inch and the 27-1/2 inch ID pipes and would, therefore, result in a less severe transient.

Additional energy release was considered from the following sources:

a) Stored heat in the reactor core
b) Stored heat in the reactor vessel piping and other Reactor Coolant System components
c) Residual heat production
d) Limited metal-water reaction energy and resulting hydrogen-oxygen reaction energy

The following loadings were considered in the design of the containment in addition to the pressure and temperature conditions described above:

a) Structure dead load
b) Live loads
c) Equipment loads
d) Internal test pressure
e) Earthquake
f) Wind
g) Tornado

The capability of the Containment to withstand additional energy releases is discussed in Chapter 14.

5.1.1.4 Engineered Safety Features System Contributions

Five types of engineered safety features were included in the design of this facility to assure containment integrity. These systems are discussed in Chapter 6 and their effectiveness is analyzed in Chapter 14.

5.1.1.5 Codes and Classifications

The design, materials, fabrication, inspections, and proof testing of the containment vessel complies with the applicable parts of the following:

STRUCTURAL

	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, C1. 1	Specification for Notch Toughness Requirements for Normalized Steel Plates for Pressure Vessels
	Firebox A-201, Gr. B	Specification for Carbon Silicon Steel Plates of Intermediate Tensile Ranges for Fusion Welded Boilers and Other Pressure Vessels

4. ASTM A-36, Gr. C	Specification for Structural Steel
5. ASTM A-131, Gr. C	Specification for Structural Steel for Ships
6. ASTM A-240	Specification for Chromium and Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Fusion-Welded Unfired Pressure Vessels
7. ASTM A-312	Specification for Seamless and Welded Austenitic Stainless Steel Pipe
8. ASTM 442, Grade 60	Standard Specification for Carbon Steel Plates with Improved Transition Properties
9. ASME Boiler & Pressure Vessel Code-Section III	Nuclear Vessels
10. ASME Boiler & Pressure Vessel Code-Section VIII	Unfired Pressure Vessels
11. ASME Boiler & Pressure Vessel Code-Section IX	Welding Qualifications
12. ASTM C-33	Standard Specifications for Concrete Aggregates
13. ASTM C-150	Standard Specifications for Portland Cement
14. ASTM C-172	Method of Sampling Fresh Concrete
15. ASTM C-31	Method of Making and Curing Concrete Compression and Flexure Test Specimen in Field
16. ASTM C-39	Method of Test for Compressive Strength of Molded Concrete Cylinders
17. ASTM C-350	Specification for Fly Ash for Use as an Admixture in Portland Cement Concrete
18. ASTM C-94	Recommended Practice for Winter Concreting
19. ASTM C-42	Methods of Securing, Preparing, and Testing Specimens from Hardened Concrete for Compressive and Flexural Strengths
20. ASTM C-494	Specifications for Chemical Admixtures for Concrete

21.	ASTM A-305	Specifications for Miminum Requirements for Deformation of Deformed Bars for Concrete Reinforcement
22.	ASTM A-408	Specifications for Special Large Size Deformed Billet-Steel Bars for Concrete Reinforcement
23.	ASTM A-432	Specification for Deformed Billet Steel Bars for Concrete Reinforcement with 60,000 psi Minimum Yield Strength
24.	Research Council of Reveted & Bolted Structural Joints of the Engineering Foundation	Specification for Structural Joints Using ASTM A-325 Bolts
25.	ACI-613	Recommended Practice for Selecting Proportions for Concrete
26.	ACI-306	Recommended Practice for Winter Concreting
27.	ACI-318, Part IV-B	Structural Analysis and Proportioning of Members Ultimate Strength Design
28.	ACI-318	Building Code Requirements for Reinforced Concrete
29.	ACI-505	Reinforced Concrete Chimney Design
30.	ACI-315	Manual of Standard Practice for Detailing Reinforced Concrete Structures
31.	ASME Nuclear Vessels Code	---
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 Building and Other Structures
34.	--	State Building and Construction Code for the State of New York
35.	SSPC-SP-6	Commercial Blast Cleaning

36. ASME Boiler & Pressure Vessel Code-Section XI — Rules for in service inspection of Nuclear Power Plant Components.

5.1.2 Containment System 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 ¼ inch is attached to the inside face of the concrete to insure a high degree of leak-tightness. The design objective of the containment structure was 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 Figure 5.1-1 and Plant Drawings 9321-F-25013, -25023, -25033, -25063, -25073, and -25083 [Formerly Figures 5.1-2 through 5.1-7], consists of side walls measuring 148 feet from the liner on the base to the springline of the dome, and has an inside diameter of 135 feet. The side walls of the cylinder and the dome are 4'-6" and 3'-6" 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 flat concrete base mat is 9 feet thick with the bottom liner plate located on top of this mat. The bottom liner plate is covered with 3 feet 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 until it is dissipated.

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 or tornado.

In addition, keying action from the reactor pit and sumps, plus friction between the concrete and rock, prevents sliding of the structure from horizontal ground motion.

The basic structural elements that were 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 reinforcing in the structure exhibits a total elastic response to all primary loads. The lower portion of the cylindrical liner is 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. 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 2'-8" thick floor slab.

A 3-foot 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-foot 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 pipings penetrate the floor. Spiral and scissor stairs provide access to the areas below the floor. There is a reinforced concrete missile shield wall around the pressurizer above the operating floor. The original design is to protect the containment steel liner from postulated valve piece or instrument missiles connected to the pressurizer. Currently these missiles have been shown not to be credible.

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 feet of concrete. The floor is 4-feet thick. The concrete walls and floor are lined with ¼-inch thick stainless steel plate. The linings provide a leakproof membrane that is resistant to abrasion and damage during fuel handling operation.

A sub-surface drainage system is provided around the Containment Building where the mat is below grade as shown in Figure 5.1-11. Since the containment is above the water table, no hydrostatic seepage will occur.

The detailed structural design and analysis of the Containment System is presented in Appendix 5A. See also Sections 16.1 and 16.4 for seismic analysis.

5.1.2.2 <u>General Design Criteria</u>

The following loads were considered to act upon the containment structure creating stresses within the component parts:

a) Dead load consisted of the weight of the concrete wall, dome, liner insulation, base slab and the internal concrete.

b) Live load consisted of snow and construction loads on the dome and major components of equipment in the containment. Snow and ice loads were assumed to be applied uniformly to the top surface of the dome. A construction live load of 50 pounds per square foot was used on the dome, but was not considered to act concurrently with the snow load. Equipment loads were considered as specified on the drawings supplied by the manufacturers of the various pieces of equipment.

c) The internal pressure transient used for the containment design and its variation with time is shown on the pressure-temperature transient curve, Figure 5.1-8. For the free volume of 2,610,000 cubic feet 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 Chapter 14.

d) Thermal expansion stresses due to internal temperature increase caused by a Loss-of-Coolant Accident were considered. This temperature and its variation with time are shown on the pressure-temperature transient curve, Figure 5.1-8. 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, the corresponding liner temperatures will be 285°F and 306°F respectively. The pressure temperature transient curves for these loading conditions are shown in Figures 5.1-9 and 5.1-10, respectively. The maximum operating temperature is 130°F. The design 24-hour mean-low ambient temperature is -5°F.

e) The ground acceleration for the operational basis earthquake was determined to be 0.1g applied horizontally and 0.05g applied vertically. These values were resolved as conservative numbers based upon recommendations from Dr. Lynch, then Director of Seismic Observatory, Fordham University.

   A dynamic analysis was used to arrive at equivalent design loads. Additionally, a design basis earthquake ground acceleration of 0.15 horizontal and 0.10 vertical was used to analyze for the no-loss of function. This is discussed in Section 5.1.3.5, Seismic Design Summary.

f) The American Standards Association "American Standard Code Requirements for Minimum Design Loads in Buildings and Other Structures" (A58.1-1955) designated the site as being in a 25 psf zone. In this code, for height zones between 100 and 499 feet, the recommended wind pressure on a flat surface was 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 stipulated a wind pressure up to 30 psf on a flat surface for heights up to 600 feet. For design, a 30 psf basic wind load was used from ground level up.

g) Internal pressure was applied to test the structural integrity of the vessel up to 115 percent of the design pressure. For this structure, the test pressure was 54 psig.

h) Tornado loads consisted of 300 mph tangential wind traveling with a forward velocity of 60 mph. Also considered as a separate and as a combined loading combination was a 3.0 psi pressure drop external to the structure. In addition, horizontal and vertical missile loads were considered as specified in Section 2.1.5 of Appendix 5A.

5.1.2.3 Material Specifications

Basically, four materials were used for the construction of the containment vessel. These are:

a) Concrete
b) Reinforcing Steel
c) Plate Steel Liner
d) Insulation

Details of material properties, fabrication and erection requirements and material test results are presented in Appendix 5A, Section 5.0. Basic specifications for these materials were as follows:

a) Concrete is a dense, durable mixture of sound coarse aggregate, fine aggregate, cement and water. Aggregates conformed to American Society for Testing Materials Specification C-33 "Standard Specification for Concrete Aggregates." Aggregates consisted of inert materials that were clean, hard, durable, free from organic matter and uncoated with clay or dirt. Fine aggregate consisted of natural sand and the coarse aggregate of crushed stone. Fine aggregate tests performed include gradation, fineness modulus, specific gravity, unit weight, organic impurities, soundness (5 cycles Na_2SO_4) silt content and structural strength of sand in relation to Ottawa sand.

   Coarse aggregate tests included gradation, fineness modulus, specific gravity, unit weight, soundness (5 cycles Na_2SO_4), soft particles and Los Angeles abrasion test.

Portland cement conformed to American Society for Testing and Material Specification C-150-65. Manufacturer's mill test reports were required covering each silo of cement drawn for the project.

Water was free from any injurious amounts of chloride, acid, alkali, salts, oil, sediment or organic matter, and fit for drinking.

A testing laboratory tested materials for concrete, designed mixes, tested check mix at batch plant and jobsite, and tested job concrete test cylinders. Design mixes were checked by the laboratory, including adjustments to obtain a workable mix based on specification requirements, and verified by trial batches and laboratory test.

The only admixtures in the concrete were PLACEWELL and AIRECON, both products of Union Carbide. PLACEWELL is a liquid, water-reducing admixture and AIRECON is a liquid air-entraining admixture, both of which enhance the properties of plastic and hardened concrete. PLACEWELL conformed to all the requirements of ASTM C494 for a Type A water reducing admixture and contained no calcium chloride. AIRECON conformed to the requirements of ASTM C260 for Air Entraining Admixtures and also contained no calcium chloride. The mixing of concrete was done with a batch mixer of approved AGC type, or in ready-mix equipment conforming to ASTM Specification C94.

b) Reinforcing steel for the dome, cylindrical walls and base mat was 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" (Revised ASTM A615-68, Grade 60). This steel had 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-inch specimen. Reinforcing bars No. 11 and smaller in diameter were 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 were 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 was issued from the steel supplier. The splices used to join reinforcing bars were sample tested to assure that they will develop at least 125% of the minimum yield point stress of the bar. The test program required cutting out, at random, approximately 2 percent of the completed splices and testing to determine their breaking strength, thus confirming the strength of both the bars and the splice.

In the Containment, vertical rebar splices were staggered a minimum of 1'-2". Seismic Diagonal Bar splices were staggered 1'-2" vertically in each direction. In the dome a 2'-0" stagger pattern was used throughout for the Cadweld splices as well as the reinforcing splice plates, except for final closure pieces at the apex of the dome. Horizontal rebar splices were splices were spliced in elevation and in cross-section (bars or bar pairs) with 2'-4" nominal and 2'-0" minimum stagger.

The above requirements were generally satisfied during construction except in special cases where physical or layout problems occurred in isolated areas in the containment.

For all other seismic Class I structures other than the Containment, rebars were lap spliced in accordance with the requirements of ACI-318-63 "Building Code

Requirements for Reinforced Concrete." No specific stagger requirements were specified. In the Containment, mechanical splices were included because of the biaxial tensile stress conditions in the concrete, which eliminate bond and require continuous rebar, and the ACI-318 requirement that lapped splices in tension cannot be used for bars greater than No. 11.

c) 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 had a minimum yield strength of 32,000 psi and minimum tensile strength of 60,000 psi with an elongation of 22 percent in an 8-inch gauge length at failure.

   The liner is ¼-inch thick at the bottom, ½-inch thick in the first three courses and 3/8-inch thick for remaining portion of cylindrical walls except ¾-inch thick at penetrations and ½-inch thick in the dome. The liner material has been tested to assure an NDT temperature more than 30 F lower than the minimum operating temperature of the liner material.

   Impact testing was done in accordance with Section N331 of Section III of the ASME Boiler and Pressure Vessel Code. A 100 percent visual inspection of the liner anchors was made prior to pouring concrete.

d) The material for insulating the liner plate is urethane foam covered with gypsum board and a stainless steel jacket and backed with asbestos paper on the unexposed side. This insulation was selected to withstand the calculated temperature and pressure conditions associated with Figures 5.1-8, 5.1-9, 5.1-10.

e) Quality of both materials and construction of the containment vessel was assured by a continuous program of quality control and inspection. These components are considered ASME Section XI Class MC or CC components and any repair or replacement activities shall be performed in accordance with ASME Section XI Subsections IWE and IWL of the ASME Code, with certain exceptions whenever specific relief is granted by the NRC. The Quality Assurance Program (described in Entergy's Quality Assurance Program Manual) covers modification and maintenance activities.

5.1.2.4 Structural Design Criteria

The design was based upon limiting load factors which were used as the ratio by which loads will be multiplied for design purposes to assure that the loading deformation behavior of the structure is one of elastic, tolerable strain behavior.

The load factor approach was used in this design as a means of making a rational evaluation of the isolated factors which must be considered in assuring an adequate safety margin for the structure. This approach permitted 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 placed 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 presented in Appendix 5A, Section 2.1.0.

The load factors utilized in this design were 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 procedures, the load factors in the design primarily provide for a safety margin on the load assumptions. Specific combined load equations used in design are presented in Section 2.1.14 of Appendix 5A. The load factors were chosen in a conservative manner to insure that the structure as designed would respond to design loads with elastic strain behavior.

The primary load in the containment structure analysis is the accident pressure load. Since the formulas for analysis were based on thin shell considerations, resulting in tensile loads resisted by reinforcing, the same results would be obtained regardless of whether a load factor (ultimate) or reduced allowable stress (working) approach was used in design.

The load factors utilized plus the conservative assumptions used in analysis insured that the design for this portion of the structure was conservative. Conservative analysis assumptions included use of full concrete section for determining flexural rigidity thus drawing moment and shear to the stiffer base and use of only hoop steel as the spring constant which controls cylinder growth in the unrestrained area above the discontinuity.

Earthquake and wind loads were based on analysis with the Containment modeled as a cantilever beam. The loads are resisted by tension in rebar thus the same results would be obtained by working strength or ultimate strength design. Secondary, thermal loads were also carried by tension in the rebar and are thus independent of ultimate or working strength design.

Equilibrium checks can be easily made for the containment shell since thin shell membrane analysis was used in the design and only the rebar was assumed to carry the load. In addition, overturning moment for earthquake and tension caused by temperature were assumed carried by rebar only and thus easily checked for equilibrium.

In addition to the above analysis, non-membrane portions of the cylinder such as the Equipment Hatch opening were analyzed by a Finite Element Computer analysis. The results were checked to insure that internal stresses times resisting rebar area gave resultant forces equal to the applied external forces.

In areas of the Containment where tensile stresses in more than one direction occur, all stresses are carried by continuous mechanically spliced rebar. Therefore, stress limits of ACI-318-63 were applicable since concrete strength and rebar bond provided by concrete were not considered in design. The f'_C dependent factors were specified in the ACI-318-63 Code used only in the design of the base mat where a uni-axial stress condition exists and in the base of the Containment cylinder wall where hoop stresses are minimal. In the cylinder, radial shear forces are resisted only by rebar. Seismic shear forces are also resisted by rebar with no account taken of resistance offered by concrete.

All structural components were designed to have a capacity required by the most severe loading combination.

Thus, the design included the consideration of both primary and secondary stresses, and the load capacity in structural members was based on the ultimate strength values presented in Part IVB of ACI-318, as reduced by the capacity reduction factor "ϕ" which provided 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 "ϕ" was established as 0.95. The factor "ϕ" was 0.90 for flexure and 0.85 for diagonal tension, bond and anchorage.

For the liner steel the factor "ϕ" was 0.95 for tension. For compression and shear, the primary membrane liner stress was maintained below 0.95 yield and elastic stability was assured as a function of liner anchorage requirements.

The liner was designed to assure that no strains greater than the strain at the guaranteed yield point will occur at the factored loads except in regions of local stress concentrations or stresses due to secondary load effects, in which case the liner strain was limited to 0.5 percent. Sufficient anchorage was provided to assure elastic stability of the liner. The basic design concept utilized stud anchorage of the liner plate to the concrete structure which assures stud failure due to shear, tension or bending stress without the stud connection causing failure or tear of the liner plate. See References 1 and 2. The studs in the ½ inch plate were installed on 24" horizontal and 28" vertical grid and in the 3/8-inch plate on a 24" horizontal and 14" vertical grid. The design considered 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.

5.1.2.5 Missile Protection

High pressure Reactor Coolant System equipment is surrounded by the 3'-0" concrete shield wall enclosing the reactor coolant loop and pressurizer and by the 2'-0" concrete operating floor.

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 that might affect the engineered safeguards components were carefully checked against the possibility of being sources of missiles. The general criterion adopted was to make 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 were not to be sources of missiles had 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 was implemented:

Valves

Valves installed in the Nuclear Steam Supply System were evaluated for the probability of their stem becoming missiles. Valve stems are not considered credible missile since at least one feature (in addition to the steam threads) is included in their design that will prevent stem ejection. Valve steams with backseats are prevented from becoming missiles by the backseat feature. Also, valve plugs are secured and locked to the valve stems to prevent loosening in service. In addition, valve stems of valves with power actuators, such as air-operated or motor-operated valves are effectively restrained by the valve actuator. Valve stems of rotary motion valves, such as plug valves, ball valves, and butterfly valves, as well as diaphragm-type valves are not considered as credible missiles. This is because these valves do not have a large reservoir of pressurized fluid acting on the valve stem; therefore there is insufficient stored energy available to produce a missile.

Valves with nominal diameter larger than 2” were designed against bonnet-body connection failure and subsequent bonnet ejection by means of: (a) using the design practice of ASME Section VIII, ASME Section III and USAS B16.5 and (b) by controlling the load during the bonnet body connection stud tightening process.

Stud and nut material is ASTM A193-B7 and A194-2H, or approved equal. The proper stud torquing procedures and/or the use of a torque wrench with indication of the applied torque, control the stress of the stud to acceptable limits established by industry standards. The complete valves were hydro tested per USAS B16.5 (1500 lbs. USAS valves are hydro to 5400 psi). The cast stainless steel bodies and bonnets were radiographed and dye penetrant tested to verify soundness.

Valves with nominal diameter of 2" or smaller were forged and have screwed bonnet with canopy seal. The canopy seal is the pressure boundary while the bonnet threads were designed to withstand the hydrostatic end forces. The pressure containing parts were designed per criteria established by USAS B16.5 specification.

Valves with nominal diameter of 2” or smaller may be supplied with a screwed bonnet and canopy seal. The canopy seal is the pressure boundary while the bonnet threads were designed to withstand the hydrostatic end force. The pressure containing parts were designed per criteria established by the USAS B16.5 specification.

Reactor Coolant Pump Flywheel

The reactor coolant pump flywheel was 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 NDTT less than 10 F. The design results in a primary stress less than 50% of the material yield strength at operating speed. The flywheel was subjected to 100% volumetric ultrasonic inspections which are repeated at intervals during plant life. The finished machined bore was subjected to examination by approved method. The design overspeed of the pump is 125%. The maximum pump overspeed on loss of external load is 112%.

5.1.2.6 Protection From Long-Term Corrosion

Steel members embedded in reinforced concrete structures are protected from corrosion by concrete in accordance with normal code requirements and hence are not exposed to the atmosphere.

All other steel appurtenances are not main structural load carrying members and are visible and accessible for regular maintenance. These components are considered ASME Section XI Class MC or CC components and any repair or replacement activities shall be performed in accordance with ASME Section XI Subsections IWE and IWL of the ASME Code, with certain exceptions whenever specific relief is granted by the NRC. They are generally shop painted with red lead and finish painted with a standard finish paint.

5.1.3 Stress Analysis

5.1.3.1 General

The structural design of the Containment met the requirements established by 1961 edition of "The State Building and Construction Code for the State of New York" so far as these provisions were applicable. All concrete structures were designed, detailed and constructed in accordance with the provisions of "Building Code Requirements for Reinforced Concrete" (ACI-318-63) so far as these provisions were applicable. A detailed description of containment component design is presented in Appendix 5A, Section 4.0.

5.1.3.2 Method of Analysis

Basically three separate structural components were analyzed, each in equilibrium with loads applied to it and with constraints occurring at the juncture of the structures. The three components were:

a) The 135-ft ID hemispherical dome
b) The 135-Ft ID Cylinder
c) The base slab.

Mathematically, the dome and cylinder were treated as thin-walled shell structures, which resulted in a membrane analysis. Since the thickness of the dome and cylinder is small in comparison with the radius of curvature (1/15) and there are no discontinuities such as sharp bends in the meridional curves, the stresses due to pressure, tornado wind or earthquake were calculated by assuming that they are uniformly distributed across the shell thickness.

Since the concrete was not assumed to resist any tensile or shear forces, radial shear reinforcing was 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. Likewise, diagonal shear reinforcing in the circumferential direction was included to resist earthquake shears for the full height of the wall and a distance above the springline into the dome until a point was reached where the dome liner and meridional and hoop reinforcing can resist the total shear. The base slab was treated as a flat circular plate supported on a rigid non-yielding foundation.

5.1.3.3 Dome Analysis

The analysis of the hemispherical dome was performed by the superposition of membrane forces resulting from gravity, accident pressure and accident thermal loads. In addition, tornado, earthquake or wind loading create both direct and shear stresses in the dome and 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. The liner of the dome above a certain point can resist shear load and the anchorages were designed to assure composite action. The dome reinforcing was spliced to the vertical steel in the cylindrical concrete wall, so that a continuity between the dome and the cylinder was realized.

Discontinuity effects at the springline 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.

5.1.3.4 Cylinder Analysis

The analysis of the cylinder was done by superposition of membrane forces resulting from gravity, pressure and thermal loads, over-turning due to tornado, earthquake or wind and shears due to tornado, earthquake or wind. The concrete was reinforced circumferentially using steel hoops and vertically by straight bars. Diagonal bars were placed to resist the horizontal and vertical shears due to tornado, earthquake or wind. The required capacity of the diagonal bars was designed so that the horizontal component per foot of the diagonals equaled the maximum value of shear flow.

A check was made to insure 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 was taken for this capacity.

Only in the upper area of the dome (beyond about 30 degrees above the springline), where the seismic shears are small, does the liner help to resist shear. For all of the cylinder and the lower areas of the dome, the diagonal reinforcing was designed to accommodate all seismic shears. No credit was taken for the dowel action of the vertical and horizontal bars in resisting seismic shear. The maximum stress in the rebar beyond about 30 degrees above the springline due to an earthquake was determined by resolution of the principal tensile stress into components parallel to the rebar. This rebar provides an adequate mechanism to resist shear. (See Section 16.1).

A detailed description of the methods of analysis used for the containment concrete structures is presented in Appendix 5A, Section 4.0.

5.1.3.5 Seismic Design Summary

The design of the Containment which is a seismic Class I structure (see Chapter 16) was based on a “response spectrum” approach in the analysis of the dynamic loads imparted by earthquake. The seismic design took into account the acceleration response spectrum curves developed by G. Housner. Seismic accelerations were computed as outlined in the AEC TID-7024[3] and Portland Cement Publication[4].

As indicated in Chapter 16, ground accelerations used for Operational Basis Earthquake are 0.1g horizontally and 0.05g vertically and for the Design Basis Earthquake are 0.15g horizontally and 0.10g vertically. The natural period of vibration was computed by a dynamic analysis; in this method, the containment structure was analyzed a simple cantilever, consisting of lumped masses and weightless elastic columns acting as spring restraints. Both bending and shear deformations were considered. The natural frequencies and mode shapes were computed from the equations of motion of the lumped masses. These equations were solved by iteration techniques by a fully tested digital computer program. Based on an uncracked concrete section, the period was determined to be 0.241 Sec.

The response of each mode of vibration to the earthquake ground motion was computed by the response spectrum technique. The participation of each mode was computed and the relative acceleration of each mass was determined using the response spectrum curves for 2% and 5% critical damping. The total response was computed as the square root sum of the squares of the individual modes.

Seismic shears are resisted by diagonal reinforcing except in the upper areas of the dome. No credit was taken for the reinforcing in compression.

From 30 degrees above the springline, the shear is resisted by the liner and rebar. The shear is transmitted to the liner by means of tees welded to the liner.

To facilitate construction, a retaining wall was built to carry the roadway at Elevation +95.0 to the northeast quadrant of the Containment Building. There is no backfill between the retaining wall and the Containment, consequently, there was no lateral earth pressure factored into the seismic design.

5.1.3.6 Tornado Design Summary

The design of the Containment, which is a seismic Class I structure, considered the effects resulting from tornado loads.

Tornado wind loading was taken as 300 mph tangential wind traveling with a forward velocity of 60 mph. Also considered as a separate and as a combined loading condition was a 3.0 psi pressure drop external to the structure.

The wind load was considered for three tornado conditions. One included a tangential velocity of 300 mph and a translational velocity of 60 mph. This load superposition depicts a tornado condition where the funnel coincides with the center of the Containment. Load pressure distribution patterns that resulted due to various locations of the funnel were considered. The structure was designed for a triangular and a rectangular wind distribution of 360 mph.

The above wind loading and pressure drop design criteria were consistent with the generally accepted tornado design criteria utilized on nuclear power plants in the eastern United States.

The forces from wind loadings were computed based on ASCE Paper 3269-"Transactions of the ASCE Vol. 126 Part II 1961."

The forces were 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 resulting stresses were limited to yield strength or its equivalent as defined in ACI-318, Part IV B and modified as required by the capacity reduction factor Ø.

The seismic bars provide a more than adequate mechanism to withstand the torsional effect from tornado winds, therefore, tornado winds were not a controlling factor in the design of the containment structure.

In addition, the containment structure will withstand the following Tornado generated missiles (only one missile was considered acting at any time simultaneously with the 360 mph wind load):

Horizontal Missiles

1) 4" x 12' wood plank at 300 mph
2) 4000 lb auto at 50 mph less than 25' above the ground (25 ft2 contact area).

Vertical Missiles

1) 4" x 12' x 12' wood plank at 90 mph

2) 4000 lb auto at 17 mph less than 25’ above the ground (25 ft2 contact area).

Specific structural effects as the result of missile impact are: 1) missile penetration and 2) structural response to dynamic impact. In addition to the overall structural effects such as overturning moment and base shear, the local structural effects must be considered in the design for tornado wind and generated missile loads. For missile loads, limited local plasticity, structural dynamic response ductility and redistribution of stresses in redundant structures due to plastic action was permitted.

Consideration of tornado loads was not a factor in the design of the Containment structure. The 3 psig negative pressure is approximately 4% of the maximum internal pressure load (1.5P=70.5 psig) thus stresses introduced into the rebar from this load are very small.

5.1.4 Penetrations

5.1.4.1 General

In general, a penetration consists of a sleeve embedded in the concrete wall 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 passed through the embedded sleeve and the ends of the resulting annulus were closed off, either by welded end plates, bolted flanges or a combination of these.

Differential expansion between a sleeve and one or more hot pipes passing through it was accommodated by using a bellows type expansion joint between the outer end of the sleeve and the outer end plate, as shown on Figure 5.1-12.

The components are considered ASME Section XI Class MC or CC components and any repair or replacement activities shall be performed in accordance with ASME Section XI Subsections IWE and IWL of the ASME Code, with certain exceptions whenever specific relief is granted by the NRC.

Pressurizing connections were provided to continuously demonstrate the integrity of the penetration assemblies.

5.1.4.2 Types

Electrical Penetrations

“Cartridge” type penetrations are used for all electrical conductors passing through the Containment. The penetrations are provided with a pressure connection to allow continuous pressurization. Insulating bushings or fused glass seals are used to provide a pressure barrier for the conductor.

These components are considered ASME Section XI Class MC or CC components and any repair or replacement activities shall be performed in accordance with ASME Section XI Subsections IWE and IWL of the ASME Code, with certain exceptions whenever specific relief is granted by the NRC.

Figure 5.1-13 shows a design of typical electrical penetrations. There are approximately 60 electrical penetrations.

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. In the case of piping carrying hot fluid, the pipe is insulated and cooling is provided to maintain the concrete temperature adjoining the embedded sleeve at or below 150 F.

These components are considered ASME Section XI Class MC or CC components and any repair or replacement activities shall be performed in accordance with ASME Section XI Subsections IWE and IWL of the ASME Code, with certain exceptions whenever specific relief is granted by the NRC.

Cooling is provided for most hot penetrations through the use of air-to-air heat exchangers. These are made in accordance with the ASME UPV Code, Section VIII, by welding together two embossed sheets of 10 gage carbon steel material, the embossments 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 centrifugal blower which is backed up with a full sized spare. The isolation features and criteria for piping penetrations are given in Chapter 6. Figure 5.1-12 shows typical hot and cold pipe penetrations.

Loss of cooling for the sleeve is highly improbable. The heat shield has no moving parts, and the cooling air is at low pressure. There are redundant blowers to assure that cooling air is not lost for a significant time. The blowers operate off a diesel bus and can be manually started following a blackout. The thermal insulation on the pipe wall reduces heat flow to the liner sleeve. Operation of the cooling unit can be ascertained by opening the “flow through” connection of the penetration pressurization system on the penetration sleeve and observing the temperature of the cooling air emerging.

In order to lose significant structural properties, concrete must be held continuously at 500 to 600 F. The hottest penetrations are the main steam lines, which normally operate at a temperature of 507 F. The results of a two dimensional transient heat transfer analysis indicated that in the improbable case that all cooling air would be lost to the main steam penetrations, the surrounding concrete would reach a maximum temperature of 200 F in approximately 100 hours and 280 F in approximately 1000 hours. It is highly improbable that cooling air would be lost a very long period of time since the failure of any of the air blower drive motors is alarmed in the control room. Even if the adjoining concrete did reach these temperatures (200 – 300 F), the strength of the structure would not be impaired for two reasons:

1) No credit was taken for the tensile strength of the concrete.

2) These temperatures have substantially no effect on the strength of the penetration sleeve or the reinforcing bar in the area of the penetration.

A total of approximately 80 pipes pass through approximately 50 penetration sleeves, 23 of which are considered thermally hot. In addition, several spare sleeves (capped and pressurized) are provided for the possible future addition of piping.

Equipment and Personnel Access Hatches

An Equipment Hatch was provided. It was fabricated from welded steel and furnished with a double-gasketed flange and bolted dished door. The hatch barrel is embedded in the containment wall and welded to the liner. Provision was 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. 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 annunciator situated in the Control Room indicate the door position status. An emergency lighting and communication 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 to the outer door from outside, is possible by the use of special door unlatching tools. The design was in accordance with Section VIII of the ASME Code.

These components are considered ASME Section XI Class MC or CC components and any repair or replacement activities shall be performed in accordance with ASME Section XI Subsections IWE and IWL of the ASME Code, 1992 Edition with certain exceptions whenever specific relief is granted by the NRC.

Outage Equipment Hatch (OEH)

Outage Equipment Hatch can be used in place of the Equipment Hatch at Elevation 95'-0" in the Containment Building during outages. The OEH will be attached to the Containment Building using the same attachments for the Equipment Hatch. The OEH door can be closed and sealed in less than 30 minutes and is designed to withstand the radiation release from a fuel handling accident involving recently-irradiated fuel (i.e, fuel subcritical for less than 84 hours).

The OEH is provided with sealed service penetrations for the passage of service lines (i.e., compressed air, electricity, fluid carrying hoses, instrumentation, fiber optic cables, etc.).

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-inch stainless pipe installed inside a 24-inch pipe. The inner pipe acts as the transfer tube. The transfer tube is fitted with a pressurized double gasketed blind flange on the refueling canal end to seal the reactor containment. The terminus of the tube outside the containment is closed by a standard gate valve. 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 operations. Bellows expansion joints are provided on the pipes to compensate for any differential movement between the two pipes or other structures. Figure 5.1-14 shows a sketch of the fuel transfer tube.

2) <u>Containment Supply and Exhaust Purge Ducts</u>

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 a signal of high containment pressure or high containment radiation level. The space between the valves is pressurized above calculated peak accident response 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 and Filtration System.

These components are considered ASME Section XI Class MC or CC components and any repair or replacement activities shall be performed in accordance with ASME Section XI Subsections IWE and IWL of the ASME Code, 1992 Edition with certain exceptions whenever specific relief is granted by the NRC.

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. The 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 valves from opening.

Failure of any of the valves to open will prevent the fans from running. Tripping or either of the purge fans will automatically close the butterfly valves and pressurize the space between the valves. Failure of any of the valves to close will prevent the adjacent space from being pressurized, and 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 and pressurization solenoids are controlled from a single control switch on the fan room control panel. The cycle is initiated by setting the control switch to “open” position. This will energize the pressurization alarm.

When the pressure between the valves has been relieved, the valves control solenoids are energized and the valves opened. If for any reason, any of the four valves fail to open within a given time after the cycle is initiated, all four valves will close and pressure will be restored. The circuit is interlocked to prevent inadvertent opening of the valves during S.I. condition.

Once all four valves have been opened, the operator has a pre-determined time (approximately one minute) to start the purge supply fan. Failure to do so will cause all four valves to close.

Position indicating lights for each of the four valves are provided on the Fan Room Control Panel and 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 Penetrations

Criteria

The liner is basically not a load-carrying member because it is subjected to strains imposed by the reinforced concrete; nevertheless, the liner was reinforced at each penetration in accordance with the ASME Code Section VII. The weldments of liner to penetration sleeve are of sufficient strength to accommodate stress concentrations and adhered strictly to ASME Code Section VIII requirements for both type and strength.

Liner stress is imposed on the cylindrical penetration as a circular uniform load acting around the circumference of the penetration. The penetration thicknesses were chosen to accommodate this load without causing severe distress at the opening.

The penetration sleeves and plates were 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).

In the design of the piping penetration sleeves and the piping going through them, maximum total stress in all cases was limited to a value below the yield stress of the material involved; therefore, no plastic design criteria were employed. In particular, piping whose failure would result in a Loss-of-Coolant Accident and the main steam and feedwater pipe penetrations and pipe supports in the Containment Building were designed to prevent the formation of a plastic “hinge” in the pipe should any of these pipes rupture. This was accomplished by effectively anchoring these pipes at 90° elbows connected to all these pipes adjacent to the penetration both inside and outside the building, and by restraining these pipes along their run inside the building and outside the building to the first stop valve. The anchors and restraints were designed to prevent a breach of containment at the piping penetrations should any of these pipes rupture inside, immediately outside, or within the penetration itself. The penetrations were designed to the strength of the pipe and no further considerations are necessary.

To insure that a Loss-of-Coolant Accident acting simultaneously with an earthquake would not result in a breach of containment by causing a failure of one or more pipe penetrations through the Containment Building wall, the following methods were used:

All auxiliary piping attached to the Reactor Coolant System which passes through penetrations in the Containment Building wall must also pass through the circular secondary shield wall approximately fifteen feet inside the building as illustrated in Plant Drawing 9321-F-25012 [Formerly Figure 5.1-2]. The total number of pipes in this category is very limited. They were examined individually and suitable restraints or anchors were used either at or within the secondary shield wall to prevent a Loss-of-Coolant Accident or a failure of one of these pipes within the secondary shield wall from causing the failure of the building penetrations through which the pipes pass. In some cases, it was physically impossible for any conceivable movement of the end of those pipes attached to the Primary Coolant System to be reflected at the building penetration and impose other than ordinary operating loads at these points. In other cases, it was necessary to design restraints for the pipes at the secondary shield wall to withstand the failure of the pipe within the wall in tension. Some auxiliary pipes attached to the Reactor Coolant System are attached at points which will not move; for instance, the reactor coolant pump seal water injection pipes and the steam generator blowdown pipes. In general, these have restraints at the secondary shield wall designed for normal loads plus the reaction forces resulting from the double ended rupture of these pipes within the shield wall.

All Containment Building piping penetrations except main steam and feedwater were designed as anchors for the pipes passing through them and 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 loads, loads induced from a Loss-of-Coolant Accident, thermal expansion of the pipe, penetration air pressure, and earthquake loads.

The piping penetrations were designed to transmit the above combined loadings to the concrete structure without exceeding the yield strength of the penetration steel. Typical penetration details are shown in Figure 5.1-12. Load transfer from the pipe to penetration anchorage is limited to the actual loads induced or to the ultimate strength capacity of the pipe in bending, shear, axial, or torsional loadings.

All piping penetrating the Containment meet the requirements of the USAS B31.1.0 Power Piping Code. In the case of the main steam and feedwater lines, the supports, inside and outside the Containment Buildings to the second isolation valve, were designed so that a failure of any one of these pipes does not result in breach of containment of the failure of any other main steam or feedwater pipe between the steam generator and the second isolation valve.

The design of all containment building piping penetration sleeves and end plates except the new Steam Generator Blowdown Penetrations (AA, BB, CC, and DD) and Service Water Penetration (SS) was in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII. The Steam Generator Blowdown Penetration Sleeves and end plates and the Service Water Penetration (SS) end plates were designed in accordance with the requirements of the 1986 edition of the ASME Boiler and Pressure Vessel Code, Section III subsection NC.

These components are considered ASME Section XI Class MC or CC components and any repair or replacement activities shall be performed in accordance with ASME Section XI Subsections IWE and IWL of the AMSE Code, with certain exceptions whenever specific relief is granted by the NRC.

Pipes which penetrate the containment building wall and which are subject to machinery originated vibratory loadings, such as the Reactor Coolant Pumps, had 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. Pipe line 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.

Materials

The material for penetrations including the Personnel and Equipment Access Hatches, together with the mechanical and electrical penetrations is carbon steel, conforming with the requirements of the ASME Pressure Vessels Code Section VIII, and exhibiting ductility and welding characteristics compatible with the main liner material. The Equipment Hatch, penetration sleeves and Personnel Lock meet the Charpy V-notch impact values for a minimum of 15 ft-lbs at -50°F.

The stainless steel expansion joints (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. These were left in place permanently if there was no interface with nearby piping or equipment.

Due to cracking in the bellows of the Main Steam and Boiler Feedwater penetrations, replacement bellows were installed. The replacement bellows are constructed of improved materials.

The materials making up the penetrations conform to the following specifications:

Item		Specification	Minimum Yield Strength (PSI)	Minimum Tensile Strength (PSI)	Elongation
1. Mech. Penetration Sleeve – 12" Dia. & under**		ASTM A333, Gr. 1	30,000	55,000	35% in 2"
2. Mech. – Over 12" Dia.**		ASTM A201 Gr. B to A300	32,000	60,000	22% in 8"
3. Rolled Shapes+		ASTM A36, ASTM A131 Gr. C	36,000 32,000	58,000 58,000	20% in 8" 21% in 8"
4. End Plates	a)	ASTM A300 C1.1 Fire Box A201, Gr B(1)	32,000	60,000	22% in 8"
	b)	ASTM A240 Type 304L+	25,000	70,000	40% in 2"
***	c)	ASTM A516, Gr. 60	32,000	60,000	21% in 8"
5. Fuel Transfer Tube+		ASTM A240 Type 304L	25,000	70,000	40% in 2"

6. Bellows+	a)	ASTM A312 Type 304L	25,000	70,000	35% in 2"
	b)	ASME SB168 Inconel 600++	35,000	80,000	30% in 2"
7. Elec. Penetrations**		ASTM A333 Gr. 1	30,000	55,000	35% in 2"
8. Equip. Hatch Insert**		ASTM A300 C1.1 Firebox A201, Gr. B	32,000	60,000	22% in 8"

** The Equipment Hatch, penetration sleeves and Personnel Lock were Charpy tested to a minimum of 15 ft-lbs at -50°F.

+ No specific NDTT requirements

++ Main Stream and Main Feedwater penetrations

*** Service Water Penetration SS end plates were Charpy V-notch tested to a minimum of 20 ft-lbs (1of 3 test only) at 0°F or lower with a minimum average of three tests of 25 lbs

Item	Specification	Minimum Yield Strength (PSI)	Minimum Tensile Strength (PSI)	Elongation
9. Equip. Hatch Flanges**	ASTM A300, C1.1 Firebox A201, Gr. B	32,000	60,000	22% in 8"
10. Equip. Hatch Head**	ASTM A300 Firebox A201, Gr. B	32,000	60,000	22% in 8"
11. Personnel Hatch**	ASTM A300, C1.1. Firebox A201, Gr. B	32,000	60,000	22% in 8"
12. Piping Penetration Reinf.*	ASTM A442, Gr.60	32,000	60,000	22% in 8"

13. Outage Equipment Hatch is designed and built in accordance with ASME Section VIII, 1989 and made of ASTM A516 Grade 60 or higher Grade material. The structural steel members are made of SA 36; Pipe penetrations are made of ASTM A 106 Grade B.

NOTE:

* The liner plates for the shell, bottom and dome were impact tested on a longitudinal section at 15 ft-lbs at a temperature 30 degrees below the service temperature of -50°F.

** The Equipment Hatch, penetration sleeves and Personnel Lock were Charpy tested to a minimum of 15 ft-lbs at -50°F.

Consideration of Jet Loads, Missile Impact and Tornado Loads for Openings

The 3'-0" thick crane wall, the 4'-0" and 6'-0" thick Refueling Canal and the 2'-0" thick operating floor are capable of resisting jet force loads and missiles from primary coolant piping. Thus, jet force loads and missiles from the potential failure of the Primary Coolant System are contained within the reactor coolant compartment shield walls and cannot impinge on the containment structure walls; consequently, these loads were not considered in design of large openings. All other missiles terminate inside these concrete shield walls and consequently were not factored into the large opening design. Large openings are shielded or are far enough away to preclude impingement from main steam and feedwater pipe break loads.

Tornado loads are small compared to the seismic loadings. The tornado shear loads from torsion and translational wind force and the overturning moments caused by wind load have a minimum factor of safety of approximately 2.5 when compared with earthquake shears and moments which were used to size the seismic reinforcing bars. The tornado moment and shears are in fact smaller than the minimum earthquake moments and shears considered in design. On this basis, the seismic bars provide more than an adequate mechanism for resisting tornado loads. In addition, tornado loads act independently of other severe loads; therefore, the Equipment Hatch and Personnel Lock reinforced concrete bosses were designed for simultaneous design basis accident and earthquake loads, which were larger than tornado loads, are of more than adequate strength to resist tornado loads.

The containment structure will not be penetrated by the tornado-generated missiles. The concrete sections around large openings are thicker than the 4'-6" Containment wall and so no further consideration of tornado missiles at the large openings was necessary. The large openings have shielded walls of sufficient thickness to protect against tornado missiles.

Consideration of Curvature of the Wail in the Finite Element Analysis

Curvature of the containment cylinder wall was included in the finite element analysis for large openings by assigning three coordinates to each node point in the model. This in effect idealizes the structure as a series of chords of a circle with radii equal to the containment cylinder reference surface. Since the widest element in the fine model at the Equipment Hatch opening is 50", or approximately 1% of the total circumference, the chords adequately represent the curvature of the containment surface. Since the shape and stiffness of the structure was accurately represented in the model, all forces and effects were included in the computer output.

The procedures used to design for the six stresses and the justification for all structural elements (rebar) provided to resist the forces or stress resultants outputted by the computer are discussed in detail in Appendix 5A. All concrete in tension was considered cracked in the finite element analysis.

5.1.4.4 Leak Testing of Penetration Assemblies

A proof test was supplied 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 leak existed.

5.1.4.5 Construction

The qualification of welding procedures and welders was in accordance with Section IX, "Welding Qualifications" of the ASME Boiler and Pressure Vessel Code. The repair of defective welds was in accordance with paragraph UW-38 of Section VIII "Unfired Pressure Vessels."

For penetrations between 9" and 18", all the reinforcing bars including primary and secondary vertical bars and diagonal bars are grouped around the penetrations. Due to the continuity of the bars and the relatively small opening size, no special provisions were needed to resist normal, shear and bending stresses. The penetrations are keyed into the concrete, thus creating an edge loading which induces torsion into the walls. The loads are small and the rebar feels little effect from this torsional loading.

For penetrations greater than 18" to 4'-0" the bars are continuous. Since reinforcing is continuous around penetrations, steps were taken to insure that no local crushing of concrete occurred.

From an article, "Detailing and Placing Reinforcing Bars" by Paul F. Rice from Concrete Construction, January 1965, it was 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 which is transmitted to the concrete in the event the bar tries to straighten out due to tension. For this reason, most bars were 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 10o bend. In these cases, the bars were bent at 30 degrees but a tie back system was used which prevents a build-up of forces. To further 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 for the vertical bars and from a vertical line for the horizontal bars and is tangent to the outside of the penetration.

Details of the Personnel and Equipment Hatch design are presented in Section 3.4 of Appendix 5A.

Concrete was poured in nominal 5' lifts, 360 degrees with no stagger. Approximately one week was allowed to elapse between pours and the surface was left rough, thoroughly cleaned by air blowdown, and all laitance removed. Joints were thoroughly wetted and slushed with a coat of neat cement grout immediately before placing of new concrete except for the exterior of the containment where surfaces were thoroughly wetted but not grouted.

5.1.4.6 Testability of Penetrations and Weld Seams

All penetrations, the Personnel Air Lock and the Equipment Hatch were designed with double seals which are normally pressurized at a minimum pressure greater than the calculated peak accident pressure. Individual testing at 115% containment design pressure is also possible.

These components are considered ASME Section XI Class MC or CC components and any repair or replacement activities shall be performed in accordance with ASME Section XI Subsections IWE and IWL of the ASME Code, with certain exceptions whenever specific relief is granted by the NRC.

The containment ventilation purge ducts are equipped with double isolation valves and the space between the valves is permanently piped up to the penetration pressurization system. The space can be pressurized to 115% 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. Most welds are continuously pressurized during power operation at a minimum pressure greater than the calculated peak accident pressure. Liner welds that are not pressurized during power operation are those welds associated with disconnected sections of the Weld Channel Pressurization System. The integrity of the welds associated with any disconnected sections of the Weld Channel Pressurization System is verified by integrated leak rate testing.

Test connections are provided on the Penetration and Weld Channel Pressurization System lines to the Equipment Hatch and Personnel Airlock to allow for leak testing of the PWCP connections.

The use of the weld channel pressurization system may necessitate periodic relief of pressure buildup within the containment, should the system leak into the containment structure.

When pressure relief of the Containment is required during normal operation, it is accomplished using the containment pressure relief line and not the containment purge lines. However, the pressure relief exhaust is routed through charcoal filters which have an iodine removal efficiency of 90.0%. Prior to pressure relief operations, the Containment Auxiliary Charcoal Filter System (see Section 5.3) may be operated to reduce the activity in the containment atmosphere. Assuming 1% fuel defects and 50 lbs/day leakage of reactor coolant into the Containment, the containment atmosphere activity has the maximum value of 20.4 x MPC for iodines and 135.5 x MPC for noble gases after approximately 16 hours of operation of the containment auxiliary charcoal filter system whose efficiency for iodine removal is 90%.
The activity released to the environment as a result of depressurizing the Containment from 1.0 psig to 0 psig at 1500 CFM for 2 hours based on the above abnormal conditions is:

a) For iodines: 8.26×10^{-15} curies expressed as equivalent I-131
b) For noble gases: 6.68 curies expressed as equivalent Xe-133

The maximum expected operating conditions considered as normal are taken as 0.2% fuel defects and 14.4 gpd leakage of reactor coolant into the Containment. For these conditions, the containment atmosphere activity is 8.16 x MPC for iodines and 65.0 x MPC for noble gases after approximately 16 hours operation of the containment charcoal filter system whose efficiency for iodine removal is 99%. The activity released to the environment as a result of depressurizing the containment from 1.0 psig to 0 psig at 1500 cfm for 2 hours through the purge line carbon filters (iodine removal efficiency of 99.0%) based on these maximum operating conditions is:

a) For iodines: 2.49×10^{-6} curies expressed as equivalent I-131
b) For noble gases: 3.21 curies expressed as equivalent Xe-133

5.1.4.7 Accessibility Criteria

The Containment is completely closed whenever the core is critical or whenever the primary system temperature is above 200 F, except as required for brief periods necessary to relieve the Containment to keep the pressure below a reasonable level (1-2 psig) or to purge the Containment in preparation for Containment entry.

Limited access to the Containment through personnel air locks is possible with the reactor at power or with the primary system at hot shutdown for special maintenance or periodic inspections. Access at power would normally be restricted to the areas external to the reactor equipment compartment primarily for inspection and maintenance of the air recirculation equipment, incore instrumentation chamber drives, and instrument calibration.

After shutdown, the Containment vessel is purged to reduce the concentration of radioactive gases and airborne particulates. This purge system was designed to reduce the radioactivity level to doses defined by 10 CFR 20 for a 40-hour occupational work week, within 2-6 hours after plant shutdown. Since negligible fuel defects are expected for this reactor, much less than the 1% fuel rod defects used for design, purging of the Containment is normally accomplished in less than 2 hours. To assure removal of particulate matter the purge air will be passed through a high efficiency filter before being released to the atmosphere through the purge vent.

The primary reactor shield was designed so that access to the primary equipment is limited by the activity of the primary system equipment and not the reactor.

5.1.5 System Design Evaluation

5.1.5.1 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 was designed to maintain isolation of the Containment from the outside environment. Provisions are made to continuously pressurize penetrations and most weld channels and to monitor leakage from this pressurization.

5.1.5.2 System Integrity and Safety Factors

Pipe Rupture – Penetration Integrity

The penetrations for the main steam, feedwater, blowdown and sample lines were designed so that the penetration is stronger than the piping system and that the vapor barrier will not be breached due to a hypothesized pipe rupture.

Major Component Support Structures

The support structures for the major components were 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 were designed for stresses not exceeding yield stress due to these forces.

5.1.5.3 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.

5.1.5.4. Performance Capability Margin

The containment structure was designed based upon limiting load factors which were used as the ratio by which accident and earthquake loads were 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. Tabulations of load combinations and load factors utilized in the design which provide an estimate of the margin with respect to all loads are referenced in Section 5.1.2.

5.1.6 Minimum Operating Conditions

The minimum operating conditions which are applicable to the Containment System are given in the Technical Specifications.

5.1.7 Containment System Structure-Inspection and Testing

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 form 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 of 7/11/67)

After completion of the containment structure and installation of all penetrations and weld channels, integrated leakage rate tests were performed prior to initial plant operations to establish the respective measured leakage rates and to verify that the leakage rate at the peak accident conditions is no greater than 0.075 percent by weight per day of the containment stream-air atmosphere at the calculated peak accident conditions. The leakage rate tests were performed using the absolute method. The duration of each test was not less than 24 hours.

Periodic Containment Leakage Rate Testing

Criterion: The containment shall be designed so that an integrated leakage rate can be periodically determined by test during plant lifetime. (GDC 55 of 7/11/67)

The peak accident pressure integrated leakage rate test is conducted at periodic intervals during the life of the plant, and also as appropriate in the event major maintenance or major plant modifications are made.

A leak rate test at the peak accident pressure using the same test method as the initial leak rate 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.

These components are considered ASME Section XI Class MC or CC components and any repair or replacement activities shall be performed in accordance with ASME Section XI

Subsections IWE and IWL of the ASME Code, with certain exceptions whenever specific relief is granted by the NRC.

Provisions for Testing of Penetrations

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 of 7/11/67)

Penetrations were designed with double seals which are continuously pressurized above accident pressure. The large access openings such as the Equipment Hatch and Personnel Air Lock are equipped with double gasketed doors and flanges with the space between the gaskets connected to the pressurization system. The system utilizes a supply of clean, dry, compressed air which places 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.

These components are considered ASME Section XI Class MC or CC components and any repair or replacement activities shall be performed in accordance with ASME Section XI Subsections IWE and IWL of the ASME Code, with certain exceptions whenever specific relief is granted by the NRC.

Provisions for Testing of Isolation Valves

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 of 7/11/67)

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. Removal or bypass of one signal channel places that circuit in the half-tripped mode.

Local leak rate testing of containment isolation valves is performed in accordance with Technical Specification 5.5.15. The Containment Leakage Rate Program is in accordance with the guidance contained on Regulatory Guide 1.163, except as noted in the Technical Specification.

Field and operational inspection and testing were divided into three phases:

1) those taking place during erection of the Containment Building liner; construction tests

2) those taking place after the containment structure was erected and all penetrations were complete and installed; pre-operational tests

3) monitoring during reactor operation; post-operational tests

These components are considered ASME Section XI Class MC or CC components and any repair or replacement activities shall be performed in accordance with ASME Section XI Subsections IWE and IWL of the ASME Code, 1992 Edition with certain exceptions whenever specific relief is granted by the NRC.

5.1.7.1 Construction Tests

During erection of the liner, the following inspection and tests were performed:

Bottom Liner Plates

All liner plate welds were tested for leak tightness by vacuum box. The box was 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 a 54 psig air pressure to the channels in the zone for a period of 15 minutes.

The zone of channel-covered welds was pressurized to 47 psig with a 20% 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 is 1 x 10-9 standard CC per second. The sniffer was held approximately ½ inch from the weld and traversed at a rate of about ½-inch per second. The detection of any amount of halogen, indicating a leak, required 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 two hours. The drop in pressure was not to exceed the equivalent of a leakage of 0.05% of the containment building volume per day. Compensation for change in ambient air temperature was made if necessary.

Vertical Cylindrical Walls and Dome

For the liner, a complete radiograph was made of the first 10 feet of full penetration weld made by each welder or welding operation. A minimum of a 12" film "spot" radiograph was made every 50 feet of weld thereafter on the side walls and dome, except where back-up plates are used. The radiograph films were given to United Engineers and Constructors for their review.

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% radiographed to determine the end of defect.

The performance and acceptance standards for all radiography is ASME Section VIII, Paragraph UW51.

The liner plate to plate welds were tested for leak tightness 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 pre-determined 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 minutes. In addition, each zone of channel covered weld was leak tested under the Freon-air mixture at 47 psig.

In location where radiography was not possible, such as the lower courses of shell plates where back-up plates were used, and where liner bottom welds and floor plates were made to angles and tees, the liner fabricator welded on a 2" long overrun coupon. The overrun coupon was chipped off, marked for location and given to United Engineers and Constructors for testing. These welds are 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 non-destructive examination of liner plate.

Liner Erection Tolerance

Deviations from the allowable erection tolerance standards were located, documented and, in most cases, eliminated during the normal erection of the liner. This was accomplished by jacking against the polar crane wall, utilizing tubular beams, capped by beams of sufficient cross-sectional area to insure against localized buckling of the liner plate. For areas above the concrete polar crane wall, the required tolerances were met and maintained by circular plate wind girders. For the isolated cases where the liner could not be jacked into tolerance, a Non-Conformance Report (NCR) was written and forwarded to the architect-engineer with a complete survey of the area for an engineering evaluation together with a waiver request. This documentation is maintained by the Authority. Only minor deviations were experienced.

Concrete Compression and Slump Testing

The compression test samples consisted of six 6" x 12" cylinders for each 100 cu. yd. or portion thereof, per class, per day. A minimum of one set of six cylinders was made for pours of less than 100 cu. yds. Three cylinders were broken at 7 days and 3 cylinders at 28 days. The basis for rejection was failure to develop a minimum compressive strength at 28 days of 15% above the nominal design strength as proven on an average of the three cylinders.

A slump test was performed on each truckload of concrete used in the first four lifts (20 feet) for the containment exterior wall and was recorded for each sample from which compression test cylinders were made. For all other concrete, a slump test was made and recorded for three truckloads of concrete from each class of concrete per 100 cubic yards (or portion thereof) placed per day. A Quality Control inspector was present during the pour and visually checked the concrete from each truck. Any concrete which appeared to be near or over the limit was slump tested. Wet loads were rejected. The maximum slump for all pours was 5 inches except for special pours when specific approval was received from the Architect-Engineer. In no case was the slump permitted to exceed 7 inches.

The statistical results of compression testing for the 28 day breaks were:

a) 100% of the cylinder break tests exceed the minimum requirement
b) 75% of the cylinder break tests exceeded the minimum requirement by at least 1000 psi
c) 50% of the cylinder break tests exceeded the minimum requirement by at least 1250 psi
d) 25% of the cylinder break tests exceeded the minimum requirement by at least 1750 psi
e) 10% of the cylinder break tests exceeded the minimum requirement by at least 2250 psi

The samples for compression and a slump testing of concrete were taken from the point of discharge from the truck. There was no occurrence of pour removal or concrete rejected from these test results.

Cadweld Splice Test Program

In the Cadweld Test Program, tests were performed on production Cadwelds which had been removed (specifically for testing) from the Containment Building after placement. Of the first 141 production Cadwelds tested in this program, all test results were in excess of the minimum specified strengths.

The following test results were obtained from the actual Cadweld test reports submitted to WEDCO from Consolidated Testing Laboratory. Of the Cadwelds tested:

100% had ultimate strengths of at least 79,000 pis
75% had ultimate strengths of at least 95,100 psi
50% had ultimate strengths of at least 97,600 psi
25% had ultimate strengths of at least 102,600 psi
10% had ultimate strengths of at least 105,100 psi

A statistical analysis of these results was performed using the methods outlined in Appendix 5A, Section 5.2.1.

The mean value of the ultimate strength of the splices was 99,580 psi with a standard deviation of 9.960 psi and a total range of 32,750 psi. Of the total at least 99% had an ultimate strength of 76,373 psi. No Cadwelds were rejected on the basis of test results from the Cadweld Test Program.

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.

5.1.7.2 Pre-Operational 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 calculated peak accident response pressure of the containment. This blocks 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. Certain liner welds are no longer continuously pressurized. Therefore, the leakage status of these welds is no longer continuously monitored. The integrity of these welds is verified by integrated leak rate testing.

After the Containment Building was complete with liner, concrete structures, and all electrical and piping penetrations, Equipment Hatch and Personnel Lock in place, the following tests were performed:

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 one hour. 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 the Containment Report, Appendix 5A.

2) Integrated Leakage Rate Tests:

   Integrated leakage rate tests were performed on the completed building using the absolute method. These leakage tests were performed with the double penetration and weld channel zones open to the containment atmosphere.

3) Sensitive Leak Rate Test:

   After it had been assured that there were no defects remaining from construction, a sensitive leak rate test was conducted. The sensitive leak rate test included only the volume of the weld channels and double penetrations. This test is considered more sensitive than the integrated leakage rate test, as the instrumentation used permits a direct measurement of leakage from the pressurized zones. The sensitive leak rate test was conducted with the penetrations and weld channels at a minimum pressure greater than the calculated peak accident pressure 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% of the containment free volume per day.

In order to verify that the structural response of the Containment to pressure loads is in accordance with design assumptions and to provide assurance that the structure was constructed in accordance with the design to resist pressure loads, a Structural Integrity Test (SIT) was performed.

Readings and measurements were taken at 0 psig, 12 psig, 21 psig, 41 psig and 54 psig (the latter is 115% of the design pressure of 47 psig) during pressurization, and at 41, 18, 21, 41, and 0 psig during depressurization.

The following gross deformation measurements were taken during the SIT using invar wire extensometers. This provided a means for taking all measurements inside the containment structure thus eliminating effects of weather and temperature. All results were remotely recorded during the test and data was quickly reduced.

a) Radial deformation of the containment wall was measured at 15 locations in the thickened Equipment Hatch boss and the transition area from the thickened boss to the 4'-6" cylinder wall.

b) Diameter change in the containment structure was measured at 10 locations spaced at approximately 10’-0" between elevations 101’-0" and 191’-0".

c) Radial deflection of the containment cylinder wall was measured at elevation 91’-0".

d) Vertical deflection of the Containment was measured at elevations 95’-0", 143’-0" and 191’-0" and at the apex of the dome. Redundancy was provided for the measurement at the apex of the dome.

Detailed crack measurements were made prior to the test, at peak test pressure of 54 psig, and following depressurization at five areas of the exterior shell, each of at least forty square feet in area. The areas of detailed measurement were: a quadrant of the personnel lock concrete boss, and ten foot wide strips spanning elevations 43’-0" to 48’-0", 115’-0" to 120’-0", and 188’-0" to 193’-0".

In addition, the exposed surface of the containment shell was visually inspected prior to the test, at 41 psig during the ILRT, and following depressurization. These inspections were for purposes of monitoring the general crack pattern and for specifically following the behavior of the most significant crack.

5.1.7.3 Acceptability of Testing Program

AEC Safety Guide No. 18 “Structural Acceptance Test for Concrete Primary Reactor Containments” was followed for testing except in the following areas:

1) The pattern of measurement points around the largest opening (equipment hatch) were not as shown in Figure C of Safety Guide 18 which indicated 12 points symmetrically located to measure radial and tangential deflections. The Indian Point 3 Structural Integrity Test required taking of radial measurements at 15 locations around the equipment hatch.

   Due to access restrictions, no deflection readings were taken on the lower vertical axis of this opening; the 15 measurement locations were symmetrically positioned in the remaining accessible area around this opening. Tangential deflections were not taken, as they were insignificant compared to the radial deflections. The second largest opening (personnel hatch) was structurally loaded in a manner similar to the equipment hatch; no deflection measurements were taken for the personnel hatch opening. This program of radial deflection measurements provided the necessary data to verify that anticipated deformations were taken into account and were within acceptable limits.

2) The structural integrity of the OEH was tested in the Vendor’s shop to 7.5 psig for 10 minutes, then the pressure was dropped to 6 psig and the air supply was closed. All tests were performed in accordance with the requirements of ASME B&VP code Section VIII, 1989 Code Part UG-99 or UG-100 for the fabricated Carbon Steel.

5.1.7.4 Post-Operational Tests

The double penetrations and most weld seam channels which were installed on the inside of the liner in the Containment are continuously pressurized to provide a continuous, sensitive and accurate means of monitoring their status with respect to leakage. Certain liner welds are no longer continuously pressurized. Therefore, the leakage status of these welds is no longer continuously monitored. The integrity of these welds is verified by integrated leak rate testing.

No periodic structural integrity tests of the Containment are planned. Periodic peak pressure containment integrated leakage rate test (ILRTs) are performed in accordance with the Technical Specifications. Peak pressure tests are to be conducted as appropriate in the event major maintenance or major plant modifications are made. As a prerequisite to the ILRT, a detailed visual examination of the accessible interior and exterior surfaces of the containment structure and its components is required to uncover any evidence of deterioration which may affect the containment integrity. However, no degradation of structural integrity is expected. The Authority does not consider periodic structural integrity tests as warranted either separately or in conjunction with other tests.

The Containment Leakage Rate Testing Program details requirements for inspection of the accessible interior and exterior surfaces of the containment structure and its components. This periodic surveillance of the Containment and associated structures is visual and includes critical areas as well as a general examination of the accessible surfaces for deterioration. The inspection is also performed prior to any integrated leak test. The insulation attached to the steel liner is designed so that sections can be removed to facilitate inspection of the liner.

These components are considered ASME Section XI Class MC or CC components and any repair or replacement activities shall be performed in accordance with ASME Section XI Subsections IWE and IWL of the ASME Code, with certain exceptions whenever specific relief is granted by the NRC.

Provisions have been made for access to the upper external parts of the containment structure. These provisions consider the use of movable scaffolding while performing periodic inspection and testing during the service life of the facility.

References

1. Stellmeyer, J. B., W. H. Munse and E. A. Selby, “Fatigue Tests of Plates and Beams with Stud Shear Connections.” Highway Research Record, Number 76.

2. Singleton, Robert C. “The Growth of Stud Welding.” Welding Engineer, July 1963.

3. United States Atomic Energy Commission – Nuclear Reactors and Earthquakes, TID-7024, 1963.

4. Blume, J., N. Newark, L. Corning – Design of Multistory Reinforced Concrete Building for Earthquake Motions – Portland Cement Association.

5. Timoshenko, S., and S. Woinowsky-Kreiger, Theory of Plates and Shells, Second Edition, McGraw-Hill, 1954.

6. American Concrete Institute, Code for Reinforced Concrete Chimney Design, ACI-505.

TABLE 5.1-1

FLOODED WEIGHTS – CONTAINMENT BUILDING

Item	Flooded/Equipment Weight, lb
Pressurizer –1	346,000
Steam Generators – 4	3,816,400
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	129,000
Other Miscellaneous Equipment	100,000
TOTAL	9,358,400

5.2 CONTAINMENT ISOLATION SYSTEM

5.2.1 Design Basis

Each system whose piping penetrates the Containment's leakage limiting boundary was designed to establish or maintain isolation of the Containment from the outside environment under the following postulated conditions:

a) Any accident for which isolation is required (severely faulted conditions) coincident with

b) An independent single failure or malfunction (expected faulted condition) occurring in any active system component within the isolated bounds.

Piping penetrating the Containment was designed for pressures at least equal to the containment design pressure. Containment isolation valves were provided, as necessary, in lines penetrating the Containment to assure 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, which connects to a source of radioactive fluid within the Containment.

In general, isolation of a line outside the Containment protects against releases due to rupture of the line inside concurrent with a Loss-of-Coolant Accident, and 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 of seismic Class I design up to and including the second isolation barrier and are assumed to be an extension of the Containment.

Normally lines located inside the Containment building that are required to function after an accident are located outside the missile barrier. An exception to this is a portion of the closed loop Component Cooling Water system which is located along the inside of the Crane Wall by the 31 Steam Generator. This is acceptable based on the "Modification of the General Design Criteria 4 requirements for protection against Dynamic effects of postulated pipe ruptures." This takes into account the Leak Before Break methodology, which relaxes the pipe rupture requirements for the Reactor Coolant Loop. The Component Cooling Water piping is also protected by concrete walls from the Pressurizer Surge line located on the other side of the Containment therefore the piping meets the intent of the original design criteria of being protected from credible missiles.

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.

The containment isolation valves were examined to assure that they are capable of withstanding the maximum potential seismic loads.

To assure their adequacy in this respect:

a) Valves were located in such a manner as to reduce the accelerations on the valves. Valves suspended on piping spans were reviewed for adequacy for the loads to which the span would be subjected. Valves were mounted in the position recommended by the manufacturer.

b) Valve yokes were reviewed for adequacy, and strengthened as required for the response of the valve operator to seismic loads.

c) Where valves are required to operate during seismic loading, the operator forces were reviewed to assure that system function is preserved. Seismic forces on the operating parts of the valve are small compared to the other forces present.

d) Control wires and piping to the valve operators were designed and installed to assure that the flexure of the line does not endanger the control system. Appendages to the valve, such as position indicators and operators, were checked for structural adequacy.

e) The design of control systems for automatic containment isolation valves is such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves requires deliberate operator action.

Containment Isolation Valves Criteria

Isolation valves were provided as necessary for all fluid system lines penetrating the Containment to assure 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 was designed to facilitate normal operation and maintenance of the systems and to ensure reliable operation of other engineered safeguards systems.

Valves utilized in systems for containment isolation service were selected based on tight shutoff requirements, speed of operations, and materials suitable for service in a particular environment relative to temperature, pressure and radiation activity.

The criteria for level of reliability for control valves listed were based on satisfactory operation of the containment isolation valves for the operating life of the plant with the required leak tightness assured by testing and corrective maintenance as required.

The criteria of reliability for swing stop, check, and gate valves listed were based on documented material, quality assurance, compliance for inspection, welding qualification, seismic criteria, testing, and technical specifications for required leak tightness complying with valve manufacturer's standard practice.

Table 5.2.1 provides a summary of containment isolation valve type, actuator, and closure time established for systems penetrating containment.

With respect to numbers and locations of isolation valves, the criteria applied were generally those outlined by the seven classes described in Section 5.2.2. Specific containment isolation valves are listed in FSAR Table 5.2-3.

5.2.2 System Design

The seven classes listed below are general categories into which line 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 Accident.

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 disk type of gate valve was used to isolate certain lines. When sealed by water or gas injection, this valve provides two barriers against leakage of radioactive liquids or containment atmosphere. In certain cases, a double disc valve was used in place of two valves in series having seal water or gas injection between them.

6) In lines isolated by globe valves in series (inboard and outboard) outside containment and provided with seal water injection, the following applies:

    a) On process lines ingressing containment (incoming lines) IVSWS will be required to wet the stem packings on both the inboard and outboard valve. IVSW wets the valve plug as well as the stem packing of the RCP seal water injection line containment isolation valves (CH-MOV-250A through D),

    b) On process lines egressing containment (outgoing lines) IVSWS will be required to wet only the stem packing on the inboard valve. One exception would be the Steam Generator Blowdown CIVs where both the inboard and outboard valves stem packings are wetted by IVSWS.

7) Excessive loss of seal water through an isolation valve that fails to close on signal is prevented by the high resistance of the seal water injection line. A water seal at the failed valve was assured 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) Lines penetrating containment were designed to the same seismic criteria as the containment vessel up to and including the second isolation barrier. These portions of the penetrating lines are therefore to be considered extensions of the containment.

A review of the Containment Isolation System (NUREG-0578) indicated that there were a number of valves, which automatically reset to the previous position upon reset of containment Phase A isolation. These valves were under operator control via operating procedures to be placed in the closed position prior to resetting of Phase A. Circuits for these valves have been modified to preclude automatic opening on reset. The modification to the valve circuits entailed the installation of pushbuttons that work in conjunction with the containment isolation reset switches so that each valve control circuit has to be reset or the valve will be inhibited from opening.

Class 1 (Outgoing Lines, Reactor Coolant System)

Outgoing lines connected to the Reactor Coolant System which are normally or intermittently open during reactor operation were provided with at least two automatic trip valves in series located outside the Containment. Automatic seal water injection was provided for line in this classification.

Class 2 (Outgoing Lines)

Outgoing lines not connected to the Reactor Coolant System which are normally or intermittently open during reactor operation, and not missile protected or which can otherwise communicate with the containment atmosphere following an accident, were provided, as a minimum, with two automatic trip valves in series outside containment. Automatic seal water injection was provided for lines in this classification with the exception of the reactor coolant pump seal water return line, which was provided with manual seal water injection. Most of these lines are not vital to plant operation following an accident.

Class 3 (Incoming Lines)

Incoming lines connected to open systems outside containment, and not missile protected or which can otherwise communicate with the containment atmosphere following an accident were provided with one of the following arrangements outside containment:

1) Two automatic trip valves in series, with automatic seal water injection. This arrangement was provided for lines, which are not necessary to plant operation after an accident.

2) Two manual isolation valves in series, with manual seal water injection. This arrangement was provided for lines, which 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 which can otherwise communicate with the containment atmosphere following an accident were provided either with two isolation valves in series outside containment with seal water injection between them 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, which contain only one isolation valve.

Exceptions are the containment spray headers and the safety injection header associated with the boron injection tank, which was valved in accordance with safeguards requirements. The containment spray headers have locked-open double disk gate valves while the safety injection header has either single normally-open double disk gate valves or two normally open gate valves arranged in series.

Class 4 (Missile Protected)

Incoming and outgoing lines which penetrate the Containment and which are normally or intermittently open during reactor operation and are connected to closed systems inside the Containment and protected for missiles throughout their length were provided with at least one isolation valve located outside the Containment. Seal water injection was provided for certain lines in this classification.

Class 5 (Normally Closed Lines Penetrating the Containment)

Lines which penetrate the Containment and which can be opened to the containment atmosphere but which are normally closed during reactor operation were provided with two isolation valves in series or one isolation valve and one blind flange.

Class 6 (Special Service)

There are a number of special groups of penetrating lines and containment access openings. Some of these are discussed below.

Each ventilation purge duct penetration was provided with two tight-closing butterfly valves, which are 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 Penetration and Weld Channel Pressurization System, whenever they are closed.

The containment pressure relief line is similarly protected. However, since the line can be 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 Penetration and Weld Channel Pressurization System whenever they are closed.

The equipment access closure is a bolted, gasketed closure, which is sealed during reactor operation. The personnel air locks consist of two doors in series with mechanical interlocks to assure that one door is closed at all times. Each air lock door and the equipment closure were provided with double gaskets to permit pressurization between the gaskets by the Penetration and Weld Channel Pressurization System, Section 6.6.

The fuel transfer tube penetration inside the Containment was designed to present a missile protected and pressurized double barrier between the containment atmosphere and the atmosphere outside the Containment. The penetration closure was treated in a manner similar to the equipment access hatch. A positive pressure is maintained between the double gaskets to complete 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 design pressure (150 psig) in the event of an accident, due to operation of the recirculation pumps:

1) Residual heat removal loop return 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
5) Residual heat removal pump miniflow line
6) Residual heat removal loop outlet line

Lines 1, 2, and 6 are isolated by double disc gate valves, while line 3, 4 and 5 are each isolated by two valves in series. These valves 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.

Lines which communicate with the containment atmosphere at all times (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 is sealed by pressurizing with air from the Penetration and Weld Channel Pressurization System. The air is introduced into each space above the containment calculated peak accident response pressure through a separate line from the Penetration and Weld Channel 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 open, or if one of the containment isolation valves fails to respond to the "trip" signal.

<u>Class 7 (Steam and Feedwater Lines)</u>

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 required that the steam generator shell, feed and steam lines within the Containment be classified and designed for the Reactor Coolant System missile-protected category. The reverse is also true in that a steam line break is not to cause damage to the Reactor Coolant System.

5.2.2.1 <u>Isolation Valves and Instrumentation Diagrams</u>

Plant Drawings 9321-F-27473, -27203, -27353, -27453, -27503, -27513 Sh. 1, -27363, -27193, -27233, -27473, -20253, -27263, -70453, -20173, -20193, -27293 Sh. 1 & 2, -27223, -20353, -40223, -26533, -20363, -26533, and -27243 [Formerly 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. Figure 5.2-29 defines the nomenclature and symbols used. Individual containment isolation valves are listed in Section 5.2 of the FSAR and Table 5.2-3.

5.2.2.2 Normally Closed Isolation Valves

Table 5.2-3 identifies those isolation valves which are either locked closed, or normally closed, (under administrative control) in normal position and relates to Figures 5.2-1 through 5.2-29.

5.2.2.3 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 5.2-3. 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 were 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 automatically tripped isolation valves are actuated to the closed position by one of two separate containment isolation signals. The first of these signals is derived in conjunction with automatic safety injection actuation, and trips the majority of the automatic isolation valves. These are valves in the so-called “non-essential”** process lines penetrating the containment. This is defined as a “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.

NOTE:

* “Essential” are those lines required to mitigate an accident, or which, if unavailable, could increase the magnitude of the event. Also, those lines which, if available, would be used in the short term (24 to 36 hours) to restore the plant to normal operation following an event which has resulted in containment isolation.

** “Non-Essential” are those lines which are not required to mitigate or limit an accident, which if required at all would be required for long-term recovery only, i.e., days or weeks following an accident.

A manual containment isolation signal can be generated from the Control Room. This signal performs the same functions as the automatically derived “T“ signal, i.e., “Phase A” isolation and automatic seal water injection.

Non-automatic 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 two seconds. The typical closing time available for large motor operated gate valves is ten to thirty seconds. These general closure times are shown on Table 5.2-1. They are not used for determining valve stroke time limits. Specific design assumptions, closure times for design basis accidents, containment response analyses and resulting off-site dose calculations are contained in specific analyses.

The large butterfly valves used to isolate the containment ventilation purge ducts are each equipped with spring-assisted air pistons capable of closing the valve in two seconds. These valves fail to the closed position on loss of control signal. They also fail closed upon loss of instrument air through use of a local air reservoir as an energy source.

5.2.2.4 Valve Operability

All containment isolation valves, actuators and controls are located so as to be protected against missiles which could be generated as a 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, were constructed of materials sufficiently temperature resistant to be unaffected by the accident environment. Special attention was given to electrical insulation, air operator diaphragms and steam packing material.

2) In addition to normal pressures, the valves were designed to withstand maximum pressure differentials in the reverse direction imposed by the accident conditions. This criterion was particularly applicable to the butterfly type isolation valves used in the containment purge lines.

5.2.2.5 Valve Position Indication and Monitoring

In general, all remote operated valves have visual position indication in the Control Room. Table 5.2-4 lists the containment isolation valves and the location of each valve’s position indicator lights. Two different types of indicating lights are used: 1) red and green lights and 2) white and red monitoring lights. The red position indicating light is on when the valve is fully open and the green position indicating light is on when the valve is fully closed. At all other positions, both the red and green position indicating lights are on. The red monitor light is on when the valve is in its safeguard position and the white monitor light is on when the motive power is available to the valve. For those valves that are normally de-energized, the white monitor light indicates that power is available to the monitor indicating circuit.

Remote operated containment isolation valves, which are under remote manual control and do not receive a signal from the ESF actuation system, were provided with visual indication of position. An audible alarm feature was provided for remote operated safeguards valves under remote manual control for safeguards functions to denote their off-normal positions.

5.2.2.6 <u>Local Leak Rate Testing of Containment Isolation Valves</u>

Local leak rate testing of containment isolation valves is performed in accordance with Technical Specification 5.5.15. The Containment Leak Rate Program is in accordance with the guidance contained in Regulatory Guide 1.163, except as noted in the Technical Specification.

Amendment No. 195 to the Technical Specifications relocated information concerning containment isolation valves from the Technical Specifications to the FSAR.

Subsequent to implementation of Option B of 10 CFR 50, Appendix J, a third-party review of NYPA's (Option B) implementation program was completed. That review, confirmed by NYPA Nuclear Safety Evaluation, determined the scope of the Appendix J, "Type C," Local Leak Rate Test Program was greater than required by regulation. Specifically:

1) Leakage testing of SI-MOV-888A and B and SI-MOV-1835A and B is not required. The valves are not required to be LLR tested for purposes of compliance with Appendix J. These valves do not represent potential primary containment atmospheric leak paths following a single active failure. Since IVSWS nitrogen will only be applied to these valves in the event of a passive failure, but in no case sooner than 24 hours post-LOCA, there are no requirements for performance of leak rate tests.

2) Leak rate testing of the remaining valves penetrations served by the high-pressure nitrogen sub-system IVSWS is not required for compliance with Appendix J. Continued testing is required to ensure adequate nitrogen supply to the affected CIVs for the initial twenty-four hour period following a LOCA.

3) Local leak rate (Type C) testing of SI-1814 A, B, and C is not required for compliance with Appendix J. These valves do not represent potential primary containment atmospheric leak paths following a single active failure.

   Note: It is understood the body and packing of the SI-1814 A, B, and C valves are an extension of the containment pressure boundary and are exposed to containment pressure during Type A tests. If that pressure boundary is "broken," i.e., to facilitate calibration of the transmitters, then appropriate testing will be performed to confirm the integrity of the pressure boundary.

4) Local leak rate (LLR) testing of AC-741 is not required for compliance with Appendix J. This valve does not represent a potential primary containment atmospheric leak path following a single active failure.

5) Local leak rate testing of SI-MOV-885 A & B is not required for compliance with Appendix J. These valves do not represent potential primary containment atmospheric leak paths following a single active failure.

6) LLR testing of the SES CIVs is not required for compliance with Appendix J. Continued testing is required to assure the potential for in-leakage of service water into the

containment following a postulated breach of the SWS integrity during the long-term post-LOCA recovery phase is within analyzed limits.

5.2.2.7 Containment Isolation During Refueling Outage

The Outage Equipment Hatch (OEH) may be used during an outage, when the permanent Equipment Hatch is removed.

The OEH will maintain containment closure during core alterations and during movement of irradiated fuel assemblies within containment building. The OEH may provide penetrations for temporary services and personnel access during an outage. The penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side and are subject to the requirements of ITS 3.9.3.

Electric penetrations used will be verified for leakage and integrity of the pressure boundary connection and not its function.

The roll-up door is an alternate device that is capable of rapid closure. It is effectively an airtight, but not pressure-resistant, door that when closed prevents direct communication between the containment atmosphere and the outside atmosphere.

Subsequent to a loss of RHR cooling as defined in ITS 3.9.4 and 3.9.5, the roll-up door provides rapid containment closure until either cooling is restored, or the main equipment hatch (or OEH) may be installed within four hours.

TABLE 5.2-1

CONTAINMENT CONTROL ISOLATION VALVES

Valve Type	Actuator	Closure*
1500# Globe	Reverse Diaphragm	6 sec
1500# Globe	Motor	10 sec
1500# D.D.V.	Motor	10/30 sec
150# Gate	Motor	10 sec
150# Saunders	Direct Diaphragm	2 sec
150# Saunders	Reverse Diaphragm	2 sec
150# Globe	Solenoid	1.5 sec
150# D.D.V.	Motor	10 sec
150# Globe	Reverse Diaphragm	6 sec
150# Butterfly	Air & Spring	2 sec
600# Plug	Air Piston	4 sec
150# Butterfly	Air Piston	3.5 sec
300# Gate (RHR V 744)	Motor	30 sec
150# Gate (Aux Coolant V 769 and V 797)	Motor	30 sec

*Note: Closure times listed are general closure times for valve types shown. They do not form the basis of the safety analysis and are not used in determining valve stroke criteria.

TABLE 5.2-2

NORMALLY CLOSED ISOLATION VALVES

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TABLE 5.2-3
(Sheet 1 of 9)
CONTAINMENT PIPING PENETRATIONS AND VALVING

Numbers shown in brackets () refer to footnotes

FIGURE NO.	SERVICE AND PENETRATION	VALVE ID or CLOSED SYSTEM	PENET CLASS (1)	VALVE TYPE	OPER. TYPE	PWR. FAIL POSITION	CONT. ISOL. TRIP	POSITION INDIC. CONT. RM	FLUID GAS / WTR.	PENETR. DESIGN (25)	NORM. POSITION	SHUT-DOWN POSITION	POST ACCID. POSITION	POST ACCID. USAGE	SEALING METHOD	MIN. TEST PRESS. (psig)	TEST FLUID (16)
5.2-1	PRESSURIZER RELIEF TANK TO GAS ANALYZER Penetration “V”	RC-AOV-549 RC-AOV-548	1	GLOBE GLOBE	AIR AIR	FC FC	T T	Yes Yes	G	H	C C	O O	C C	No No	Water (A)(4) Water (A)(4)	47	W W
5.2-1	PRESSURIZER RELIEF TANK N_2 SUPPLY Penetration “Y”	RC-518 RC-AOV-550	3	CHECK DIA.	- AIR	- FC	- T	No Yes	G	C	- O	- O	- C	No No	- -	43	G G
5.2-1	PRESSURIZER RELIEF TANK MAKE-UP Penetration “Y”	RC-AOV-552 RC-AOV-519	3	DIA. DIA.	AIR AIR	FC FC	T T	Yes Yes	W	C	C(9) C(9)	C C	C C	No No	Water (A)(4) Water (A)(4)	47	W W
5.2-2	RESIDUAL HEAT REMOVAL RETURN Penetration “J”	AC-741 AC-MOV-744	6	CHECK DDV	- MOTOR	- FAI	- -	No Yes	W	H	- O(8)	- O	- O	No Yes	(5) Nitro(M)(32)	N/A 43 (15)	N/A N
5.2-2	RESID. HEAT REMOVAL LOOP TO SI PUMPS Penetration “QQ”	SI-MOV-888A SI-MOV-888B CS	6	DDV DDV -	MOTOR MOTOR -	FAI FAI -	- - -	Yes Yes	W	H	C(8) C(8)	LC(28) LC(28)	O O	Yes Yes	Nitro(M)(31) Nitro(M)(31)	N/A	N/A N/A
5.2-2	RESID. HEAT REMOVAL LOOP TO SAMPLING SYS. Penetration “QQ”	SP-AOV-958 SP-AOV-959 SP-990C	6	GLOBE GLOBE GLOBE	AIR AIR MANUAL	FC FC -	T T -	Yes Yes No	W	H	C C LC(8)	C(12) C(12) C(12)	C(12) C(12) C(12)	No(12) No(12) No(12)	Nitro(M)(32) Nitro(M)(32) Nitro(M)(32)	50	N N N
5.2-2	RESID. HEAT REMOVAL LOOP TO RHR PUMP MINIFLOW Penetration “QQ”	AC-MOV-1870 AC-MOV-743	6	GLOBE GATE	MOTOR MOTOR	FAI FAI	- -	Yes Yes	W	H	LTh(8) O(8)	LTh O	O O	Yes Yes	Nitro(M)(32) Nitro(M)(32)	50	N N
5.2-2	RESID. HEAT REMOVAL LOOP OUT Penetration “K”	AC-732	6	DDV	MANUAL	-	-	No	W	H	LC(8)	O	C	No	Nitro. (M)(32)	50 (15)	N
5.2-2	CONTAINMENT SUMP RECIRC. LINE Penetration “OO”	SI-MOV-885A SI-MOV-885B	5	DDV(23) DDV(23)	MOTOR MOTOR	FAI FAI	- -	Yes Yes	W	H	C(8) C(8)	C LC(28)	C(18) C(18)	No(18) No(18)	(5) (5)	N/A	N/A N/A
5.2-3	LETDOWN LINE Penetration “X”	CH-AOV-201 CH-AOV-202 CS	1	GLOBE GLOBE -	AIR AIR -	FC FC -	T T -	Yes Yes -	W	H	O O	C(9) C(9)	C C	No No	Water (A)(4) Water (A)(4)	47	W W
5.2-3	CHARGING LINE Penetration “R”	CH-MOV-205 CH-MOV-226 CH-227 CS	3	GATE GATE GLOBE -	MOTOR MOTOR MANUAL -	FAI FAI - -	- - - -	No No No -	W	C	O(8) O(8) LC(8)	C(9) C(9) C	C C C	No No No	Water(M)(4) Water(M)(4) Water(M)(4)	47	W W W
5.2-4	REACTOR COOLANT PUMP SEAL WATER SUPPLY LINES Penetration “Z”	CH-MOV-250A CH-MOV-250B CH-MOV-250C CH-MOV-250D CH-MOV-441 CH-MOV-442 CH-MOV-443 CH-MOV-444	3	GLOBE GLOBE GLOBE GLOBE GLOBE GLOBE GLOBE GLOBE	MOTOR MOTOR MOTOR MOTOR MOTOR MOTOR MOTOR MOTOR	FAI FAI FAI FAI FAI FAI FAI FAI	- - - - - - - -	No No No No No No No No	W	C	O(8) O(8) O(8) O(8) O(8) O(8) O(8) O(8)	C(9) C(9) C(9) C(9) C(9) C(9) C(9) C(9)	C(11) C(11) C(11) C(11) C(11) C(11) C(11) C(11)	No(11) No(11) No(11) No(11) No(11) No(11) No(11) No(11)	Water(M)(4) Water M)(4) Water(M)(4) Water(M)(4) Water(M)(4) Water(M)(4) Water(M)(4) Water(M)(4)	47	W W W W W W W W

TABLE 5.2-3
(Sheet 2 of 9)
CONTAINMENT PIPING PENETRATIONS AND VALVING

Numbers shown in brackets () refer to footnotes

FIGURE NO.	SERVICE AND PENETRATION	VALVE ID or CLOSED SYSTEM	PENET CLASS (1)	VALVE TYPE	OPER. TYPE	PWR. FAIL POSITION	CONT. ISOL. TRIP	POSITION INDIC. CONT. RM	FLUID GAS / WTR.	PENETR DESIGN (25)	NORM. POSITION	SHUT-DOWN POSITION	POST ACCID. POSITION	POST ACCID. USAGE	SEALING METHOD	MIN. TEST PRESS. (psig)	TEST FLUID (16)
5.2-4	REACTOR COOLANT PUMP SEAL WATER RETURN Penetration “R”	CH-MOV-222	2	DDV	MOTOR	FAI	P	Yes	W	C	O(11)	O	C(11)	No(11)	Water (M)(4)	47	W
5.2-5	REACTOR COOLANT SYSTEM SAMPLE LINES Penetration “W”	SP-AOV-956E SP-AOV-956F	1	GLOBE GLOBE	AIR AIR	FC FC	T T	Yes Yes	W	H	O O	C C	C C	No No	Water (A)(4) Water (A)(4)	47	W
5.2-5	FUEL TRANSFER TUBE Penetration “HH”	-	6	BLIND FLANGE (27)	-	-	-	-	W	H	-	-	-	-	(17)	-	-
5.2-6	CONTAINMENT SPRAY HEADERS Penetrations “GG” and “P”	SI-869A SI-869B SI-867A SI-867B SI-878A SI-878B	3	DDV DDV CHECK CHECK GLOBE GLOBE	MANUAL MANUAL - - MANUAL MANUAL	- - - - - -	- - - - - -	No No No No No No	W	C	LO(8) LO(8) - - LC(8) LC(8)	C C - - C C	O O - - C C	Yes Yes Yes Yes Yes Yes	Water(M)(4) Water(M)(4) - - - -	47 47 43 43 43 43	W W G G G G
5.2-7	SAFETY INJECTION HEADERS Penetrations “Q” and “NN”	SI-MOV-1835A SI-MOV-1835B SI-MOV-851A SI-MOV-850C SI-MOV-850A	3	DDV DDV DDV GATE GATE	MOTOR MOTOR MOTOR MOTOR MOTOR	FAI FAI FAI FAI FAI	S S - - -	Yes Yes Yes Yes Yes	W	H	O(8) O(8) O(8) LO(8) LO(8)	C C C C C	O(19) O(19) O(19) O(19) O(19)	Yes(33) Yes(33) Yes(19) Yes(19) Yes(19)	Nitro.(M)(33) Nitro.(M)(33) Water (M)(4) Water (M)(4) Water (M)(4)	N/A N/A 47 47 47	N/A N/A W W W
5.2-7	SAFETY INJECTION TEST Penetration “Y”	SI-859A SI-859C	5	GLOBE GLOBE	MANUAL MANUAL	- -	- -	No No	W	C	LC(8) LC(8)	C C	C C	No No	Water (A)(4) Water (A)(4)	47 47	W W
5.2-8	ACCUMULATOR NITROGEN SUPPLY Penetration “RR”	NNE-1610 NNE-AOV-863	5	CHECK GLOBE	- AIR	- FC	- T	No Yes	G	C	- C(9)	- C	- C	No No	- -	43 43	G G
5.2-8	ACCUMULATOR SAMPLE Penetration “RR”	SP-AOV-956G SP-AOV-956H	2	GLOBE GLOBE	AIR AIR	FC FC	T T	Yes Yes	W	C	C(12) C(12)	C C	C(12) C(12)	No(12) No(12)	Water (A)(4) Water (A)(4)	47 47	W W
5.2-9	PRIMARY SYSTEM VENT AND NITROGEN SUPPLY Penetration “V”	WD-AOV-1786 WD-AOV-1787 WD-AOV-1610 WD-1616	2 3	DIA DIA . DIA. CHECK	AIR AIR AIR -	FC FC FC -	T T T -	Yes Yes Yes No	G	H	O(9) O(9) O -	C C O -	C C C -	No No No -	Water (A)(4) Water (A)(4) - -	47 47 43 43	W W G G
5.2-9	REACTOR COOLANT DRAIN TK. TO GAS ANALYZER Penetration “V”	WD-AOV-1788 WD-AOV-1789	2	DIA. DIA.	AIR AIR	FC FC	T T	Yes Yes	G	H	C(13) C(13)	O C(13)	C C	No No	Water (A)(4) Water (A)(4)	47 47	W W
5.2-9	RCDT PUMP DISCHARGE Penetration “Z”	WD-AOV-1702 WD-AOV-1705	2	DIA. DIA.	AIR AIR	FC FC	T T	Yes Yes	W	C	C(9) C(9)	O O	C C	No No	Water (A)(4) Water (A)(4)	47 47	W W
5.2-10	REACTOR COOLANT PUMP COOLING WATER IN Penetration “N”	AC-MOV-797 AC-MOV-769	3	GATE GATE	MOTOR MOTOR	FAI FAI	P P	Yes Yes	W	C	O(11) O(11)	C(11) C(11)	C(11) C(11)	No(11) No(11)	Water (M)(4) Water (M)(4)	47 47	W W

TABLE 5.2-3
(Sheet 3 of 9)
CONTAINMENT PIPING PENETRATIONS AND VALVING

Numbers shown in brackets () refer to footnotes

FIGURE NO.	SERVICE AND PENETRATION	VALVE ID or CLOSED SYSTEM	PENET CLASS (1)	VALVE TYPE	OPER. TYPE	PWR. FAIL POSITION	CONT. ISOL. TRIP	POSITION INDIC. CONT. RM	FLUID GAS / WTR.	PENETR DESIGN (25)	NORM. POSITION	SHUT-DOWN POSITION	POST ACCID. POSITION	POST ACCID. USAGE	SEALING METHOD	MIN. TEST PRESS. (psig)	TEST FLUID (16)
5.2-10	REACTOR COOLANT PUMP COOLING WATER OUT 6" Penetration "O"	AC-MOV-784 AC-MOV-786	2	GATE GATE	MOTOR MOTOR	FAI FAI	P P	Yes Yes	W	C	O(11) O(11)	C(11) C(11)	C(11) C(11)	No(11) No(11)	Water (M)(4) Water (M)(4)	47 47	W W
5.2-10	REACTOR COOLANT PUMP COOLING WATER OUT 3" Penetration "O"	AC-FCV-625 AC-MOV-789	2	GATE GATE	MOTOR MOTOR	FAI FAI	P P	Yes Yes	W	C	O(11) O(11)	C(11) C(11)	C(11) C(11)	No(11) No(11)	Water (M)(4) Water (M)(4)	47 47	W W
5.2-11	RESIDUAL HEAT EXCHANGERS COOLING WATER IN Penetrations "KK" and "VV"	AC-751A AC-751B CS	4	CHECK CHECK -	- - -	- - -	- - -	No No -	W	C	- -	- -	- -	Yes Yes	- -	N/A N/A	N/A N/A
5.2-11	RESIDUAL HEAT EXCHANGERS COOLING WATER RETURN Penetrations "JJ" and "UU"	AC-MOV-822A AC-MOV-822B CS	4	GATE GATE -	MOTOR MOTOR -	FAI FAI -	S S -	Yes Yes -	W	C	C(8) C(8)	O O	O O	Yes Yes	- -	N/A N/A	N/A N/A
5.2-12	RECIRC. PUMP COOLING WATER SUPPLY Penetration "LL"	AC-752F AC-753F CS	4	GLOBE GLOBE -	MANUAL MANUAL -	- - -	- - -	No No -	W	C	O(8) O(8)	O O	O O	Yes Yes	- -	N/A N/A	N/A N/A
5.2-12	RECIRC. PUMP COOLING WATER RETURN Penetration "LL"	AC-752J AC-753J CS	4	GLOBE GLOBE -	MANUAL MANUAL -	- - -	- - -	No No -	W	C	O(8) O(8)	O O	O O	Yes Yes	- -	N/A N/A	N/A N/A
5.2-13	EXCESS LETDOWN HEAT EXCHANGER COOLING WATER IN Penetration "U"	AC-AOV-791 AC-AOV-798	4	DIA. DIA.	AIR AIR	FC FC	T T	Yes Yes	W	C	C(9) C(9)	O O	C C	No No	Water (A)(4) Water (A)(4)	47 47	W W
5.2-13	EXCESS LETDOWN HEAT EXCHANGER COOLING WATER OUT Penetration "R"	AC-AOV-796 AC-AOV-793	4	GLOBE DIA.	AIR AIR	FC FC	T T	Yes Yes	W	C	C(9) C(9)	O O	C C	No No	Water (A)(4) Water (A)(4)	47 47	W W
5.2-13	CONTAINMENT SUMP PUMP DISCHARGE Penetration "Y"	WD-AOV-1728 WD-AOV-1723	2	DIA. DIA.	AIR AIR	FC FC	T T	Yes Yes	W	C	O O	O O	C C	No No	Water (A)(4) Water (A)(4)	47 47	W W
5.2-14	CONTAINMENT AIR SAMPLE IN- RAD. MONITORING SYSTEM Penetration "RR"	VS-PCV-1234 VS-PCV-1235	6	DIA. DIA.	AIR AIR	FC FC	T T	Yes Yes	G	C	O O	O O	C(20) C(20)	No(20) No(20)	Air (A)(7) Air (A)(7)	43 43	G G
5.2-14	CONTAINMENT AIR SAMPLE OUT – RAD. MONITORING SYSTEM Penetration "RR"	VS-PCV-1236 VS-PCV-1237	6	DIA. DIA.	AIR AIR	FC FC	T T	Yes Yes	G	C	O O	O O	C(20) C(20)	No(20) No(20)	Air (A)(7) Air (A)(7)	43 43	G G
5.2-14	AIR EJECTOR DISCHARGE TO CONTAINMENT Penetration "R"	CA-PCV-1229 CA-PCV-1230	6	GLOBE GLOBE	AIR AIR	FC FC	T T	Yes Yes	G	C	C C	C C	C C	No No	Air (A)(7) Air (A)(7)	43 43	G G

TABLE 5.2-3
(Sheet 4 of 9)
CONTAINMENT PIPING PENETRATIONS AND VALVING

Numbers shown in brackets () refer to footnotes

FIGURE NO.	SERVICE AND PENETRATION	VALVE ID or CLOSED SYSTEM	PENET CLASS (1)	VALVE TYPE	OPER. TYPE	PWR. FAIL POSITION	CONT. ISOL. TRIP	POSITION INDIC. CONT. RM	FLUID GAS / WTR.	PENETR DESIGN (25)	NORM. POSITION	SHUT-DOWN POSITION	POST ACCID. POSITION	POST ACCID. USAGE	SEALING METHOD	MIN. TEST PRESS. (psig)	TEST FLUID (16)
5.2-15	MAIN STEAM HEADERS Penetrations "A,B,C and D"	CS	7	-	-	-	-	-	G	H	-	-	(22)	Yes(22)	-	-	-
	MAIN STEAM TO AUX. FW PUMP TURBINE	CS	-	-	-	-	-	-	G		-	-	-	Yes	-	-	-
5.2-15	MAIN FEEDWATER HEADERS Penetrations "E,F,G and H"	CS	7	-	-	-	-	-	W	H	-	-	-	Yes	-	-	-
	AUXILIARY FW TURBINE DRIVEN	CS	-	-	-	-	-	-	W		-	-	-	Yes	-	-	-
	AUXILIARY FW MOTOR DRIVEN	CS	-	-	-	-	-	-	W		-	-	-	Yes	-	-	-
5.2-15	STEAM GENERATOR BLOWDOWN Penetrations "AA,BB, CC, and DD"	BD-PCV-1214		GLOBE	AIR	FC	T	Yes			O	C	C	No	Water (A)(4)	47	W
		BD-PCV-1215	2	GLOBE	AIR	FC	T	Yes			O	C	C	No	Water (A)(4)	47	W
		BD-PCV-1216		GLOBE	AIR	FC	T	Yes			O	C	C	No	Water (A)(4)	47	W
		BD-PCV-1217		GLOBE	AIR	FC	T	Yes	W	H	O	C	C	No	Water (A)(4)	47	W
		BD-PCV-1214A		GLOBE	AIR	FC	T	Yes			O	C	C	No	Water (A)(4)	47	W
		BD-PCV-1215A		GLOBE	AIR	FC	T	Yes			O	C	C	No	Water (A)(4)	47	W
		BD-PCV-1216A		GLOBE	AIR	FC	T	Yes			O	C	C	No	Water (A)(4)	47	W
		BD-PCV-1217A		GLOBE	AIR	FC	T	Yes			O	C	C	No	Water (A)(4)	47	W
5.2-15	STEAM GENERATOR BLOWDOWN SAMPLE Four Lines @ Penetration "W"	BD-PCV-1223	2	GLOBE	AIR	FC	T	Yes			O	C	C	No	Water (A)(4)	47	W
		BD-PCV-1224		GLOBE	AIR	FC	T	Yes			O	C	C	No	Water (A)(4)	47	W
		BD-PCV-1225		GLOBE	AIR	FC	T	Yes	W	H	O	C	C	No	Water (A)(4)	47	W
		BD-PCV-1226		GLOBE	AIR	FC	T	Yes			O	C	C	No	Water (A)(4)	47	W
		BD-PCV-1223A		GLOBE	AIR	FC	T	Yes			O	C	C	No	Water (A)(4)	47	W
		BD-PCV-1224A		GLOBE	AIR	FC	T	Yes			O	C	C	No	Water (A)(4)	47	W
		BD-PCV-1225A		GLOBE	AIR	FC	T	Yes			O	C	C	No	Water (A)(4)	47	W
		BD-PCV-1226A		GLOBE	AIR	FC	T	Yes			O	C	C	No	Water (A)(4)	47	W
5.2-16	VENTILATION SYSTEM COOLING WATER IN Penetrations "La,Lb,Lc, Ld and Le"	SWN-41-1		BV	MANUAL	-	-	No			O(8)	O	O	Yes	(6)	47	W
		SWN-41-2		BV	MANUAL	-	-	No			O(8)	O	O	Yes	(6)	47	W
		SWN-41-3		BV	MANUAL	-	-	No	W	C	O(8)	O	O	Yes	(6)	47	W
		SWN-41-4		BV	MANUAL	-	-	No			O(8)	O	O	Yes	(6)	47	W
		SWN-41-5		BV	MANUAL	-	-	No			O(8)	O	O	Yes	(6)	47	W
		SWN-43-1		GATE	MANUAL	-	-	No			C(8)	C	C	No	(6)	47	W
		SWN-43-2		GATE	MANUAL	-	-	No			C(8)	C	C	No	(6)	47	W
		SWN-43-3	4	GATE	MANUAL	-	-	No			C(8)	C	C	No	(6)	47	W
		SWN-43-4		GATE	MANUAL	-	-	No			C(8)	C	C	No	(6)	47	W
		SWN-43-5		GATE	MANUAL	-	-	No			C(8)	C	C	No	(6)	47	W
		SWN-42-1		RV	-	-	-	No			-	-	-	-	(6)	47	W
		SWN-42-2		RV	-	-	-	No			-	-	-	-	(6)	47	W
		SWN-42-3		RV	-	-	-	No			-	-	-	-	(6)	47	W
		SWN-42-4		RV	-	-	-	No			-	-	-	-	(6)	47	W
		SWN-42-5		RV	-	-	-	No			-	-	-	-	(6)	47	W
		CS		-	-	-	-	-			-	-	-	-			

TABLE 5.2-3
(Sheet 5 of 9)
CONTAINMENT PIPING PENETRATIONS AND VALVING

Numbers shown in brackets () refer to footnotes

FIGURE NO.	SERVICE AND PENETRATION	VALVE ID or CLOSED SYSTEM	PENET CLASS (1)	VALVE TYPE	OPER. TYPE	PWR. FAIL POSITION	CONT. ISOL. TRIP	POSITION INDIC. CONT. RM	FLUID GAS / WTR.	PENETR DESIGN (25)	NORM. POSITION	SHUT-DOWN POSITION	POST ACCID. POSITION	POST ACCID. USAGE	SEALING METHOD	MIN. TEST PRESS. (psig)	TEST FLUID (16)
5.2-16	VENTILATION SYSTEM	SWN-44-1		BV	MANUAL	-	-	No			LTh(8)	LTh	LTh	Yes	(6)	47	W
	COOLING WATER OUT	SWN-44-2		BV	MANUAL	-	-	No	W	C	LTh(8)	LTh	LTh	Yes	(6)	47	W
	Penetrations “Ma, Mb,	SWN-44-3		BV	MANUAL	-	-	No			LTh(8)	LTh	LTh	Yes	(6)	47	W
	Mc, Md, Me, and SS”	SWN-44-4		BV	MANUAL	-	-	No			LTh(8)	LTh	LTh	Yes	(6)	47	W
		SWN-44-5		BV	MANUAL	-	-	No			LTh(8)	LTh	LTh	Yes	(6)	47	W
		SWN-51-1		GATE	MANUAL	-	-	No			O(8)	O	O	Yes	(6)	47	W
		SWN-51-2	4	GATE	MANUAL	-	-	No			O(8)	O	O	Yes	(6)	47	W
		SWN-51-3		GATE	MANUAL	-	-	No			O(8)	O	O	Yes	(6)	47	W
		SWN-51-4		GATE	MANUAL	-	-	No			O(8)	O	O	Yes	(6)	47	W
		SWN-51-5		GATE	MANUAL	-	-	No			O(8)	O	O	Yes	(6)	47	W
		SWN-71-1		GLOBE	MANUAL	-	-	No			Th(8)	Th	Th	Yes	(6)	47	W
		SWN-71-2		GLOBE	MANUAL	-	-	No			Th(8)	Th	Th	Yes	(6)	47	W
		SWN-71-3		GLOBE	MANUAL	-	-	No			Th(8)	Th	Th	Yes	(6)	47	W
		SWN-71-4		GLOBE	MANUAL	-	-	No			Th(8)	Th	Th	Yes	(6)	47	W
		SWN-71-5		GLOBE	MANUAL	-	-	No			Th(8)	Th	Th	Yes	(6)	47	W
		CS		-	-	-	-	-									
5.2-17	STATION AIR Penetration “Y”	SA-24-1	3	DIA.	MANUAL	-	-	No	G	C	LC(8)	LC(8)	LC	No	Water (A)(4)	47	W
		SA-24-2		DIA.	MANUAL	-	-	No			LC(8)	LC(8)	LC	No	Water (A)(4)	47	W
5.2-17	WELD CHANNEL	PS-PCV-1111-1	4	BALL	MANUAL	-	-	No			LO(8)	LO	LO	Yes	(17)	N/A	N/A
	PENETRATION	PS-PCV-1111-2		BALL	MANUAL	-	-	No	G	C	LO(8)	LO	LO	Yes	(17)	N/A	N/A
	PRESSURE SYSTEM	CS (inside)		-	-	-	-	-			-	-	-	-			
	Penetration “Y”	CS (outside)		-	-	-	-	-			-	-	-	-			
5.2-19	PURGE SUPPLY DUCT VENTILATION Penetration “EE”	VS-FCV-1170	6	BV	AIR	FC	T (2)	Yes	G	C	C	O	C	No	Air (A)(7)	43	G
		VS-FCV-1171		BV	AIR	FC	T (2)	Yes			C	O	C	No	Air (A)(7)	43	G
5.2-19	PURGE EXHAUST DUCT VENTILATION Penetration “FF”	VS-FCV-1172	6	BV	AIR	FC	T (2)	Yes	G	C	C	O	C	No	Air (A)(7)	43	G
		VS-FCV-1173		BV	AIR	FC	T (2)	Yes			C	O	C	No	Air (A)(7)	43	G
5.2-19	CONTAINMENT	VS-PCV-1190		BV	AIR	FC	T (2)	Yes	G	C	C(14)	C	C	No	Air (A)(7)	43	G
	PRESSURE RELIEF	VS-PCV-1191	6	BV	AIR	FC	T (2)	Yes			C(14)	C	C	No	Air (A)(7)	43	G
	VENTILATION Penetration “PP”	VS-PCV-1192		BV	AIR	FC	T (2)	Yes			C(14)	C	C	No	Air (A)(7)	43	G
5.2-20	RECIRCULATION PUMP DISCHARGE SAMPLE LINE Penetration “TT”	SP-MOV-990A	6	GATE	MOTOR	FAI	-	No	W	C	LC(8)	C	LC (12)	No	Nitro(M)(32)	50	N
		SP-MOV-990B		GATE	MOTOR	FAI	-	No			LC(8)	C	LC (12)	No	Nitro(M)(32)	50	N
5.2-20	PRESSURIZER STEAM SAMPLE LINE Penetration “W”	SP-AOV-956A	1	GLOBE	AIR	FC	T	Yes	W	H	C	C	C	No	Water (A)(4)	47	W
		SP-AOV-956B		GLOBE	AIR	FC	T	Yes			C	C	C	No	Water (A)(4)	47	W
5.2-20	PRESSURIZER LIQUID SAMPLE LINE Penetration “W”	SP-AOV-956C	1	GLOBE	AIR	FC	T	Yes	W	H	C	C	C	No	Water (A)(4)	47	W
		SP-AOV-956D		GLOBE	AIR	FC	T	Yes			C	C	C	No	Water (A)(4)	47	W

TABLE 5.2-3
(Sheet 6 of 9)
CONTAINMENT PIPING PENETRATIONS AND VALVING

Numbers shown in brackets () refer to footnotes

FIGURE NO.	SERVICE AND PENETRATION	VALVE ID or CLOSED SYSTEM	PENET CLASS (1)	VALVE TYPE	OPER. TYPE	PWR. FAIL POSITION	CONT. ISOL. TRIP	POSITION INDIC. CONT. RM	FLUID GAS / WTR.	PENETR DESIGN (25)	NORM. POSITION	SHUT-DOWN POSITION	POST ACCID. POSITION	POST ACCID. USAGE	SEALING METHOD	MIN. TEST PRESS. (psig)	TEST FLUID (16)
5.2-21	CONTAINMENT PRESSURE INSTRUMENTATION LINE Penetration "RR"	SI-1814A CS	6	GLOBE -	MANUAL -	- -	- -	No -	G	C	LO(8)	O	O	Yes	- -	(34)	N/A
5.2-21	CONTAINMENT PRESSURE INSTRUMENTATION LINE Penetration "LL"	SI-1814B CS	6	GLOBE -	MANUAL -	- -	- -	No -	G	C	LO(8)	O	O	Yes	- -	(34)	N/A
5.2-21	CONTAINMENT PRESSURE INSTRUMENTATION LINE Penetration "O"	SI-1814C CS	6	GLOBE -	MANUAL -	- -	- -	No -	G	C	LO(8)	O	O	Yes	- -	(34)	N/A
5.2-22	POST ACCIDENT CONTAINMENT SAMPLING SUPPLY AND RETURN LINES Penetrations "R, TT, LL, Z, and O"	SP-SOV-506 SP-SOV-507 SP-SOV-508 SP-SOV-512 SP-SOV-513 SP-SOV-511 SP-SOV-516 SP-SOV-509 SP-SOV-510 SP-SOV-514 SP-SOV-515	5	GLOBE GLOBE GLOBE GLOBE GLOBE GLOBE GLOBE GLOBE GLOBE GLOBE GLOBE	SOL. SOL. SOL. SOL. SOL. SOL. SOL. SOL. SOL. SOL. SOL.	FC FC FC FC FC FC FC FC FC FC FC	T(10) T(10) T(10) T(10) T(10) T(10) T(10) T(10) T(10) T(10) T(10)	Yes Yes Yes Yes Yes Yes Yes Yes Yes Yes Yes	G	C	C(8) C(8) C(8) C(8) C(8) C(8) C(8) C(8) C(8) C(8) C(8)	C C C C C C C C C C C	C(12) C(12) C(12) C(12) C(12) C(12) C(12) C(12) C(12) C(12) C(12)	Yes (12) Yes (12) Yes (12) Yes (12) Yes (12) Yes (12) Yes (12) Yes (12) Yes (12) Yes (12) Yes (12)	Air (A)(7) Air (A)(7) Air (A)(7) Air (A)(7) Air (A)(7) Air (A)(7) Air (A)(7) Air (A)(7) Air (A)(7) Air (A)(7) Air (A)(7)	43 43 43 43 43 43 43 43 43 43 43	G G G G G G G G G G G
-	CONTAINMENT SUMP RECIRCULATION (SPARE) Penetration "O_1 O_1"	CS (3)	6	-	-	-	-	-	-	C	-	-	-	-	(17)	-	-
5.2-25	INSTRUMENT AIR – P. A. VENTING SYSTEM SUPPLY Penetration "Y"	IA-39 IA-PCV-1228	6	CHECK DIA.	- AIR	- FC	- T	No Yes	G	C	- O	- O	- C (24)	No No (24)	- -	43 43	G G
5.2-25	POST ACCIDENT VENTING SYSTEM EXHAUST LINE Penetration "LL"	PS-7 PS-8 PS-9 PS-10	5	DIA. DIA. DIA. DIA.	MANUAL MANUAL MANUAL MANUAL	- - - -	- - - -	No No No No	G	C	LC(8) LC(8) LC(8) LC(8)	LC LC LC LC	C (24) C (24) C (24) C (24)	No (24) No (24) No (24) No (24)	(17) (17) (17) (17)	43 43 43 43	G G G G
5.2-26	CONTAINMENT LEAK TEST INSTRUMENT SENSOR LINE Three lines @ Penetration "RR"	CS (3)	-	-	-	-	-	-	G	C	-	-	-	No	(17)	-	-
5.2-26	CONTAINMENT LEAK TEST AIR LINE Penetrations "XX and YY"	CS (3)	6	-	-	-	-	-	G	C	-	(30)	-	No	(17)	-	-
5.2-27	EQUIPMENT ACCESS	CB-7 CB-8 CB-5 CB-6	6	BALL BALL CHECK(26) CHECK(26)	MANUAL MANUAL - -	- - - -	- - - -	(29)	G	C	C(8) C(8) - -	C C - -	C C – -	No No	(17) (17) - -	43 43 43 43	G G G G
5.2-27A	PERSONNEL AIR LOCK	CB-3 CB-4 CB-1 CB-2	6	BALL BALL CHECK(26) CHECK(26)	MANUAL MANUAL - -	- - - -	- - - -	(29)	G	C	C(8) C(8) - -	C C - -	C C - -	No No	(17) (17) - -	43 43 43 43	G G G G
5.2-28	DEMIN. WTR. INTO CONTAINMENT Penetration "Y"	DW-AOV-1 DW-AOV-2	6	PLUG PLUG	AIR AIR	FC FC	T T	Yes Yes	W	C	C C	C(21) C(21)	C C	No No	Water (A)(4) Water (A)(4)	47 47	W W

TABLE 5.2-3
(Sheet 7 of 9)
CONTAINMENT PIPING PENETRATIONS AND VALVING

Numbers shown in brackets () refer to footnotes

ABBREVIATIONS:

A	Automatic
AMB	Ambient
BV	Butterfly Valve
C	Cold
CS	Closed System
COL	Check Off List
DDV	Double Disc Gate Valve
DIA	Diaphragm Valve
FAI	Fail As Is
FC	Fail Closed
FO	Fail Open
G	Gas
H	Hot
LC	Locked Closed
LO	Locked Open
LTh	Locked Throttled
M	Manual
N	Nitrogen
POP	Plant Operating Procedures
P	Containment Isolation Signal Phase B
T	Containment Isolation Signal Phase A
Th	Throttled
RV	Relief Valve
S	Safety Injection Signal (Opens valves on SI signal)
SOP	System Operating Procedures
SOL	Solenoid Operated Valves
W	Water

TABLE 5.2-3
(Sheet 8 of 9)
CONTAINMENT PIPING PENETRATIONS AND VALVING

Numbers shown in brackets () refer to footnotes

DEFINITIONS:

NORMAL POSITION: Defined as RCS operation above 200^{0}F to Full Power. Valve positions as defined by POP's, SOP's, and COL's

SHUTDOWN POSITION: Defined as RCS 200^{0}F and below, not in refueling and not at reduced inventory. Valve positions as defined by POP's and SOP's.

POST ACCIDENT POSITION: Defined as SI with Phase A and B isolation. Note: valve position may differ based on the accident in progress (i.e. phase B may not be required).

POST ACCIDENT USAGE: Defined as Design Basis Accident valve usage based on position during long term recirculation, assuming no failures. Note: valve position may differ based on the accident in progress, equipment failure, and recommendations during the recovery phase.

NOTES:

1.Penetration class is described in subsection 5.2.2.	**16**. Test Fluid "G" signifying Gas indicates either air or nitrogen as test medium.
2.Also tripped closed by high radiation in containment.	**17**. Seal air via WCCPP, continuously pressurized.
3.Penetration sealed at both ends.	**18.** May be opened Post Accident if normal path from recirc. pumps not available.
4. Sealed by Isolation Valve Seal Water System.	**19.** Valves may be closed Post Accident if not in service.
5. "Sealed" by Residual Heat Removal System or recirculation sump fluid. Not a "seal system" as defined in 10 CFR 50, Appendix J.	**20.** May be opened Post Accident when the containment pressure is below 5 psig.
6. "Sealed" by Service Water System fluid. Not a "seal system" as defined in 10 CFR 50, Appendix J. LLR testing is not required for Appendix J compliance but is required to limit in-leakage to the containment given a postulated breech of SWS integrity during the long-term recovery phase.	**21.** Valves may be opened for maintenance.
7. Sealed by Weld Channel and Containment Penetration Pressurization System.	**22.** Valves outside containment in these lines will automatically isolate for steamline break or Hi-Hi containment pressure.
8.Non-Automatic Containment Isolation Valves open continuously or intermittently for plant operation (under administrative control).	**23.** DDV modified due to press. locking to function as a std.gate valve. 885A upstream disc drilled with 3/16"dia. hole, 885B bonnet connection bypasses downstream disc.
9.Valves may be operated as required to support plant operation.	**24.** Valves may be opened intermittently during Post Accident venting.
10.These series valves have non-redundant phase A automatic signals and therefore are treated as non-automatic containment isolation valves.	**25.** Penetrations identified as H (hot) indicates designed with expansion bellows or expansion coil, C (cold) indicates designed without an expansion bellows or expansion coil.
11.Isolated when Reactor Coolant Pumps are stopped.	**26.** Spring-loaded check valves (pressure relieving).
12.Valves opened intermittently to take samples.	**27.** Flange is double gasketed type, located in refueling canal.
13.Valve opened periodically by the Gas Analyzer.	**28.** Necessary to LC & de-energize if AC-730 & 731 are de-energized open.
14.Opened intermittently for pressure relief.	**29.** Control Rm. Annunciator "Personnel hatches not shut" alarm indication provided.
15. Testable only at Cold Shutdown.	

TABLE 5.2-3
(Sheet 9 of 9)
CONTAINMENT PIPING PENETRATIONS AND VALVING

Numbers shown in brackets () refer to footnotes

NOTES:

30. A Seismic Class I QA Augmented Quality Related temporary fiber optic penetration flange (TFP) may be installed in cold shutdown / refueling conditions to satisfy containment isolation function for refueling operations.	
31. Once opened to facilitate high head or hot leg recirculation, valves would remain open unless closed to isolate a postulated passive failure during the long-term recovery phase. LLR testing is not required.	
32. LLR testing performed to verify adequacy of on-site nitrogen inventory. LLR is not required for Appendix J compliance.	
33. Valves remain open to facilitate high head or hot leg recirculation unless closed to isolate a postulated passive failure during the long-term recovery phase. LLR testing is not required.	
34. LLRT is not required. Valve / penetration is open during Type A ILR test.	

TABLE 5.2-4

CONTAINMENT ISOLATION
VALVE POSITION INDICATION

Valve Number	Control Board Panel Location Red & Green Indicating Lights	Control Board Panel Location (Two-is-True) Monitor Lights
AC-MOV-743	SNF	SB1F
AC-MOV-744	SGF & SB1F	SB1F
CH-AOV-201	SFF & SNF	SNF
CH-AOV-202	SFF & SNF	SNF
CH-MOV-222	SFF & SNF	SNF
RC-AOV-519	SAF	SNF
RC-AOV-548	SNF	SNF
RC-AOV-549	SNF	SNF
RC-AOV-552	SAF	SNF
AC-FCV-625	SGF & SNF	SNF
AC-MOV-769	SGF & SNF	SNF
AC-MOV-784	SGF & SNF	SNF
AC-MOV-786	SGF & SNF	SNF
AC-MOV-789	SGF & SNF	SNF
AC-AOV-791	SGF & SNF	SNF
AC-AOV-793	SGF & SNF	SNF
AC-AOV-796	SGF	SNF
AC-MOV-797	SGF & SNF	SNF
AC-AOV-798	SGF & SNF	SNF
SP-AOV-956A	Sampling System Panel *	SNF
SP-AOV-956B	Sampling System Panel *	SNF
SP-AOV-956C	Sampling System Panel *	SNF
SP-AOV-956D	Sampling System Panel *	SNF
SP-AOV-956E	Sampling System Panel *	SNF
SP-AOV-956F	Sampling System Panel *	SNF
SP-AOV-956G	Sampling System Panel *	SNF
SP-AOV-956H	Sampling System Panel *	SNF
SP-AOV-959	Sampling System Panel *	SNF
VS-FCV-1170	SLF and Fan Room Ctr. Cab.*	SNF
VS-FCV-1171	SLF and Fan Room Ctr. Cab *	SNF
VS-FCV-1172	SLF and Fan Room Ctr. Cab *	SNF
VS-FCV-1173	SLF and Fan Room Ctr. Cab *	SNF
VS-PCV-1190	SLF and Fan Room Ctr. Cab *	SNF
VS-PCV-1191	SLF and Fan Room Ctr. Cab *	SNF
VS-PCV-1192	SLF and Fan Room Ctr. Cab *	SNF
BD-PCV-1214	SCF	SNF
DW-AOV-1	SKF	SNF
DW-AOV-2	SKF	SB1F
RC-AOV-550	SKF	SB1F

* Not located in control room

TABLE 5.2-4
(Cont.)

CONTAINMENT ISOLATION
VALVE POSITION INDICATION

Valve Number	Control Board Panel Location Red & Green Indicating Lights	Control Board Panel Location (Two-is-True) Monitor Lights
SP-AOV-958	CR Isolation Valve Panel JK1	-
WD-AOV-1610	SKF	SNF
SI-MOV-850A	SB2F&WD Extension *	-
SI-MOV-850C	SB2F&WD Extension *	-
BD-PCV-1214A	SCF	SNF
BD-PCV-1215	SCF	SNF
BD-PCV-1215A	SCF	SNF
BD-PCV-1216	SCF	SNF
BD-PCV-1216A	SCF	SNF
BD-PCV-1217	SCF	SNF
BD-PCV-1217A	SCF	SNF
BD-PCV-1223	Sampling System Panel*	SNF
BD-PCV-1223A	Sampling System Panel *	SNF
BD-PCV-1224	Sampling System Panel *	SNF
BD-PCV-1224A	Sampling System Panel *	SNF
BD-PCV-1225	Sampling System Panel *	SNF
BD-PCV-1225A	Sampling System Panel *	SNF
BD-PCV-1226	Sampling System Panel *	SNF
BD-PCV-1226A	Sampling System Panel *	SNF
IA-PCV-1228	SNF	SNF
CA-PCV-1229	SNF	SNF
CA-PCV-1230	SNF	SNF
VS-PCV-1234	SNF	SNF
VS-PCV-1235	SNF	SNF
VS-PCV-1236	SNF	SNF
VS-PCV-1237	SNF	SNF
WD-AOV-1702	Waste Disposal System Panel*	SNF
WD-AOV-1705	Waste Disposal System Panel *	SNF
WD-AOV-1723	Waste Disposal System Panel *	SNF
WD-AOV-1728	Waste Disposal System Panel *	SNF
WD-AOV-1786	Waste Disposal System Panel *	SNF
WD-AOV-1787	Waste Disposal System Panel *	SNF
WD-AOV-1788	SNF	SNF
WD-AOV-1789	SNF	SNF
AC-MOV-822A	SGF & SB1F	SB1F
AC-MOV-822B	SGF & SB1F	SB1F
SI-MOV-851A	SB2F	SB2F
NNE-AOV-863	SMF	SNF

*Not located in control room

TABLE 5.2-4
(Cont.)

CONTAINMENT ISOLATION
VALVE POSITION INDICATION

Valve Number	Control Board Panel Location Red & Green Indicating Lights	Control Board Panel Location (Two-is-True) Monitor Lights
SI-MOV-885A	SB1F	SB1F
SI-MOV-885B	SB1F	SB1F
SI-MOV-888A	SB1F	SB1F
SI-MOV-888B	SB1F	SB1F
SI-MOV-1835A	SB2F	SB2F
SI-MOV-1835B	SB2F	SB2F
AC-MOV-1870	SNF	SB1F
CH-MOV-205	WD Extension *	-
CH-MOV-226	WD Extension *	-
CH-MOV-250A	WD Extension *	-
CH-MOV-250B	WD Extension *	-
CH-MOV-250C	WD Extension *	-
CH-MOV-250D	WD Extension *	-
CH-MOV-441	WD Extension *	-
CH-MOV-442	WD Extension *	-
CH-MOV-443	WD Extension *	-
CH-MOV-444	WD Extension *	-
SP-MOV-990A	WD Extension *	-
SP-MOV-990B	WD Extension *	-
SP-SOV-506	CR Isolation Valve Panel JK1	-
SP-SOV-507	CR Isolation Valve Panel JK1	-
SP-SOV-508	CR Isolation Valve Panel JK1	-
SP-SOV-509	CR Isolation Valve Panel JK1	-
SP-SOV-510	CR Isolation Valve Panel JK1	-
SP-SOV-511	CR Isolation Valve Panel JK1	-
SP-SOV-512	CR Isolation Valve Panel JK1	-
SP-SOV-513	CR Isolation Valve Panel JK1	-
SP-SOV-514	CR Isolation Valve Panel JK1	-
SP-SOV-515	CR Isolation Valve Panel JK1	-
SP-SOV-516	CR Isolation Valve Panel JK1	-

*Not located in control room

5.3 CONTAINMENT VENTILATION SYSTEM

5.3.1 Design Basis

5.3.1.1 Performance Objectives

The Containment Ventilation System was designed to accomplish the following:

a) Remove the normal heat loss from all equipment and piping in the Reactor Containment during plant operation and maintain a normal ambient temperature of 130°F or less

b) Provide sufficient air circulation and filtering 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% of the fuel rods.

c) Provide for positive circulation of air across the refueling water surface to assure personnel access and safety during shutdown

d) Provide a minimum containment ambient temperature of 50°F during reactor shutdown

e) Provide for purging of the containment vessel to the plant vent for dispersion to the environment. The rate of release is controlled by IP3 RECS / ODCM, such that automatic termination of release occurs prior to impacting 10 CFR 20 limits.

f) Provide for depressurization of the containment vessel following an accident. The post-accident design and operating criteria are detailed in Chapter 6

g) Provide ventilation to remove radiogas when steam generator primary man ways are removed

h) Provide means for measurement of flow in main plant ventilation exhaust duct

In order to accomplish these objectives the following systems were provided:

a) Containment Air Recirculation Cooling and Filtration System
b) Control Rod Drive Mechanism Cooling System
c) Reactor Compartment Cooling System
d) Containment Purge System
e) Containment Auxiliary Charcoal Filter System
f) Containment Post-Accident Charcoal Filter System (Described in Section 6.4)
g) Steam Heating System
h) Steam Generator Maintenance Exhaust 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 was determined from the heavier duty specifications associated with the Design Basis Accident (DBA), detailed further in Chapter 6.

The fan motors match the power requirements of the fans, which require a maximum power input of 219 horsepower under accident operation. The fan cooler heat removal rate, as a function of the containment pressure, is presented in Section 14.3.6 covering the Containment Integrity Evaluation. For example, this rate at 271°F and 47 psi containment temperature and pressure is 49.0 x 10^6 Btu/hr per air handling unit. As noted in the Containment Integrity Evaluation, the ability of the Containment Air Recirculation Cooling and Filtration System to function properly in the accident environment was demonstrated by the computer code "HECO." The code determines the plate fin coil heat removal rate when operating in a saturated steam-air mixture.

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-F-40223 [Formerly Figure 6.4-2]. The containment ventilation systems and main plant vent were designed as seismic Class I structures.

5.3.2.2 Containment Recirculation Ventilation

Air recirculation cooling and filtering during normal operation is accomplished using all five air handling units discharged to a common headered ductwork distribution system to assure adequate flow of filtered and cooled air throughout the Containment. The cooling coils in each air handling unit transfer up to 2.3 x 10^6 Btu/hr to the Service Water System during normal plant operation and 49.0 x10^6 Btu/hr/FCU in the event of an accident when supplied with 1400 gpm cooling water at 95°F inlet temperature.

Each air handling unit consists of the following equipment arranged so that during normal operation air flows through the unit in the following sequence: cooling coils, 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. In the event of an accident, the flow path will be diverted automatically by air operated dampers through a compartment containing moisture separators, HEPA filters and charcoal filters. It will then flow through the cooling coils and centrifugal fan and into the distribution header. The normal air flow rate per air handling unit is approximately 70,000 cfm and the post-accident flow rate will be approximately 34,000 cfm, with a 8,000 cfm through the filtration section. Section 6.4.2 provides additional information on the operation of this system.

The recirculating ductwork located in the annulus of the Containment Building was provided with spring loaded relief dampers designed to open inward when the external pressure on the ductwork reaches 2 psig. This is discussed in Section 6.4

The Control Rod Drive Cooling System supplements the main containment recirculation system. The Control Rod Drive Cooling System consists of fans and ductwork to circulate air through the control drive mechanism shroud and discharge it to the main containment volume. Four 1/3 capacity direct driven axial flow fans are used.

5.3.2.3 Containment Purge System

The Containment Purge System includes provisions for both supply and exhaust air. The purge system is maintained isolated whenever the plant is above the cold shutdown condition. 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. Provision was made to measure isokinetic flows at the radiation monitor using pitot tubes. The purge system flow rate is 28,000 cfm; however, the isolation valves will be shut prior to going above cold shutdown and will remain closed during normal operation. The quick closing purge isolation valves are capable of closing within two seconds of receipt of the accident signal. The weld channel and penetration pressurization system pressurizes the space between the purge valves and therefore serves as a continuous on-line monitoring system for valve leakage.

During power operation, containment integrity is maintained with no release from the containment ventilation system to the atmosphere. Prior to purging the Containment, air particulate and gas monitor indications of the closed containment activity levels are used as a guide to making routine releases from the Containment. During power operation, the containment air particulate and gas monitor indications help determine the desirability of using either one or both of two auxiliary particulate and charcoal filter units installed in the Containment primarily for pre-access cleanup.

When the containment purging for access following reactor shutdown is in progress, releases from the plant vent are continuously monitored for radiogas and particulates and sampled for iodine and tritium. A wide range plant vent gas monitor (Section 11.2.3.1) provides continuous indication of noble gas releases passing through the plant vent to the atmosphere.

5.3.2.4 Isolation Valves

The purge supply and exhaust ducts butterfly valves, both inside and outside the containment, are closed during power operation. The spaces between the closed valves are pressurized with air by the Penetration and Weld Channel Pressurization System. The valves were designed for rapid automatic closing by the containment isolation signal (derived from any automatic safety injection signal), or upon a signal of high activity level within the Containment in the event of a radioactivity release when the purge line is open.

5.3.2.5 Containment Pressure Relief Line

The normal pressure changes in the Containment during reactor power operation 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, or by a containment high radioactivity signal. The two intra-valve spaces are pressurized with air by the Penetration and Weld Channel Pressurization System when the valves are closed. The pressure relief line discharges through roughing, HEPA, and charcoal filters to the plant vent. While the valves are fully capable of closing from a 60° open position during accident conditions, mechanical stops prevent the valves from opening more than 40° (90° = full open).

5.3.2.6 Steam Generator Maintenance Exhaust System

Steam generator maintenance ventilation is accomplished by use of two 3000 cfm fans driven by 5 hp motors. These fans connect to 14" diameter exhaust ducts, which allow maintenance on the steam generators when the manways are removed. The fans exhaust into the containment purge exhaust duct.

5.3.2.7 Pressurizer Relief Tank Venting

During shutdown conditions, the potential exists for radioactive gases to be vented from the Pressurizer Relief Tank. These gases are therefore routed to the containment purge exhaust duct where their radioactive content can be monitored (see Section 11.2).

The system uses a jet eductor, using station air to vent the tank. The system is shut down during normal operation.

TABLE 5.3-1

PRINCIPAL COMPONENT DATA SUMMARY

System	Units Installed	Unit Capacity	Units Required for Normal Operation
Containment Recirculation			
Demister	5	8,000 cfm	0
Cooling Coils – Normal	5	2.3 x 10^6 Btu/hr	5
Cooling Coils – DBA	5	49.0 x 10^6 Btu/hr	0
HEPA Filters	5	8,000 cfm	0
Fans	5	70,000* cfm	5
Fans Pressure – Normal	-	6.3 in H20	
Fan Motors (440 V, 3 phase)	5	225 hp	5
DBA Charcoal Filters	5**	8,000 cfm	0
Temperature Switches	30**		
Control Rod Drive Mechanism Cooling			
Fans, Standard Conditions	4	15,000 cfm	3
Fan Pressure	-	5-1/2 in H20	
Fan Motors	4	25 hp	3
Reactor Compartment Cooling			
Part of CB Recirculation System	-	12,000 cfm	
Refueling Canal Air Sweep			
Part of CB Recirculation System	-	17,5000 cfm	

TABLE 5.3-1
(Cont.)

PRINCIPAL COMPONENT DATA SUMMARY

System	Units Installed	Unit Capacity	Units Required for Normal Operation
Containment Ventilation/Purge Supply			
Fans, Standard Conditions	1	40,000 cfm	Optional
Fan Pressure	-	Approximately 2.75 in H_20	
Fan Motors	1	40 hp	
Pre-heat Coils	1 Set		Optional
Air Filters, Roughing	1	40,000 cfm	1
Exhaust			
Fans,* Standard Conditions	2	70,000 cfm**	Optional
Fan Pressure	-	12.5 in H_20	
Fan Motors	2	150 hp	
Plenums	2	40,000 cfm	
HEPA Filters	1 Bank	40,000 cfm	Optional
Roughing Filters	1 Bank	40,000 cfm	Optional
Charcoal Filters	1 Bank	40,000 cfm	Optional
Containment Auxiliary Charcoal Filters			
Fans, Standard Conditions	2	8,000 cfm	Optional
Fan Pressure	-	4.75 in H_20	
Fan Motors	2	10 hp	
Filters; Roughing, HEPA and Charcoal Filters	2	8,000 cfm	Optional

*Note: The two exhaust fans are used interchangeably or as backup for:
1. Ventilation of Primary Auxiliary Building (70,000 cfm)
2. Containment Building Purge System (40,000 cfm)

**Note: Normal System Flow for Containment Building Purge Exhaust is 28,000 cfm.

TABLE 5.3-1
(Cont.)

PRINCIPAL COMPONENT DATA SUMMARY

System	Units Installed	Unit Capacity	Units Required for Normal Operation
Steam Heating			
Heaters, 25 psig steam	2	400,000 Btu/hr each	Optional
Steam Generator Maintenance Exhaust System			
Centrifugal Fan	2	3000 cfm	2
Fan Pressure	-	4.5 in H_20	-
Fan Motors	2	5 hp	2
Containment Building Pressure Relief			
Fan, Standard Conditions	1	1500 cfm	Optional
Fan Pressure	-	3.5 in H_20	
Fan Motor	-	5 hp	
Filters; Roughing, HEPA and Charcoal Filters	1	1500 cfm	Optional

Note: The operating configuration for the Containment Building Pressure Relief system involves limiting the three containment isolation valves to a minimum position of 40° open. This causes a decrease in system flow.

5.4 POST ACCIDENT CONTAINMENT VENTING SYSTEM [Historical Information]

*NOTE: The Post Accident Containment Venting System was retired 04/02/03.

5.4.1 Function

Following a Design Basis Accident, hydrogen gas may be generated inside the Containment by reactions such as zirconium metal with water, corrosion of materials of construction, and radiolysis of aqueous solution in the sump and core. The Post-Accident Containment Venting System permits controlled venting of the containment atmosphere to maintain the hydrogen concentration at a safe level.

5.4.2 Design Basis

The Post-Accident Containment Venting System was designed to limit the hydrogen concentration in the Containment to three percent by volume.

5.4.3 System Description

The Post-Accident Containment Venting System consists of a single line penetrating the Containment, which will be used alternately to supply hydrogen free air to the Containment or exhaust hydrogen bearing gases from the Containment. These exhaust gases are directed through roughing, HEPA, and charcoal filters to the plant vent. The major components of the Post-Accident Containment Venting System are as follows:

5.4.3.1 Containment Air Supply

Hydrogen free air is admitted to the Containment through the single supply/exhaust line. The supply air is provided from the Instrument Air System, which is in use during normal plant operation. The nominal flow rate from either of the two instrument air compressors is 200 scfm. If the Instrument Air System is not available, the Station Air System with a nominal capacity of 600 scfm is available as a backup.

5.4.3.2 Containment Air Exhaust

From inside the Containment, hydrogen bearing gases are exhausted through the single supply/exhaust line. Outside containment is a normally closed, manually operated containment isolation valve followed by a branch connection with an additional manual isolation valve in each branch. Between these adjacent isolation valves, there is a connection through parallel redundant manual valves to the containment penetration pressurization system. Thus, the exhaust line at the containment penetration can be manually sealed with air following a LOCA. Following this valve in each line are:

a) Local pressure indicator for containment pressure
b) Remote manual air operated stop valve
c) Self-contained pressure control valve
d) Manual flow control valve.

The two branches go into a common header and the single line to the plant vent passes through the following:

a) Flow indicator/integrator which provides remote readout of both instantaneous flow rate and integrated flow

b) Local temperature indicator
c) Roughing filter
d) HEPA filter
e) Carbon filter.

The latter three components are shielded by 16 inches of normal concrete.

All the components inside the Containment Building, the penetrations, and the piping and valves to the second isolation valves are seismic Class I design. The remaining components in the exhaust line (except the plant vent, which is seismic Class I) are seismic Class III.

All active components of the system, namely, valves, instruments, controls and associated electrical supplies, are redundant. All passive components, namely, piping and the three filters, are not and need not be redundant.

The system was designed to obtain a flow of 200 scfm with containment pressure at 1.9 psig. For this flow rate, the residence time in the charcoal filters is approximately 0.4 seconds.

5.4.4 Operation

The flow rate and the duration of venting required to maintain the hydrogen concentration at or below 3 percent of the containment volume are determined from the containment hydrogen concentration measurements and the hydrogen generation rate (Section 6.8). The containment pressure necessary to obtain the required vent flow is then determined. Using one of the two instrument air compressors, hydrogen free air is pumped into the Containment until the required containment pressure is reached. The air supply is then stopped and the supply/exhaust line is isolated by valves outside the Containment.

The addition of air to pressurize the Containment will also dilute the hydrogen; therefore, the Containment will remain isolated until analysis of samples indicates that the concentration is again approaching 3 percent by volume. Venting is then started by opening either the primary or bypass exhaust line, and adjusting the hand controlled throttle valve to obtain the required flow.

This process of containment pressurization followed by venting is repeated as may be necessary to maintain the hydrogen concentration at or below 3 percent by volume.

Post Loss-of-Coolant Accident purging provides a backup method to the hydrogen recombiners in the Containment (Section 6.8) for controlling the potential hydrogen accumulation in the Containment. The analysis of offsite doses using purging to control hydrogen was based on the Westinghouse model for hydrogen production and accumulation discussed in Section 14.3.7.

The purging system requires a differential pressure between the Containment and the outside atmosphere in order to permit purging. If required, the containment is pressurized with diluent air when the hydrogen reaches 3 percent by volume after the Loss-of-Coolant Accident. The hydrogen concentration is reduced by this pressurization. Purging is thus delayed until the hydrogen concentration in the Containment has once again built up to 3 percent by volume. The 3 percent hydrogen level was selected as the point of starting the purge because of the following factors:

1) This level allows a sufficient margin of safety below the lower flammability limit of 4.1 percent

2) It provides a sufficient margin so that purging could be delayed a few days if so desired. With neither containment purging nor recombiner operation, the hydrogen generation rate is sufficiently low so that 45 days are required for the hydrogen concentration in the Containment to build up from the 3 percent level to the 4.1 percent level

3) The optimum starting time for the purge, from the standpoint of minimizing the doses, is the latest time.

The hydrogen concentration in the Containment will slowly decrease from 3 percent as purging continues. The required purge rate is based on the hydrogen production rate at the time of purge initiation.

The dose analysis is based on the activity released from the Containment after the time of the postulated Loss-of-Coolant Accident until all the activity in the Containment is either removed or released. The infinite-time thyroid, beta and gamma doses as a function of distance from the plant due to activity release from containment leakage following the postulated Loss-of-Coolant Accident are computed using the core activity release model described in Section 14.3.5. Then the analysis is repeated except that the doses are based on activity released from both containment leakage and purging. The offsite doses due to purging are then determined by subtraction of the doses due to containment leakage and purging. The parameters used to compute the activity releases from containment leakage and from purging are given in Tables 5.4-1 and 5.4-2.

The dose models discussed in Reference (1) and the atmospheric dispersion factor given in Section 14.3.5 are used in determining doses following the Loss-of-Coolant-Accident. In the evaluation of doses from activity released to the atmosphere after 720 hours (30 days), the annual average dispersion factor at the site boundary of 2.6 x 10^{-5} sec/m^3 was used. The thyroid beta and gamma doses at the site boundary due to containment purging to control hydrogen are given in Table 5.4-4.

The thyroid, beta and gamma doses due to containment purging to control hydrogen were also determined using the assumptions outlined in Reference (2) and the dos models given in Reference (1). The parameters used to determine the activity release resulting from leakage and purging using the model in Reference (2) are given in Tables 5.4-1 and 5.4-3, and the boundary doses are listed in Table 5.4-4.

References

1) “Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant-Accident for Pressurized Water Reactors,” Safety Guides for Water Cooled Nuclear Power Plants, Safety Guide No. 4, Division of Reactor Standards, U.S. Atomic Energy Commission, November 1971.

2) “The Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident,” Safety Guides for Water Cooled Nuclear Power Plants, Safety Guide No. 7, Division of Reactor Standards, U.S. Atomic Energy Commission.

TABLE 5.4-1

PARAMETERS USED TO DETERMINE CONTAINMENT LEAKAGE ACTIVITY RELEASE

Plant Power	3216 MWt	
Containment free volume	2.61×10^6 ft^3	
Unsprayed containment volume	5.22×10^5 ft^3	
Mixing rate between sprayed and unsprayed containment volume	2.4×10^4 cfm (filtered) 1.26×10^5 cfm (unfiltered)	
Spray removal coefficient for elemental iodine	32 hr^{-1} until DF=100	
Containment design leak rate	0.1% per day (0-24 hours)	
	0.045% per (>24 hours)	
Containment Air Recirculation and Cooling System filter efficiencies	Westinghouse Model	AEC Model
Elemental Iodine	90%	90%
Methyl Iodine	70%	5%
Particulate Iodine	90%	90%

TABLE 5.4-2

PARAMETERS USED TO DETERMINE
HYDROGEN PURGING ACTIVITY RELEASE – W MODEL

1. Westinghouse Basis H_2 Generation*

2. Pressurize Containment to 2.14 psig, if required, when H_2 reaches 3.0 percent by volume (day 36)

3. Purge at 15.3 scfm continuously once H_2 reaches 3.0 percent by volume again (day 50)

4. Containment Air Recirculation and Cooling System filter efficiencies

Elemental Iodine	90%
Organic Iodine	70%
Particulate Iodine	90%

*NOTE: Discussed in Section 14.3.7.

TABLE 5.4-3

PARAMETERS USED TO DETERMINE
HYDROGEN PURGING ACTIVITY RELEASE – AEC MODEL

1. AEC Basis H_2 Generation*

2. Pressurize Containment to 2.14 psig, if required, when H_2 reaches 4.0 percent by volume (day 23)

3. Purge at 21.8 scfm continuously once H_2 reaches 4.0 percent by volume again (day 33)

4. Containment Air Recirculation and Cooling System filter efficiencies

Elemental Iodine	90%
Organic Iodine	5%
Particulate Iodine	90%

*Safety Guide No.7

TABLE 5.4-4

DOSES FROM CONTAINMENT PURGING TO CONTROL HYDROGEN

Dose at Site Boundary (Rem)

	Westinghouse Model	AEC Model
Thyroid	4.5×10^{-1}	5.1
Beta	1.4	1.6
Gamma	1.7×10^{-2}	5.0×10^{-2}

5.5 CONTAINMENT PARAMETERS

The description of the instrumentation system included in the Indian Point 3 design for remote monitoring of post-accident conditions within the primary containment is presented in Appendix 6F. Non-nuclear process instrumentation of the containment is described in Section 7.5.

Containment Building Pressure

The containment pressure is transmitted to the main control board for post accident monitoring. Six transmitters, two in each of three safety channels, are installed outside the containment to prevent potential missile damage. The pressure is indicated on the main control board; the range is –5 psig to 75 psig.

In addition, monitoring of the containment building pressure during and following an accident is effected by two Safety Related redundant systems. Pressure signals are obtained at the pipe penetration area and brought to transmitters outside containment. These same signals are transmitted to the two-recorders at the control room recorder cabinet. Continuous monitoring of containment pressure is possible in the –5 to 200 psig range. Power requirements for the two systems are met from vital instrument buses. The installation of cable and conduit is consistent with separation criteria, as outlined in Section 8.4.

Containment Building Water Level Monitoring

There are three sumps in the Containment Building: Reactor Pit Sump, Recirculation Sump and Containment Sump. Associated with the Recirculation Sump and Containment Sump, there are two redundant, separately channeled and powered level measurement loops. Associated with the Reactor Pit is a level sensor, alarmed on the Control Room Supervisory Panel. These provide continuous level and alarm indication in the Control Room. Additionally, a water level transmitter installed at the top of Containment Sump will provide a Containment Sump overflow alarm indication in the Control Room.

Containment Building Hydrogen Concentration

Hydrogen concentration indication is provided by a measuring system which consists of the following: redundant analyzers and continuously recording two (2) single pen recorders. The recorders are located in the control room and the analyzers are located in the pipe penetration area of the fan house. Samples are drawn from containment recirculation fans via the retired post-accident sample system and returned to the general area of the containment building.

APPENDIX 5-A

[Historical Information]

WESTINGHOUSE NUCLEAR ENERGY SYSTEMS
UNITED ENGINEERS AND CONSTRUCTORS

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

CONTAINMENT DESIGN REPORT

September 1970

B. Scott
J. Slotterback
J. D. Stevenson

5A-i

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1.0 INTRODUCTON

1.1.0 PURPOSE & SCOPE OF REPORT

The object of this report is to illustrate the design adequacy of the containment structure for the Indian Point Nuclear Generating Unit No. 3. To this end, this documentary report describes the design of the structure, as well as the construction procedures, to demonstrate fulfillment of the design criteria.

The following sections of this report enumerate the basic criteria that were used, the analyses that were developed to satisfy these criteria, the various loading combinations under normal and postulated accident conditions (including seismic effects), and the construction and testing procedures that were employed to ultimately construct the containment structure at the site.

1.2.0 FUNCTION OF CONTAINMENT STRUCTURE

The containment structure 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 structure will provide biological shielding for normal and accident situations.

The containment structure is designed to safely withstand several conditions of loading and their credible combinations. The limiting extreme conditions are:

a) Occurrence of a gross failure of the reactor coolant system which creates a high pressure and temperature condition within the containment.

b) Coincident failure of the reactor coolant system with an earthquake or wind.

The design pressure and temperature of the containment will be, as a minimum, equal to 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 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 will be 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 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; the latter utilizing only the emergency electric power supply.

The design pressure and temperature on the containment structure will be those created by the hypothetical loss-of-coolant accident. The reactor coolant system will contain approximately 512,000 lbs. of coolant at a weighted average enthalpy of 595 Btu/lb. for a total energy of 304,000,000 Btu. In a hypothetical accident, this water is released through a double-ended break in the largest reactor coolant pipe, causing a rapid pressure rise in the containment. The reactor coolant pipe used in the accident will be the 29-in. ID section because rupture of the 31-in. ID section requires that the blowdown go through both the 29-in. and the 27-1/2-in. ID pipes and would, therefore, result in a less severe transient.

Additional energy release was considered from the following sources:

a) Stored heat in the reactor core.

b) Stored heat in the reactor vessel piping and other reactor coolant system components.

c) Residual heat production.

d) Limited metal-water reaction energy and resulting hydrogen-oxygen reaction energy.

The following loadings will be considered in the design of the containment in addition to the pressure and temperature conditions described above:

a) Structure dead load.

b) Live loads.

c) Equipment loads.

d) Internal test pressure.

e) Earthquake.

f) Wind. (Tornado)

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 containment mat.

A 3-ft. thick concrete ring wall serving as a 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. The floor is 5 ft. thick. The concrete walls and floor are lined with ¼ inch thick stainless steel plate. The linings provide a leak- proof membrane that is resistant to abrasion and damage during fuel handling operation.

1.3.0 CONTAINMENT DESCRIPTION

The reactor containment structure is a reinforced concrete vertical right cylinder with a flat base and a hemispherical dome. A welded steel liner with a minimum thickness of ¼ inch is attached to the inside face of the concrete shell to insure a high degree of leak-tightness. 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 consists of side walls measuring 148-feet from the liner on the base to the springline of the dome, and has an inside diameter of 135-feet. The side walls of the cylinder and the dome is 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 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. structural slab of concrete which serves to carry internal equipment loads and forms the floor of the containment. The internal pressure within the containment is self-contained in that the vector sum of the pressure forces is zero; therefore, there is no need for mechanical anchorage between the bottom mat and underlying rock. The base is supported directly on rock.

The basic structural elements considered in the design of the containment structure is 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 so that it forms an integral part of the entire composite structure under all membrane loadings. The reinforcing in the structure has an elastic response to all primary loads with limited maximum strains to insure the integrity of the steel liner. The lower 20 feet of the cylindrical liner is insulated to avoid excess deformation of the liner due to restricted radial growth when subjected to a rise in temperature.

2.0 CONTAINMENT STRUCTURAL DESIGN BASIS

2.1.0 DESIGN LOAD CRITERIA

The following loads were considered to act upon the containment structure creating stresses within the component parts.

2.1.1 DEAD LOADS

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 were as follows:

a) Reinforced Concrete : 150 lb/ft^3

b) Steel Lining : 490 lb/ft^3 using nominal cross-sectional area

d) Insulation : 6 lb/ft^3 including stainless steel jacket.

2.1.2 OPERATING LIVE LOADS

Operating live loads consist of the weight of major components of equipment in the containment. Equipment loads were those specified on the drawings supplied by the manufacturers of the various pieces of equipment.

All major pieces of equipment are supported on the 3'-0" base slab or on the interior concrete, which in turn bears directly on the 9'-0" mat.

Item		Flooded Operating Weight, lb.
Pressurizer	-1	346,000
Steam Generators	-4	3,746,000
Reactor	-1	
a) Vessel		868,000
b) Internals		420,000
RCS Piping		1,000,000
Reactor Pumps	-4	824,000
Accumulator Tanks	-4	529,000
175 Ton Polar Crane	-1	650,000
Ventilation Fans	-4	656,000
Reactor Coolant Drain Tank	-1	20,000
Pressure Relief Tank	-1	100,000
Other Misc. Equipment		100,000
		9,259,000

Other Uniform Live Loads

@ El. 68' – 10 ft. strip adjacent to crane wall = 600 psf

Remaining strip = 100 psf

@ El. 95' - 0" – Concrete Slab = 500 psf

Grating areas = 100 psf

2.1.3 SNOW LOADS

Snow and ice loads have been applied uniformly to the top surface of the dome at an estimated value of 20 pounds per square foot of horizontal projection of the dome. This loading represents approximately 2-ft. of snow, which was considered to be a conservative amount since the slope of the dome tends to cause much of the snow to slide off.

2.1.4 CONSTRUCTION LOADS

A construction live load of 50 pounds per square foot has been used on the dome, but was not considered to act concurrently with the snow load.

A load equivalent to the weight of wet concrete, placed in sections during construction of the concrete dome, was used for the design of the stiffened dome liner plate. During the pressure test of containment, the concrete will crack and thereby relieve the effects of shrinkage and creep.

2.1.5 WIND LOADS

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. In this code, for height zones between 100 and 499 feet, 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 26 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 uniform 30 psf basic wind load has been used from ground level up.

The tornado loads considered in design are as follows:

a. cyclonic wind velocity = 300 mph
b. translational wind velocity = 60 mph
c. differential pressure drop = 3 psi in 3 seconds
d. missile - 4" x 12" x 12' plank at 300 mph horizontal, or at 90 mph, vertical.
e. Missile – 4000 lb passenger car, not exceeding 25 feet above the ground, at 50 mph horizontal or at 17 mph vertical (25 ft^2 contact area).

2.1.6 OPERATING TEMPERATURE LOADS

The operating temperature assumed in the design of the containment structure in 120°F, with a -5°F outside winter temperature. Thermal loads induced in the containment as a result of operating temperature effects are composed of a) the steady state temperature gradient through the wall as shown in Figure 2.1 for Winter conditions for both the insulated and uninsulated portions of the liner and b) the effective load induced in the concrete shell as the concrete acts to restrain the steel liner when the mean temperature of the concrete differs from that of the liner.

2.1.7 CREEP AND SHRINKAGE LOADS

The containment structure has been investigated for end of life creep and shrinkage factor as follows:

(a) $k_{(creep)} = 0.22\,x10^{-6}\ in/in/psi$

(b) $k_{(shrinkage)} = 70\,x10^{-6} in/in$

The maximum stress induced in the steel reinforcement by this maximum condition is less than 4000 psi. Since the limiting case for design is accident pressure load which effectively cracks the concrete and places the reinforcement into membrane tension creep and shrinkage induced stress are not a limiting factor in design.

2.1.8 SEISMIC LOADS

The ground acceleration for the Operational Basis Earthquake, “OBE” was determined to be 0.1g applied horizontally and 0.05 applied vertically. These values were resolved as conservative numbers based upon recommendation from Dr. Lynch, Director of Seismic Observatory, Fordham University. A dynamic analysis has been used to arrive at equivalent design loads. Additionally, a Design Basis Earthquake, “DBE” acceleration of 0.15 horizontally and 0.10 vertically has been used to analyze for the no-loss of function.

A damping factor of 2 percent was assumed for the reinforced concrete containment structure for the OBE and 5 percent for the DBE. The response spectra used were based on the Spectrum curves presented in Figures A.1-1 and A.1-2 of Appendix A1 normalized to 0.15g zero period ground acceleration as required.

2.1.9 ACCIDENT PRESSURE LOADS

The design basis accident pressure load is shown in Figure 5.1-8 of the FSAR as a function of time. This design value is at least 5 percent in excess of maximum calculated containment pressure.

2.1.10 ACCIDENT TEMPERATURE LOADS

The design basis accident containment temperature assumed in the design of the containment is also shown in Figure 5.1-8 of the FSAR as a function of time. This containment temperature induces loads in the concrete shell as the concrete acts to restrain liner thermal expansion. This thermal load effect on the liner is combined with pressure load effects to develop design basis accident design load requirements as a function of time. Accident temperature induced thermal gradients through the wall are not a factor in concrete shell design since the accident temperature effect penetrates approximately 10 percent of the containment wall thickness during the significant overpressure phase of the accident and the cracking of the concrete shell due to containment pressurization acts to relieve secondary stresses induced by thermal gradient effect.

2.1.11 LOADS AT PENETRATIONS

The effect of growth of the liner due to accident conditions has been considered in the design of penetrations and sleeves together with the effects of lateral loads due to thermal expansion of pipes, seismic motion, pipe break loads and pressure loads. In addition, stress concentration effects on large penetrations have been considered.

2.1.12 MISSILE LOADS

Potential external missiles (tornado; turbine failure) have been considered in design as described in Appendix 14A to the Report.

2.1.13 TEST PRESSURE LOADS

Internal pressure will be applied to test the structural integrity of the vessel up to 115 per cent of the design pressure of 47 psi. For this structure the test pressure will be 54 psig.

2.1.14 COMBINED FACTORED LOAD EQUATIONS

The design was based upon limiting load factors which were used as the ratio by which loads were multiplied for design purposes to assure that the loading formation behavior of the structure was one of elastic, tolerable strain behavior. The load factor approach was used in this design as a means of making a rational evaluation of the isolated factors which must be considered in assuring 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:

a) $C = 1.OD \pm 0.05D + 1.5P + 1.0\ (T + TL)$ (2.1.1)

b) $C = 1.OD \pm 0.05D + 1.25P + 1.0\ (T' + TL') + 1.25E$ (2.1.2)

c) $C = 1.OD \pm 0.05D + 1.OP + 1.0\ (T'' + TL'') + 1.OE'$ (2.1.3)

d) $C = 1.OD \pm 0.05D + 1.OW'$ (2.1.4)

Symbols used in these formulas are defined as follows:

C: = Required load capacity 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 temperatures 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 operational basis earthquake.

T": = Load due to maximum temperature gradient through the concrete shell, and mat based upon temperature associated with the accident pressure.

TL": = Load exerted by the liner based upon temperatures associated with the accident pressure.

E': = Load resulting from design basis earthquake.

W': = Tornado wind load and the pressure drop effect.

Load condition a) indicates that the containment has the capacity to withstand loadings at least 50 per cent greater than those calculated for the postulated loss-of-coolant accident alone.

Load condition b) indicates that the containment has the capacity to withstand loadings at least 25 per cent greater than those calculated for the design basis accident with a coincident operational basis earthquake.

Load condition c) indicates the containment will withstand loads at least equal to those calculated for the design basis accident coincident with a design basis earthquake. The Indian Point Unit No. 3 containment has the capacity to withstand loadings associated with the design basis accident and a coincident earthquake within ACI specified Ultimate Strength Design stress level allowable limits.

Load condition d) indicates the containment will withstand loads at least equal to those calculated for the design basis tornado within ACI specified Ultimate Strength Design stress level allowable limits.

All structural components have been designed to have a capacity required by the most severe loading combination. The loads resulting from the use of these equations will hereafter be termed “factored loads.” Specific resultant loading diagrams are presented in Figures 2.2 through 2.9.

The load factors utilized 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 procedures, the load factors in the design primarily provide for a safety margin on the load assumptions.

2.2.0 STRESS, STRAIN OR DEFORMATION CRITERIA

The containment is designed such that under all factored load conditions the behavior of the structure will be in the small deformation elastic range. This behavior range is defined by the stress limits contained in the ACI-318-63 code to include additional margin as provided by the capacity reduction factor, $\varnothing$.

2.2.1 CAPACITY REDUCTION FACTOR $\varnothing$

The theoretical member capacity is lowered by the reduction factor $\varnothing$ to recognize variation in quality of materials and permissible tolerances in bar and plate areas and section dimensions, as well as approximations inherent in theoretical analysis. In theory the capacity reduction factor should be divided into the calculated load effect to determine actual design load requirements. Since $\varnothing$ is less than one this always results in a design load requirement in excess of calculated requirements.

As a practical matter in the design of this containment the capacity reduction factor has been applied as a multiplier to the theoretical stress criteria. This has the result of reducing the allowable stress as a function of the type of load being carried.

The following $\varnothing$ factors for both concrete and steel are used in design:

$\varnothing$ = .95 (tension)
$\varnothing$ = .90 (flexure)
$\varnothing$ = .85 (diagonal tension, bond and anchorage)

2.2.2 CONCRETE STRESS CRITERIA

The stress criteria governing behavior are as specified in Part IV-B of the ACI-318-63 Code. Specifically the code limitations on concrete compression, tension, shear strength with and without web reinforcement, bond and anchorage are followed. These values are further reduced by applicable capacity reduction factors.

2.2.3 CONCRETE REINFORCING STEEL

The calculated structural capacity of reinforced concrete sections is based on the specified minimum yield strength of the reinforcement using the design methods specified in Part IV-B of the ACI-318-63 Code. This limiting stress value is further reduced by the applicable capacity reduction factor.

2.2.4 STEEL LINER PLATE

The maximum steel stress is limited to 0.95 yield under all primary loading conditions. In regions of local stress concentrations or stresses due to localized secondary load effects the maximum liner strain is limited to 0.5 per cent. (Detailed finite element computer analysis has identified regions of high localized liner stresses which would not have been detected using conventional analytical techniques.)

2.2.5 PENETRATIONS

The steel penetration elements not backed up by concrete are designed to carry design basis accident loads plus operational basis earthquake loads (unfactored) within the stress limitations of the ASME Section VIII Unfired Pressure Vessel Code stress limitations. It should be noted the ASME Code is a "working stress" design code and as such has safety margin contained in the reduced stress levels rather than in the factored load concept.

2.2.6 SUMMARY OF MATERIAL STRESS STRAIN PROPERTIES

The materials used in containment conform to stress-strain limitations as follows:

	Item	Specification	Min. Yield Strength (PSI)	Min Ultimate Strength (PSI)	Elongation
1.	Concrete	ACI-318	-	3,000	-
2.	Reinforcing Steel	ASTM A432 ACI-318	60,000	90,000	7% in 8"
3.	Liner Plate	ASTM A442, Gr.60	32,000	60,000	22% in 8"
4.	Mech. Penetration Sleeve-12" Dia. & under	ASTM A333 Gr. I	30,000	55,000	35% in 2"
5.	Mech. – Over 12" Dia.	ASTM A201 GR. B to A300	32,000	60,000	22% in 8"
6.	Rolled Shapes	ASTM A36	36,000	58,000	20% in 8"
		ASTM A131 GR. C	32,000	58,000	21% in 8"
7.	End Plates a)	ASTM A300 C1.1 Firebox A201, Gr. B	32,000	60,000	22% in 8"
	b)	ASTM A240 Tp. 304L	25,000	70,000	40% in 2"
8.	Fuel Transfer Tube	ASTM A240 Tp. 304L	25,000	70,000	40% in 2"
9.	Bellows a)	ASTM A312 Tp. 304L	25,000	70,000	35% in 2"
	b)	ASME SB168 Inconel 600	35,000	80,000	30% in 2"
10.	Elec. Penetrations	ASTM A333 Gr.1	30,000	55,000	35% in 2"
11.	Equip. Hatch Insert	ASTM A300 C1.1 Firebox A201, Gr. B	32,000	60,000	22% in 8"
12.	Equip, Hatch Flanges	ASTM A300 C1.1 Firebox A201, Gr. B	32,000	60,000	22% in 8"
13.	Equip. Hatch Head	ASTM A300 C1.1 Firebox A201, Gr. B	32,000	60,000	22% in 8"
14.	Personnel Hatch	ASTM A300 C1.1-Firebox	32,000 A201, Gr. B	60,000	22% in 8" 5A-16

3.0 CONTAINMENT ANALYSIS METHODS AND COMPARISON WITH CRITERIA

3.1.0 GENERAL CONTAINMENT LOADS

3.1.1 DEAD LOAD

The weight of the concrete structure above the point under consideration based on a density of 150#/ft^3 which includes only the weight of the reinforced concrete structure. Since the maximum rebar stress occurs in tension it is conservative not to consider snow loads or any other non-permanent load which will add to the dead load.

The formula for dead load in k/ft at any point is

$$^{T}DL_i = .150 V_i / 2\pi R \qquad (3.1.1)$$

where:

V_i = the volume of concrete in feet cubed above point i

R = mean radius in feet

$^{T}DL_i$ = the dead load at any point in the structure (k/ft) of wall

i = the point under consideration

The horizontal thrust from the dead weight of the dome is computed by considering

$$H = -T + wr\cos\varnothing_O \qquad (3.1.2)$$

and

$$T = W / 2\pi r \sin^2\varnothing_O \qquad (3.1.3)$$

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where:

W = $2\pi r^2 w(1-cos\varnothing_o)$; the total weight of the dome above the point defined by $\varnothing_O$ in kips (3.1.4)

H = the horizontal or hoop thrust in the dome in k/ft of shell

r = mean radius of dome in feet

$\varnothing_O$ = the central angle measured from the top of the dome to the point under consideration

w = the dead load per unit surface area of shell in k/ft.^{2}

T = the vertical or meridinal thrust in the dome in k/ft of shell

3.1.2 DESIGN BASIS ACCIDENT PRESSURE LOAD

Membrane pressure loads in the vertical direction in the cylinder and either direction in the dome are determined by

$$P = \frac{pR}{2} \quad (3.1.5)$$

For the horizontal or hoop direction in the cylinder

$$P = pR \quad (3.1.6)$$

where:

P = pressure load in #/in of wall

p = internal design pressure in #/in^2

R = mean radius in inches

3.1.3 DISCONTINUITY MOMENT AND SHEAR LOAD

The bending moments, shears and deflections induced in the cylindrical shell by the restraint provided by the base are found by considering a cylindrical shell with a uniform internal pressure. (1) Using the general equations for deflection and slope for a cylinder with end moment and shear, and substituting boundary conditions of $w = \delta$ and $\theta = 0$ at $x = 0$ (the built in end) where δ = the unrestrained growth of a cylinder under uniform internal pressure, one obtains formula for the moment and shear at the built-in end to cause zero deflection and rotation

$$Mo = P/2\beta^2 \quad \text{and} \quad Q_O = -P/\beta \qquad (3.1.7)$$

where:

P	=	the internal pressure in #/in^2	
β^4	=	$E_s h_s / 4a_c^2 D$	(3.1.8)
D	=	$E_c h_c^3 / 12(1-\mu)$ (flexural rigidity of the shell)	(3.1.9)
h_s	=	area of horizontal steel and liner in the cylinder which acts as a spring constant (in^2 / in)	
a_c	=	mean radius of the containment cylinder in inches	
h_c	=	effective depth or thickness of the wall	
μ	=	Poisson's ratio = 0 for cracked concrete	
E_s	=	modulus of elasticity of steel = 29 x 10^6 psi	
E_c	=	modulus of elasticity of concrete = 3.2 x 10^6 psi	
M_o	=	moment at built in Section to cause 0 rotation	
Q_o	=	shear at built in Section to cause 0 deflection	

Substituting these values in the following expressions, values for bending moment, shear and deflection at any distance from the end can be found:

$$\Delta_x = w = \frac{-1}{2\beta^3 D} \left[\beta M_0 \gamma(\beta x) + Q_o \theta(\beta x)\right] \tag{3.1.10}$$

$$\theta_x = \frac{dw}{dx} = \frac{1}{2\beta^2 D} \left[2\beta M_0 \theta \quad (\beta x) + Q_o \phi(\beta x)\right] \tag{3.1.11}$$

$$M_x = D\frac{d^2 w}{dx^2} = \frac{-1}{2\beta D} \left[2\beta M_0 \phi(\beta x) + 2Q_o \delta(\beta x)\right] D \tag{3.1.12}$$

$$V_x = D\frac{d^3 w}{dx^3} = \frac{1}{D} \left[2\beta M_0 \delta(\beta x) - Q_o \gamma(\beta x)\right] D \tag{3.1.13}$$

where:

$$\phi(\beta x) = e^{-\beta x}(cos\,\beta x + Sin\,\beta x)$$

$$\gamma(\beta x) = e^{-\beta x}(cos\,\beta x + Sin\,\beta x)$$

$$\theta(\beta x) = e^{-\beta x} cos\,\beta x$$

$$\delta(\beta x) = e^{-\beta x} S\,in\,\beta x$$

Δ_x = the deflection of the shell at x

θ_x = the slope of the shell at x

M_x = the moment of the shell at x

V_x = the shear in the shell at x

From these values Figures 3.1 and 3.2 are plotted showing moment and shear v. height of wall in inches. Since no backfill is present shifts in moment and shear, due to backfill restraint, will not occur.

The problem of determining the discontinuity moment and shear at the springline is similar to that at the base. Discontinuity forces at the dome-cylinder junction are only a function of the relative deformation at this point, since the rotations of the cylinder and the dome due to the internal pressure are zero and therefore present no discontinuity. The extension of the radius of the cylindrical shell due to the internal pressure is given by

$$\delta_c = \left(Pa_c^2/E_s h_s^c\right)\left(1-\mu/2\right) \tag{3.1.14}$$

and the unrestrained extension of the dome $\left(\delta_D\right)$ is given by

$$\delta_d = \left(Pa_d^2/2E_s h_s^d\right)\left(1-\mu\right) \tag{3.1.15}$$

where

a_d = mean radius of the containment dome in inches

h_s^d = area of horizontal steel and liner of the dome which acts as a spring constant (in^2 / in)

Since the area of the hoop steel per foot in the dome is approximately one half that of the cylinder, the values of δ_c and δ_d are nearly equal and therefore the relative deformation is insignificant.

In calculating the discontinuity effects, the bending is of a local character so that an approximate solution can be obtained by assuming that the bending is of importance only in the zone of the dome close to the springline and that this zone can be treated as a portion of a long cylindrical shell. Equations of continuity for deflection and rotation are written such that the values of M_o and Q_o at the springline may be found. The distribution of the moment and shear into the dome and the cylinder are then found by substituting M_o and Q_o into equations 3.1.12 and 3.1.13. The resulting moments and shears are insignificant.

3.1.4 BASE MAT LOADS

The beam shears and moments in the base mat can be calculated by considering the loads shown in Figure 3.3 acting on a 1' –0 wide beam. The 1' –0 strip of mat to be considered is located at the point where the uplift from the overturning moment in the containment due to earth quake is maximum.

This gives the maximum moments and shears in the strip.

The loads considered as shown in Figure 3.3. are:

$$U_T = P + T_{EQ} + T_v - T_{DL} \text{ in k/ft} \quad (3.1.16)$$

where:

P = design basis loss-of-coolant accident pressure effect load in the wall in k/ft of wall

T_{EQ} = the tensile load k/ft of wall developed by the earthquake overturning moment

T_v = the effective tensile load or reduction of dead load in k/ft of wall caused by response of the containment structure to vertical earthquake motion

T_{DL} = the dead load in the wall in k/ft

M = base discontinuity moment defined in Section 3.1.3

V'_u = base discontinuity shear defined in Section 3.1.3

D = the dead weight of the base mat on the outside of containment cylindrical wall centerline in k/ft

C = the reaction of the internal structural support columns which are based on the 3'–0 reinforced concrete fill mat; in all cases equal to 50K spaced every 23'–0

w = $12p + \frac{\rho z}{12}$; the effective uniform load acting on a 1' wide segment of the base slab per inch of segment length where

p = the containment internal pressure in kips/in^2

ρ = the density of reinforced concrete in #/ft^3 = 150#/ft^3

z = the total depth of section including the 3' –0 fill slab.

The crane wall reaction in k/ft is determined by

$$R = D_c + D_o + P_c \quad (3.1.17)$$

where:

D_c = $\rho t_1 \; H$; or the dead weight of the crane wall in k/ft

t_1 = the thickness of the crane wall = 3.0 ft.

H = the height of the crane wall = 50.0 ft.

D_o = $\pi R_1^2 t_2 \rho / 2\pi R_2$ or the approximate dead weight of the operating floor in k/ft

R_1 = the outside radius of the operating floor = 53' –0

R_2 = the mean radius of the crane wall

= 51' –6

P_c = 12 pt_1 or the pressure load acting on the top of the crane wall with t_k given in inches

t_2 = the thickness of the operating floor = 2' –0

Moments and shears are calculated by writing equations for moment and shear in terms of x as the origin, with x increasing toward the center of the containment building and x measured in inches.

The formulas are as follows:

For $0 \le x \le 201$

$V_x = U_T - D - 2C^* - wx$ (3.1.18)

* $-2C$ when $x \ge 201$

$-c$ when $x < 201$

with V_x assumed constant and equal to the value of V_x at 201 inches for the region under the crane wall $201 \le x \le 237$.

For $x \ge 237$

$$V_x = U_T - D - C - w(201) - R - C\,w(x-237) \quad (3.1.19)$$

or

$$V_x = U_T - 2C - D - R - wx + 36w \quad (3.1.20)$$

where:

V_x = uplift shear at any point x (inches) in k/ft

Equation 3.1.20 is applicable until $V_x \le 0$.

The design moment in the base slab is determined for $0 \le x \le 201$

$$M_x = M + V'_u e + D(x+19.5) + \frac{wx^2}{2} + C(x-27) - U_T x \quad (3.1.21)$$

With M_x assumed constant and equal to the value of M_x at 201 inches for the region under the crane wall $201 \le x \le 237$. 5A-24

For $x \geq 237$

$$M_x = M + V'_u e + D(x + 19.5) + w(201)(x - 105.5) + C(x-27)$$

$$+ C(x-201) + R(x-219) + \frac{w(x-237)^2}{2} - U_T x \qquad (3.1.22)$$

or

$$M_x = M_u + V'_u e + D(x + 19.5) + x(1/2\, x^2 - 36x + 6850) + 2Cx - 228C$$

$$(3.1.23)$$

$$+ R(x-219) - U_T x$$

where:

e = the effective depth of the 9' –0 base mat divided by 2 and

M_x = the base moment at any point x (inches) in in-k/ft

At the point where $V_x \leq 0$ flexural beam action is no longer considered since uplift is 0 and the mat acts as a flat circular plate supported on a rigid non-yielding foundation.

Again it should be noted that these maximum values for shear and moment occur at only one point on the base slab circumference where the uplift from the horizontal earthquake is maximum and decreases to zero 90° from this point; therefore, it is considered that the calculations shown are conservative.

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 the stresses determined are negligible. Accident temperatures have no appreciable effect on the base slab.

3.1.5 SEISMIC LOAD

Horizontal Earthquake

The loads on the containment structure caused by the earthquake are determined by Dynamic Analysis of the structure. The Dynamic Analysis is made on an idealized structure of lumped masses and weightless elastic columns acting as spring restraints. The model representation is essentially that of a cantilever beam. Since the containment is founded on rock, no translation or rotation of the structure as a rigid body is considered.

The analysis is performed in two stages: The determination of the natural frequencies of the structure and its mode shapes, and the modal response of these modes to the earthquake by the spectrum response-method.

The natural frequencies and mode shapes are computed from the equations of motion of the lumped masses. These equations are solved by iteration techniques by a fully tested digital computer program. The form of the equation is:

$$(k)\Delta = \omega^2 (M)\Delta \qquad (3.1.24)$$

(k)	=	Matrix of stiffness coefficients including the combined effects of shear and flexure.
(M)	=	Matrix of concentrated masses. Each mass may have up to six degrees of freedom.
Δ	=	Matrix of mode shapes
ω	=	angular frequency of vibration

The results of this computation are the several values of ω_n and mode shapes $(\Delta)_n$ for n = 1, 2, 3…N, where N is the number of degrees of freedom assumed in the idealized structure.

The response of each mode of vibration to the earthquake ground motion is computed by the response spectrum technique as follows:

The participation of each mode, P_n is computed from:

$$P_n = \sum_{m=1}^{N} \Delta_{mn}^{x} M_m \Big/ \sum_{m=1}^{N} \Delta_{mn}^{2} M_m \qquad (3.1.25)$$

Where Δ_{mn} is the deflection of mass point m in mode n.

Δ_{mn}^{x} is the component of Δ_{mn} in the direction of the earthquake.

The relative deflection of each mass is determined from

$$Y_{mn} = \Delta_{mn} x P_n S_{an} \Big/ \omega_n^2 \qquad (3.1.26)$$

Where S_{an} = the spectral acceleration of a single degree of freedom system with a frequency ω_n and damping coefficient, a.

The shear at any section of the cylinder is determined by

$$V_s = RMS\left(V_{im} = \Sigma F_{rim}\right)$$

where:

V_{im} = the seismic shear at point i for mode m

F_{rim} = the horizontal inertial force at r nodes above elevation i for mode m

V_s = the root mean square of V_1 for each mode

Figures 3.4 and 3.5 show the base shear and moment distribution up the wall.

The shear flow is determined by consideration of a hollow ring with a total thickness of 2t.

$$S_f = V_s Q/I \qquad (3.1.28)$$

where:

S_f = shear flow in the wall

V_s = shear at the elevation under investigation as determined by (Eqt. 3.1.27)

$$Q = 2\int_o^{\pi/2} ydA$$

where:

y = distance from element under consideration to the neutral axis of the circular tube cross section of R sin θ

R = radius of containment in inches

θ = angle from neutral axis to the element under consideration in radians

dA = the area of the element under consideration of Rtdθ

t = the thickness of the containment shell in inches

I = moment of inertia of section about neutral axis in $in.^4$

Vertical Earthquake:

The frequency of the containment structure in the vertical direction is determined as described in the dynamic analysis for horizontal earthquake loads.

Using this frequency of , 12.0 cps and the given value response spectral acceleration curve with 5% critical damping a coefficient of spectral acceleration of 0.11g (1.OE') is obtained for the design basis earthquake For the operational basis earthquake with 2% critical damping a coefficient of spectral acceleration of 0.065 is obtained. Multiplying this coefficient by the total mass of the structure yields the vertical earthquake reaction in k/ft of wall. The model in the vertical direction assumes a single degree of freedom response.

$$T_{V_i} = k M_i / 2\pi R \qquad (3.1.29)$$

where:

k = coefficient of seismic acceleration in the vertical direction: (0.111 for 1.OE'), (0.0813g for 1.25E)

M_i = mass of containment shell above point i in kips/g

R = radius of containment in feet

Uplift from the Horizontal Earthquake:

The horizontal inertial forces on the containment structure produce overturning movements which in turn produce tension on one side of the containment and compression on the other side in the direction of the earthquake. These forces per foot of wall section are computed by dividing the overturning moment on the section, considering the containment a cantilever beam, by the moment of inertia of the containment as a hollow cylinder. Since the concrete shell is assumed cracked and in tension under the loss-of-coolant accident pressure condition, only the area of the containment vertical rebar and liner are considered in determining the moment of inertia.

The seismic overturning moment above a point i about point i is determined:

$$M_i = RMS\ (M_{im} = F_{irm}\ h_{ir}) \tag{3.1.30}$$

where:

M_{im} = the seismic overturning moment above a point i about point i for mode m

h_{ir} = the distance from the location of forces F_{irm} to the point i

F_{irm} = the horizontal inertial forces on the r segments above point i for mode m

M_i = the root mean square of M_{im} for each node

The moment of inertia is computed by

$$I = \pi t_1 r^3 \text{ (A hollow circular ring)} \tag{3.1.31}$$

where:

t_1 = equivalent thickness of vertical reinforcing steel, including liner in sq. in. per inch of wall

r = mean radius of containment in inches

and $$T_{EQ} = M_i C t_1 / I \qquad (3.1.32)$$

where:

T_{EQ} = vertical force in k/in induced in the containment wall by the seismic overturning moment

C = distance from neutral axis to outermost fiber of containment cross section.

Torsional effects from an earthquake are negligible due to the symmetry of the containment structure and therefore are not considered.

3.1.6 TEMPERATURE EFFECT LOAD

An increase in internal temperature caused by a loss-of-coolant accident has been considered. The maximum temperatures, which do not occur at the same time as the maximum pressures, related to the design (P), 1.25P and 1.5P cases are 247°F, 285°F and 306°F respectively. This increase in temperature causes compressive forces in the restrained liner which in turn induces tensile stresses into the rebar. The equivalent force induced in the containment wall is determined:

$$F_c = A_L \varepsilon_{TL} E_S \qquad (3.1.33)$$

where:

F_c = the equivalent tensile load induced in concrete containment shell by the attempted expansion of the liner

ε_{TL} = final compressive strain in the liner after pressure and temperature conditions and elastic relaxation of the concrete shell have been considered

E_S = modulus of elasticity for the liner steel

A_L = area of liner steel in in^2/ft

In addition to the liner temperature effect on the containment shell the effect of operating thermal gradients through the wall have been considered in analysis of the containment as shown in Section 3.2.5.

The effect of accident thermal gradients has been investigated and found to penetrate less than 10 percent of the containment wall thickness during the maximum temperature-pressure transient following a loss-of-coolant accident. For this reason, the accident temperature transient thermal gradient effect has not been considered in design analysis.

3.1.7 WIND LOAD

The wind load will be determined by considering a conservative wind pressure of 30 psi for ground level up as stipulated in the state building and construction code for the State of New York.

The forces due to the wind loading are given by

$$V_i = P_1 A_i \qquad (3.1.34)$$

where:

V_i = the wind shear at point i

P_1 = the wind pressure of 30 psf

A_i = the projection, perpendicular to the direction of the wind, of the area of containment above the point i

and

$$M_i = P_1 A_i L \qquad (3.1.35)$$

where:

M_1 = overturning moment about point i determined from the wind load

L = the moment arm from the centroid of the projected area above point i to point i

In all cases the magnitude of the design wind loads are less than the seismic loads as shown in Table 4.1; therefore no stresses are calculated.

3.1.8 TORNADO WIND AND MISSILE LOADS

Tornado loads consist of extreme wind including associated pressure difference and missiles. They are assumed to occur independent of any other extreme load condition.

The wind load is considered for three tornado conditions. One includes a tangential velocity of 300 mph and a translational velocity of 60 mph. This load superposition (Case I, Fig. 3.9) depicts a tornado condition where the funnel coincides with the center of the containment. Load pressure distribution patterns that will result due to various locations of the funnel are considered. The structure will be designed for a triangular (Case II, Fig. 3.9) and a rectangular (Case III, Fig. 3.9) wind distribution of 360 mph.

<u>Case I</u>

For Case I, a torsional effect is induced into the containment structure. This torsional effect results from the tangential wind striking the containment building at an angled α from the normal (See Fig. 3.10). The torsional force is due to the component of the wind tangential to the surface of the containment building and is equal to

$$F_t(\text{lbs.}) = AC_D q \quad Sin\alpha \qquad (3.1.36)$$

where:

A = surface area of the containment (FT^2)

C_D = 0.45[2] (coefficient of drag)

q = $0.002558\, V^2$ [2] in pounds per square foot

where V = the wind velocity in miles per hour = 300 mph

α = 45°

This assumption is conservative in that the actual tangential force would be the result of skin friction and the effects would be negligible. The shear force F_t which is a maximum at the juncture of the walls and base slab and varies to zero at the top of the dome is distributed over the containment circumference to obtain a maximum shear force per foot.

The average shear force from the translational velocity of 60 mph is equal to

$$F^{(lbs)} = C_D q\, A^{(2)} \qquad (3.1.37)$$

where:

C_D = .45[5] (coefficient of drag)

q = $0.002558\, V^2$ [2] in pounds per sq. ft.

where V = wind velocity in miles per hr. = 60 mph

A = projected area of the containment normal to the wind direction (FT^2)

The maximum shear equals twice the average shear from above since the shape factor for a hollow circular ring is equal to two. This maximum shear force which occurs at the juncture of the walls and base slab and varies to zero at the top of the dome is distributed over the containment circumference to obtain a maximum shear force per foot.

The total shear per ft. for Case I is equal to 18^k/FT + $.583^k$/FT = 18.583^k/FT. This is less than the smallest earthquake shear of 46.5^k/FT. The seismic steel is designed to resist an earthquake causing 61.5^k/FT which is 3.3 times greater than the tornado shear.

<u>Case II</u>

For Case II, a torsional effect is induced into the containment structure by the tangential wind which is assumed to strike the containment in such a way that one-half of the containment surface is affected by a frictional force F_t.

F_t is calculated by equation 3.1.36

where:

A = one-half the surface area of the containment and V = 360 mph.

The average shear force from the translational velocity of 360 mph is calculated by Eqt. (3.1.37). The average shear force is equivalent to one-half the force from Eqt. 3.1.37 since a triangular load distribution is assumed rather than the rectangular distribution assumed for the 60 mph wind in Case I. The maximum shear force equals twice the average shear from above.

The torsional and translational wind forces which are greatest at the juncture of the walls and base slab and vary to zero at the top of the dome are distributed over the containment circumference to obtain a maximum shear force per foot.

The total shear per ft. for Case II is equal to 12.85^k/FT + 10.4^k/FT = 23.25^k/FT which has a factor of safety of 2.65 with the maximum earthquake shear force for which the seismic steel is designed.

Case III

Case III, considers a 300 mph tangential wind traveling with a forward velocity of 60 mph for a total load of 360 mph with a rectangular distribution. The average shear force from the translational velocity of 360 mph is calculated by Eqt. (3.1.37). The maximum shear force equals twice the average shear from above.

The maximum translational wind force which is greatest at the juncture of the walls and base slab and varies to zero at the top of the dome is distributed over the containment circumference to obtain a maximum shear force per foot.

The total shear per foot for Case III is equal to 21.2^k/FT which has a factor of safety of 2.9 with the maximum earthquake shear force for which the seismic steel is designed.

Since the maximum base shear for Case II is smaller than the base shear from both earthquakes, the seismic steel, which is sized for the earthquake producing the largest base shear, provides an adequate mechanism for resisting all tornado shear loads. Since the tornado acts independently of other severe loads, it is not necessary to do a stress analysis for these smaller loads.

Overturning Moment from Wind Load

The maximum overturning moment is produced by the 360 mph wind with a rectangular distribution in Case III. The overturning moment is calculated from Eqt. 3.1.35.

where:

$$P_i = q\,C_D^{(5)} \qquad (3.1.38)$$

q and C_D are as defined in Eqt. (3.1.37) and equal .45 and 330 psf respectively.

The maximum M_i occurs at the wall base mat juncture and varies to zero at the top of the dome. The maximum overturning moment equals 6.15 x 10^9 #-in which is less than the overturning moment for the smallest earthquake (17.2 x 10^9 #-in).

Missile Loads

The containment structure is designed to resist the following missiles:

1. 4" x 12" wood plank @300 mph
2. 4000# auto at 50 mph less than 25'-0 above the ground

Only one missile is considered acting at any time simultaneously with the 360 mph wind load and 3 psi negative pressure if it is conservative to consider the 3 psi negative pressure. The capability of the containment shell to withstand missile impact was calculated by the procedures presented in Reference 5.

The results of this analysis indicate a percentage depth of penetration equal to $\left(\frac{\text{Pent.}}{3.5'}\right)$ and $\left(\frac{\text{Pent.}}{4.5'}\right)$ for the plank and automobile respectively.

3 psi Negative Pressure

The 3 psi negative pressure is not a design consideration when acting independently or in combination with the wind and/or missile. The containment is designed for a maximum no loss of function factored load pressure of 70.5 psi. In combination with the missile or external wind load the negative uniform pressure is assumed to act radially hence does not contribute to the rigid body failure modes of containment.

Vertical Missile Loads

Vertical missile loads are not a factor in the containment design. Since the height of the containment above grade is more than 25'-0 the auto is not a factor.

A wood plank falling at 90 mph, in the unlikely event that it reaches heights greater than the containment, would produce very small loads in comparison to the horizontal auto missile and therefore is not a design consideration.

3.1.9 LOAD COMBINATIONS

The loads discussed above were combined to design the containment structure as given in Section 2.1.12.

3.2.0 GENERAL STRESS/STRAIN FORMULA

3.2.1 DEAD LOAD STRESS

$\sigma_{T_i} = T_{DLi} / A_{S_i}$ when overall effect is tension (3.2.1)

$\sigma_{C_i} = T_{DLi} / A_{C_i}$ when overall effect is compression (3.2.2)

where:

A_{S_i} = area of vertical steel including liner, per foot of wall

A_{C_i} = area of concrete per foot of wall

T_{DLi} = dead load as defined in Section 3.1.1

3.2.2 DESIGN BASIS ACCIDENT PRESSURE LOAD STRESS

$\sigma = P / A_s$ (3.2.3)

where:

A_s = area of vertical steel or hoops, including liner, per foot of wall

P = pressure induced membrane force per foot of wall.

3.2.3 DISCONTINUITY MOMENT AND SHEAR LOAD STRESS

The stress induced in the containment shell wall from the discontinuity moment is calculated by considering formula (16-1) of the ACI 318-63 Code "Ultimate Strength Design."

$$M = A_{S_1} f_s (d - a/2) \quad (3.2.4)$$

$$f_s = M / A_{S_1} (d - a/2) \quad (3.2.5)$$

where:

$$a = A_{S_1} f_y / .85 f'_c b \quad (3.2.6)$$

and

A_{S_i} = area of steel on the tension side of the containment wall in in^2/ft

f_s = stress in the steel in k/in^2

f_y = yield strength of the steel in k/in^2

f'_c = 3000 psi 28 day design compressive stress of concrete in k/in^2

b = width of cross section: in all cases assumed equal to 12"

d = effective depth of cross section in inches = 45"

M = resisting moment in inch-kips per foot. The basis for this number is Figure 3.1. This is less that the ultimate moment since $f_s < f_y$.

The stress in the stirrups is computed from Equation (17-6) of the ACI 318-63 Code – Ultimate Strength Design.

$$A_v = V's / F_s d (Sin\,\alpha + Cos\,\alpha) \quad (3.2.7)$$

$$f_s = V's / A_v d (Sin\,\alpha + Cos\,\alpha) \quad (3.2.8)$$

where:

A_v = total area of web reinforcement in tension within a distance, s measured in a direction parallel to the longitudinal reinforcement in in^2.

V' = total shear to be carried by web reinforcement in kips

s = spacing of stirrups or bent bars in a direction parallel to the longitudinal reinforcement in inches

f_s = stress in the stirrups in k/in^2

d = effective depth of cross section in inches = 45"

α = angle between inclined web bars and longitudinal axis of member = 45°

3.2.4 BASE MAT STRESS

Stress from the moment is calculated by considering formula (16-1) of the ACI-318-63 code ultimate strength design as shown in Eqs. 3.2.4, 3.2.5 and 3.2.6.

where:

A_s = area of steel on the tension face of the containment base slab in in^2/ft

f_s = stress in the steel in k/in^2

f_c = 3000 psi 28 day design compressive stress of concrete in k/in^2

b = width of cross section – in all cases assumed equal to 12"

d = effective depth of cross section in inches = 100"

M = resisting moment in inch – kips per foot

Stress from the uplift shear is computed from Eqt. (17-6) of the ACI-318-63 code as shown in Eqts. 3.2.7 and 3.2.8.

where:

$$V' = V - V_c \qquad (3.2.9)$$

where:

v = total shear

$v_c = v_c^{bd}$ (3.2.10)

and

v_c = the allowable concrete shear stress or $2\varnothing\sqrt{f'_c} = 93k/in^2$

ϕ = capacity reduction factor = .85

f_s = stress in the stirrups in k/in^2

α = angle between inclined web bars and longitudinal axis member = 45°

b = width of the section = 12 inches

d = effective depth of the cross section = 100"

Additional web reinforcement was also provided on the basis of a minimum spacing of s equal to 0.75d.

Bond stresses in the stirrups are computed by considering the formula

$$\mu = A_v f_s / \varepsilon_o L \quad (3.2.11)$$

where:

μ = the bond stress in k/in^2

ε_o = sum of perimeters of all effective bars crossing the section on the tension side

L = the anchorage length above or below the mid height of the mat. No credit is taken for additional anchorage provided by the bend in the bar.

The allowable bond stress for tension bars with deformations conforming to ASTMA-408 and other than top bars is

$$\mu_A = (.8) \quad \sqrt[6]{f'_c} \quad (3.2.12)$$

where:

μ_A = the allowable bond stress in k/in^2

.8 is the factor allowed by the ACI-318-63 ultimate strength design code for anchorage bond

A finite element analysis was performed on the Unit No. 2 base mat utilizing loads for the three basic loading conditions specified in the Containment Design Report. Since earthquake loads are smaller for Unit No. 3 than for Unit No. 2, due to differences in percent critical damping for the design basis earthquake and the fact that a modal analysis is performed on Unit No. 3, the results of the Unit No. 2 analysis can conservatively be used for Unit No. 3. Maximum hoop moment caused by lack of symmetry of the seismic loading was found to be 454 in.-k/in. This compares with a capacity of 690 in.-k/in. for the in place hoop reinforcing. In all cases tornado loads are smaller than earthquake loads, therefore, no tornado analysis is required.

3.2.5 SEISMIC LOAD STRESS

Horizontal or Vertical Earthquake Effects

$$\sigma = \frac{Load}{A_s} \quad = \quad \text{(For structure in membrane tension)} \qquad (3.2.13)$$

$$\sigma = \frac{Load}{A_c} \quad = \quad \text{(For structure in membrane compression)} \qquad (3.2.14)$$

where:

A_s = area of vertical reinforcing steel, per foot of shell

A_c = area of concrete per foot of shell

Load = force per foot of shell resulting from dead load response to vertical earthquake acceleration or overturning moment induced by horizontal earthquake acceleration.

The basic assumptions considered in the seismic analysis are:

1) Maximum stress in the seismic reinforcing occurs under the action of seismic shear at 90° points from the direction of seismic motion.

2) The liner does not participate in resisting seismic shear.

3) The stress limitations on intersection bars under the combination of pressure plus earthquake shear in one bar may reach 95% of yield and the opposing bar may relieve stress to 0 ksi. Under this consideration only half of the seismic diagonal steel is considered active in resisting earthquake shear at any given instant.

4) The concrete in the containment does not participate in resisting membrane seismic shear.

Thus, the stress can be calculated by considering the shear flow in the wall being resisted by diagonal bars in a hollow ring.

$$A_{S_s} = 1.414 S_f / 2 f_s \tag{3.2.15}$$

$$f_s = 1.414 S_f / 2 A_{S_s} \tag{3.2.16}$$

where:

A_{S_s} = area of diagonal steel per foot, in one direction, measured along a horizontal plane

f_s = stress in the steel in k/in^2

S_f = the shear flow as determined from Eq. 3.1.27

The 1.414 take the 45° angle of inclination of the diagonal bars into account.

3.2.6 TEMPERATURE EFFECT STRESSES

As discussed in Section 3.1.6 temperature considerations must involve both temperature gradient and the interaction effects of the liner on the containment shell. The following development for interaction takes both of these phenomena into account.

Temperature effects as shown in Figure 3.6 are combined with dead load, pressure, and earthquake uplifts in the following manner.

Due to the redistribution of stresses in the rebar, the reinforcing steel is considered to carry an equal amount of tension which must balance the compression in the liner to satisfy $\Sigma F_x = 0$

To satisfy equilibrium conditions:

$$F_{Liner} = F_{Wall} \qquad (3.2.17)$$

$$A_L\, {}^{\varepsilon}TL\, E = -A_s\, {}^{\varepsilon}TL'\, E$$

$$A_L\left[\frac{{}^{\varepsilon}TL_x + {}^{\mu\varepsilon}TL_y}{1-\mu^2}\right]E = -A_s\, {}^{\varepsilon}TL'\, E \qquad (3.2.18)$$

$${}^{\varepsilon}TL'_x = \frac{A_L}{A_S}\left[\frac{{}^{\varepsilon}TL_x + {}^{\mu\varepsilon}TL_y}{1-\mu^2}\right]$$

The 2nd condition which must be satisfied is the deformation compatibility

$${}^{\varepsilon}TLx + {}^{\varepsilon}\Delta T = {}^{\varepsilon}T + {}^{\varepsilon}TL'$$

$${}^{\varepsilon}TLx + {}^{\varepsilon}\Delta T = {}^{\varepsilon}T - \frac{A_L}{A_S}\left[\frac{{}^{\varepsilon}TLx + {}^{\mu\varepsilon}TLy}{1-\mu^2}\right] \qquad (3.2.19)$$

Let $\varepsilon_x = \varepsilon_T - {}^{\varepsilon}\Delta_T$

$$^{\varepsilon}TLx\left[1+\frac{A_L}{A_s\left(1-\mu^2\right)}\right] = {}^{\varepsilon}x - \frac{A_L}{A_s}\left[\frac{\mu\varepsilon TLy}{1-\mu^2}\right]$$

$$^{\varepsilon}TLx = \frac{^{\varepsilon}x}{1+\frac{A_L}{A_S\left(1-\mu^2\right)}} - \frac{\mu\varepsilon TLy}{\frac{A_s}{A_L}\left(1-\mu^2\right)+1} \qquad (3.2.20)$$

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Let $\mu = .25$

$$\varepsilon_{TLx} = \frac{\varepsilon_x}{1+1.067\frac{A_L}{A_S}} - \frac{.25\varepsilon_{TLy}}{.9375\frac{A_S}{A_L}+1} \quad (3.2.21)$$

$$\varepsilon_{TLy} = \frac{\varepsilon_y}{1+1.067\frac{A_L}{A_S}} - \frac{.25\varepsilon_{TLy}}{.9375\ \frac{A_S}{A_L}+1} \quad (3.2.22)$$

to solve Eq. 3.2.18 for the strain in the rebar induced by liner compression solve Eq. 3.2.21 and 3.2.22 simultaneously and insert values for ε_{TLx} and ε_{TLy} into Eq. 3.2.18.

The definitions of the terms used in the above derivations are:

ε_T = strain in the rebar induced by the dead load, pressure and uplift from horizontal and vertical earthquakes.

ε_{TL} = Final strain in liner causing stress or the restrained portion of the potential strain of the liner due to the temperature increase (X or Y direction)

$\varepsilon_{TL'}$ = strain in rebar from stress induced by liner compression. (X or Y direction)

μ = Poissons Ratio = .25

A_L = Area of liner in in^2/ft

A_s = Area of rebar in in^2/ft

E = the modulus of elasticity of steel when the section is in tension $\left(\varepsilon_T + \varepsilon_{TL'} \geq 0\right)$ and modulus of elasticity of concrete when the section is in compression $\left(\varepsilon_T + \varepsilon_{TL'} \leq 0\right)$.

All preceding developments are for the section in tension since this will yield the maximum rebar stress.

$^{\varepsilon}\Delta T$ = the strain in the liner if unrestrained growth were allowed or $\alpha\Delta T$

where:

α = coefficient of thermal expansion in inch/inch/degree F = 6.5×10^{-6}

ΔT = the difference in temperature between the accident temperature felt by the liner and the temperature of the neutral surface (or the point through the wall where no thermal stress exists because of a thermal gradient through the wall).

The gradient is assumed linear with the inside temperature equal to the operating temperature of 120°F and the outside surface temperature of 0°F.

ΔT can be considered in two steps

ΔT gradient = 120° - ^{T}neutral surface

ΔT interaction – ^{T}Max - 120°

This shows the contribution of both the gradient and interaction effects.

The effect of accident temperatures on thermal gradients has not been considered since analysis has shown only 10 percent of the wall located on the inner face of the containment sees any change of thermal gradient during the pressure phase of the accident. In actuality the stresses induced by thermal gradients in the concrete shell are secondary in nature and are largely relieved by the shell cracking under design accident pressure load conditions. For conservatism, however, the operating temperature gradient was included in the stress analysis.

The location and temperature at the neutral surface as shown in Figure 3.7 is found by equating tension on the outside of the neutral surface to compression on the inside assuming the concrete carries no tension. This development of thermal stresses in the rebar is based on the method presented in ACI chimney code [4].

The total compressive force is equal to

$$\tfrac{1}{2}\alpha k^2 t T_x E_c + \alpha k T_x \frac{E_s A_L}{b} + \alpha\left(k - Z_q\right)\frac{T_x E_s A_s}{b} + \alpha\left(k - Z_8\right)\frac{T_x E_s A_s}{b} \qquad (3.2.23)$$

and the total tensile force is equal to

$$\alpha \frac{T_x E_s A_s}{b}\left(Z_7 + Z_6 - 2k\right) \qquad (3.2.24)$$

When equating total tension to total compression the result is the following

$$k^2 + \frac{2k\, nt_L}{t} + \sum_{l}^{i} \frac{2n A_s}{bt} \qquad \left(k - Z_i\right) = 0 \qquad (3.2.25)$$

where:

i = number of layers of reinf. type

k = distance from the liner to the neutral surface divided by the total thickness of the wall

b = rebar spacing in inches

n = $\frac{E_s}{E_c}$

t = total wall thickness

t_L = liner thickness in inches

Z = distance from the liner to the rebar under consideration divided by the total thickness of the wall.

α = coefficient of thermal expansion in inch/inch/degree

F = 6.5×10^{-6}

The temperature at the neutral surface = $(1 - k)\ \Delta T_1$ (3.2.26)

where:

$\Delta T_1 = 120° - 0° = 120°$

to get the final stress in the rebar due to temperature, pressure, earthquake and dead load.

$$\sigma = \left({}^{\varepsilon}t + {}^{\varepsilon}TL'\right) E_s \qquad (3.2.27)$$

3.2.7 TORNADO WIND AND MISSILE STRESSES

Tornado- caused base shears, overturning moments, and internal pressures are all less than design loads used for the containment and, therefore, stresses are not computed for these loads.

The local stresses caused by tornado generated missiles (4000# auto at 50 mph) can be quite large depending on the area of the containment assumed engaged by the missile and mechanisms considered for absorbing the kinetic energy of the missile. Gross shear and overturning effects have been considered in Section 3.1.18. Local structural integrity of the shell is assured by application of emperically derived penetration formulas[5] to determine structural adequacy.

3.3.0 DETAILED ANALYSIS OF CONTAINMENT AT REPRESENTATIVE LOCATIONS

In order to perform a specific comparison between actual stress-strain levels and limiting behavior criteria several representative points on the containment shell to include the base, cylinder and dome are selected for analysis.

The selected points are shown in Figure 3.8 and described in Section 3.3. and through 3.3.8. Detailed tabulation of design loads for the eight points listed are found in Section 3.3.9 with the resultant stresses and allowable stress criteria presented in Section 3.3.10. The detailed determination of the loads and stresses shown in Sections 3.3.9 and 3.3.10 are based on the equations given in Sections 3.1 and 3.2. The actual calculations are in the files of United Engineers and Constructors, Inc., Philadelphia, Pennsylvania.

3.3.1 POINT 1

Point 1 is located in the base mat at a point adjacent to the outside face of the crane wall in a region of negligible uplift, where the mat begins to act as a flat circular plate supported on a rigid non-yielding foundation, and high positive moment, point 1 is located at coordinates H = 53 ft., V = 43 ft.

3.3.2 POINT 2

Point 2 is located in the base mat near the containment wall in a region of high uplift and negative moment adjacent to the knuckle of the liner. Point 2 is located at coordinates H = 67 ft. and V = 43 ft.

3.3.3 POINT 3

Point 3 is located in the cylindrical portion of the containment shell in a region of very high negative discontinuity moment at a point adjacent to the knuckle at the cylinder-base mat junction which is insulated against any thermal effects. It is located at coordinates H = 67.5 ft. and V = 45.7 ft.

3.3.4 POINT 4

Point 4 is located in the cylindrical portion of the containment shell in a region of relatively high positive discontinuity moment adjacent to the cut off point for liner insulation at coordinates H = 67.5 ft. and V = 64 ft.

3.3.5 POINT 5

Point 5 is located in the cylindrical portion of the containment shell, about half way between the base mat and the springline, in a region of membrane stresses only at coordinates H = 67.5 ft. and V = 117 ft.

3.3.6 POINT 6

Point 6 is located in the cylindrical portion of the containment shell at a point just below the springline. It is an area of membrane stress only since the discontinuity effects at the springline are insignificant because the deflection of the dome and cylinder are essentially equal due to the changing steel areas. It is located at coordinates H = 67.5 ft. and V = 191.0 ft.

3.3.7 POINT 7

Point 7 is located in the dome portion of the containment shell at a point just above the springline. It is an area of membrane stress only since the discontinuity effects at the springline are insignificant because the deflection of the dome and cylinder are essentially equal due to the changing steel areas. Point 7 is located at coordinates H = 67.5 and V = 191.0 + ft.

3.3.8 POINT 8

Point 8 is located in the dome portion of the containment shell at a point approximately defined by a 30° arc from the springline in a region of membrane stresses only. The seismic bars are terminated at this point and seismic shear is resisted by hoop and meridional rebar. Point 8 is located at coordinates H = 57.8 ft. and V = 225.8 ft.

3.3.9 SUMMARY OF CONTAINMENT DESIGN LOADINGS

In this Section are presented two tables relative to the design Points 1 through 8 shown in Figure 3.8. In Table 3.1 is shown the material and section properties relative to the eight design points selected while Table 3.2 shows the resultant loads for the points selected which were developed from the equations given in Section 3.1 for the load factors and combinations presented in Section 2.1.12.

3.3.10 SUMMARY OF CONTAINMENT DESIGN STRESSES COMPARED TO CRITICAL STRESS LEVELS

In Table 3.3 is presented the stress resultants for the loads given for selected points in Table 3.2 Section 3.3.9. The Table also presents a comparison between resultant stress and allowable stress levels.

3.4.0 EQUIPMENT HATCH & PERSONNEL LOCK— BOSS DESIGN

3.4.1 INTRODUCTION

There are two large openings in the Indian Point – Unit No. 3 Containment Structure. The Personnel Lock is located in the South East quadrant with a center line elevation of 83' –6 and an opening size of 8' –6 diameter. The Equipment Hatch is located in the North East quadrant of the Containment with a center line elevation of 101' –6 and opening size of 16' –0 diameter. Both of these openings along with their thickened reinforced concrete bosses are located a sufficient distance above the fixed base mat at El. 43' –0 that all moments and shears created at this discontinuity have substantially dissipated in the hatch area.

Both hatch and lock are constructed of ASTM 516 GR 60 (formerly A201 GRB) steel normalized to meet the requirements of ASTM A300. The material has been impact tested to meet the requirements of Section N331 of Section III of the ASME Boiler and Unfired Pressure Vessel Code.

All reinforcing steel in the cylindrical wall and the heavily reinforced hatch areas 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 tensile strength of 90,000 psi and a minimum elongation of 7% in an 8-in. specimen.

Bars No. 14S and 18S are spliced by the Cadweld process only. The splices used to join these bars are designed to develop at least 125% of the minimum yield point stress of the bar.

The plate steel liner inside the cylindrical wall including the hatch areas is carbon steel conforming to ASTM Designation A442-65 Grade 60 "Standard Specification for Carbon Steel Plates With Improved Transition Properties." 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% in an 8-in. gauge length at failure. The liner material is tested to assure an NDT temperature more than 30°F lower than the minimum operating temperature of the liner material. Impact testing was done in accordance with Section N331 of Section III of the ASME Boiler and Pressure Vessel Code.

Internal forces and stresses in the concrete containment shell were determined for the factored load combinations listed in Section 3.4.3.1 of the Containment Design Report for Unit #2 (Docket 50-247). In verifying the adequacy of resistance to these factored loads, capacity reduction factors recommended in ACI 318-63 Building Code Requirements for Reinforced Concrete were applied where applicable.

Under loadings which include incident pressure and temperature, some local yielding of the liner may occur; however, this has no adverse strength implications for the containment wall. Moreover, the ductility of the liner fastening studs is sufficient to tolerate local inelastic buckling without stud failure.

Under load combinations a, b, and c on Page 5A-11 dropping the thermal effects, and with the liner contribution to strength disregarded, calculated rebar stresses do not exceed $\varnothing f_y$ (where $\varnothing$ is the capacity reduction factor). Under load combination a involving a factor not greater than 1.0 on reactor incident, and with the liner stress (and temperature) accounted for, calculated rebar stresses do not exceed $\varnothing f_y$. Under factored load combinations b and c involving a factor greater than

1.0 on reactor incident, and with the liner (and temperature loads) accounted for, a limited amount of local rebar yielding is permitted. These criteria guarantee not only assured resistance to the active loads but also minimize any local inelastic strains which may be associated with stress redistribution due to local rebar yielding.

The hatch and lock are anchored into reinforced concrete bosses by means of stud anchors. Along the Equipment Hatch there are 16 rows of 5/8" ∅ x approximately 15" long studs to extend beyond the first row or hoop rebar with 100 per row around the hatch for a total of 1600 studs. Along the Personnel Lock there are 9 rows of 5/8" ∅ x approximately 15" long studs to extend beyond the first row of hoop rebar with 44 per row around the lock for a total of 396 studs. In the areas adjacent to the penetrations, the liner is thickened to 3/4" and is anchored into the concrete by hooked L – anchors of 1/2" ∅ x 9" long (minimum including 2" hook).

The reinforced concrete bosses are thickened to 7'-6 at the Equipment Hatch and 5'-6 at the Personnel Lock. The bosses have flat outside faces and a smooth transition to the dimensions of the wall beyond the effects of the discontinuities (see Figure 3.11).

The hatch and lock have been designed to withstand the internal Containment pressure plus operating and earthquake loads associated with the design accident in accordance with Section III Subsection B of the ASME Boiler & Pressure Vessel Code – Nuclear Vessels. The anchors have been designed to transmit these loads back into the reinforced concrete boss.

Both the Equipment Hatch and Personnel Lock penetrate the concrete shell. In the case of the 16' ∅ Equipment Hatch, a personnel lock is mounted in the head of the hatch and transmits all pressure loads thru the barrel to the concrete when the inside door is closed. Should the personnel door be left open on this lock, the temperature and pressure loads are transmitted to the lock but not into the concrete due to the space between the lock and hatch. Where the 8'-6" Personnel Lock is mounted in the concrete, the temperature and pressure loads inside the lock are transmitted to the concrete if the inside door is left open. (See Figure 3.12.)

3.4.2 DESCRIPTION OF OPENING REINFORCEMENT

The thickened boss has been heavily reinforced in addition to the dense reinforcing which already exists in the 4'-6 thick Containment cylinder wall. The hoop, vertical and seismic wall reinforcing are bent around the openings to provide continuity of reinforcing and assure flow of membrane forces around the openings. All splices will be by the Cadweld process only. The splices are designed to assure that they will develop at least 125% of the minimum yield point stress of the rebar. Several secondary bars have been terminated by means of mechanical anchorage. At the continuous bar bends, hooked bars are provided to prohibit any local crushing of the concrete. In addition the radius of the bar bends is such that crushing of the concrete will not occur. Due to bending the main bars around the large openings, a void in reinforcing is created on the horizontal and vertical center lines. To prevent any cracking and spalling of concrete and to resist membrane tensions, these voids are filled with added rebar which are terminated by hooks at each end.

To accommodate stress concentrations and discontinuity effects of the opening hoop reinforcing is provided around the opening.

In addition to the membrane forces a moment on the ring is produced by the shear load from the pressure on the door of the hatch tending to cause the ring to rotate inside out. Since the ring is restrained from warping, bending moments occur in the cross section of the ring which are resisted by the additional hoops in the reinforced boss. The hoops are designed to resist the tensile loads in addition to bending mentioned above. Since the ring tends to rotate inside out and detach itself from the Containment shell about its outer boundary, a tensile load is induced on the inside surface of the ring and containment. This is resisted by the main vertical and horizontal reinforcing in the Containment cylinder continuous wall.

Since there is an eccentricity between the center of the wall and the center of the thickened ring, moments causing tension on the inside face of the ring develop. These moments are resisted in tension by the main vertical and horizontal bars which are continuous around the opening. In addition these bars assist in resisting membrane tensile loads.

In addition to the main vertical and horizontal reinforcing in the Containment cylinder wall, the two-way seismic reinforcing in the wall is continuous around the opening, thus increasing the steel area available to carry discontinuity forces and moments.

Transverse shears radial to the center of the containment and in plane shears are resisted by #8 stirrups placed radially to the opening at 6" centers around the opening. Popout shears along the circumference of the opening caused by edge reactions from the pressure against the barrel head are resisted by 2-#9 bars @12" around the opening placed through the cross section perpendicular to the reference plane. These bars are spaced at d/3 to insure that at least one bar will cross a potential diagonal crack through the cross section. One end of the bar will be hooked in order to develop adequate anchorage from the point of crack formation to the end of the bars. In addition to the above mentioned stirrups; concrete, extra stirrups at the voids created by the main horizontal and vertical rebar bending around the opening and inclined horizontal and vertical rebar are also available to resist shear loads. See Figures 3.11 and 3.13.

3.4.3 DESIGN OF OPENINGS

The design of the Unit #3 Equipment Hatch and Personnel Lock is identical to that used in Unit #2 Hatches.

The resultant stresses in the containment shell are modified slightly due to the movement of the containment shell seismic reinforcement toward the outer face to facilitate placement and the 6 percent reduction in total containment reinforcement resulting from the reduced seismic load based on 5 percent rather than 2 percent damping.

The methods used in the design of the equipment hatch and personnel lock were verified by a finite element analysis, the details of which are presented in Section 3.4 of the Containment Design Report of the FSAR for Unit #2.

References

(1) Timoshenko, S. P., Woinowsky-Krieger, S., Theory of Plates and Shells, 2nd Ed., McGraw Hill Book Co., New York, 1959.

(2) Nuclear Reactors and Earthquakes, TID 7024, Division of Technical Information, USAEC August, 1963.

(3) Blume, J., Newmark, N., et al., Design of Multistory Reinforced Concrete Buildings for Earthquake Motion, Portland Cement Association, 1961.

(4) American Concrete Institute, "Specification for the Design and Construction of Reinforced Concrete Chimneys (ACI 505-54)," ACI Manual of Concrete Practice, Part 2, 1967.

(5) Trexel, C.A., Tests and Design of Bombproof Structures of Reinforced Concrete, Navy Department, U.S. Government Printing Office, Washington, 1961.

4.0 CONTAINMENT COMPONENT DESIGN

4.1.0 CONTAINMENT SUMPS

There are three containment sumps that cause projections of the bottom of the containment base mat. The largest is the containment reactor sump which is a key shaped reinforced concrete pit located in the center of the base slab (Figure 4.1). This sump, which is 52.5 feet long and 25'-6" deep, encloses the bottom section of the reactor vessel and the in-core instrumentation leads. The side walls and floor of the sump are 4.5 feet thick supporting the ¼" steel liner. An additional 2 feet of concrete is poured over the liner.

Since the reactor sump walls and floor are poured directly against the rock foundation, rigid support conditions have been considered in the design at the sump structural elements to withstand load. Also, since this sump is located in the central portion of the base slab which is poured directly on the rigid rock foundation, negligible bending shears and moments exist in the base slab at the sump location under all load conditions. The reinforcing steel in the sump includes an extension of the reinforcement with the standard detailing procedures specified in ACI-315 being followed. Temperature steel is included in the sump to meet the requirements of ACI-318.

The next largest sump encloses the intakes for the recirculating pumps and consists of a rectangularly shaped reinforced concrete pit 18 feet by 12 feet in plan and 12 feet deep. The side walls and floor of this sump are 9 feet thick supporting the sump liner with an additional 3 feet covering the pit liner floor and 1 foot covering the liner enclosing the sides of the sump (Figs. 4.2, 4.3, 4.4, and 4.5). As in the case of the reactor sump, the walls and floor of this sump are supported by the rigid rock foundation and the sump is located in a region of negligible bending stresses in the base mat. The walls and floor of the sump are considered structurally as part of the base mat.

The smallest sump encloses the containment sump intake and measures 7.5 feet by 7.5 feet by 5.75 feet deep. It has side walls and floor 7.25 feet thick with a 1 foot covering on the liner. As in the case of the recirculating water sump, the walls of the sump are considered as part of the base mat and are located in a region of negligible bending moment and shear.

The three sumps and in particular the concrete cover over the sump liners, also serve as excellent shear keys in transferring seismic or thermal shear loading from the containment internal structure to the base mat. While it is anticipated most of the shear load would be transmitted by friction between the containment base liner and the containment mat the concrete cover area of the sumps acting alone is capable of transmitting full seismic shear load for a 0.15 earthquake at an average shearing stress of 120 psi.

4.2.0 CONTAINMENT BASE MAT

The containment base mat is a reinforced concrete slab 146 ft. in diameter and 9 ft thick (Figure 4.6). The base slab is designed as a flat circular plate supported on a rigid non-yielding foundation. For loads applied uniformly around the slab, the analysis considers a one foot 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.

4.2.1 SHEAR REINFORCEMENT DESIGN OF SLAB

The limiting loading condition for shear is defined by the 1.25P factored load equation which results in the base mat loads as shown in Figure 4.7. The external shear load per foot of 1 foot wide section of the mat is determined from Eqs. 3.1.18 and 3.1.20.

The maximum shear stress permitted on an unreinforced web subjected to combined shear and bending is given by (ACI-318; Eq. 17-2)

$$v_c = \varnothing\left(1.9\sqrt{f'_c} + 2500\frac{P_w Vd}{M}\right) \qquad (4.2.1)$$

where:

v_c = shear stress carried by concrete

$\varnothing$ = capacity reduction factor for shear (0.85)

p_w = reinforcement ration (As/bd)

V = total shear at section

M = bending moment at section

d = depth of section from compression fact to centroid of tensile steel (100 in)

Solving Eqs. 4.2.1 and 4.2.3 for the loads defined in Figure 4.7 the distance x determined as the cut off point for shear reinforcement is 16.0 ft. or just inside of the crane wall.

The shear load V used for the design of shear reinforcement is determined at a distance d from the edge of the slab. This value for the loading given in Figure 4.7 is 183 k/ft. The shear load which is assumed carried by other than shear reinforcement is determined as shown in Eq. 3.2.10 equal to 108 k/ft.

V_c = vbd = 108 k/ft

where:

v = $2\varnothing\sqrt{fc'}$ (ACI-318, section 1701); $\varnothing = 0.85$

d = effective depth of base mat slab (100 in.)

b = width of wedge shaped section at x = d = 100 in.

The required area of shear reinforcement per foot is determined by Eqs. 3.2.9 and 3.2.7 as shown in Figure 4.8.

4.2.2 MOMENT REINFORCEMENT DESIGN OF SLAB

As in the case for shear, moment was calculated by writing equations for moment in terms of x using the center of the containment wall-base slab juncture as the origin with x increasing toward the center of the containment building. For the 1.5P limiting case the discontinuity moment is 1210 k. ft/ft, the discontinuity shear is 157 k/ft as shown in Figure 4.9. The expressions for the moment as a function of x are shown in Equations 3.1.21 and 3.2.23.

The loading diagram in the mat is shown in Figure 4.9. The equation for moment as a function of x is set equal to zero and the distance x at which the condition of tension in the top of the mat would discontinue is found to be 6.5 feet. The expression for shear is also set equal to zero and the distance x at which the maximum positive moment (1208 k. ft) occurs is found to be 20.6 feet.

The moment steel provided for the maximum negative moment of 1210 k ft/ft which occurs along the perimeter of the slab is also assumed to carry one half of the discontinuity shear of 157 k/ft as an axial load which results in a direct stress of 18.4 psi. The section is designed according to Part IV-B Structural Analysis and Proportioning of Members – Ultimate Strength Design of the ACI-318-63 Code as shown in Section 3.2.3 of this report.

The value of fy used is reduced to 41.6 KSI since 18.4 KSI is taken by the discontinuity shear and the ultimate moment is found to be 1,250 K ft/ft which is greater than the maximum applied negative moment value of 1210 K ft/ft.

For all combinations of pressure, dead load and earthquake loadings which tend to cause uplift in the base slab, the dead weight of the crane wall greatly reduces uplift. This forms a rigid central region in the base slab which is supported on an essentially rigid non-yielding foundation. The model used to analyze this condition is a circular and solid flat plate with a central rigid portion subjected to an external moment (Figure 4.10). The maximum radial stress at the inner edge is given by [(1)].

$$\sigma_R = \beta \frac{M_{EXT}}{a\,t^2} \qquad (4.2.2)$$

where:

M_{EXT} = external overturning moment

β = parameter which depends on ratio of a to the radius of the central rigid portion of the slab.

a = radius of the circular slab (875 in)

t = thickness of the slab

The radial stress due to an internal moment is:

$$\sigma R = \frac{M_{INT} C}{I} = \frac{6 M_{INT}}{t^2} \quad (4.2.3)$$

where:

M_{INT} = internal moment in base slab

C = distance from neutral axis to outer fiber of section

I = moment of inertia of section

By equating Eqs. 2.11 and 2.12, the expression for internal moment as a function of the external overturning moment is:

$$M_{INT} = \frac{\beta M_{EXT}}{6a} \quad (4.2.4)$$

The external overturning moment ^{M}EXT is that due to the seismic shear forces. The maximum positive moment acting on the slab base occurs for the 1.25 P factored load case at the crane wall. The uplift pressure is added to the internal moment due to the seismic overturning moment.

Temperature steel was also added in the base mat to meet the requirements of article 807 of the ACI 318 Code. In the circumferential direction reinforcement is placed in the top and bottom of the base slab. In the central region of the base slab for a radius of 28 feet the temperature steel is placed in an orthogonal grid pattern.

4.3.0 CONTAINMENT CYLINDER WALLS

The analysis of the cylinder was accomplished by the superposition of membrane forces resulting from gravity, internal design basic accident, temperature and pressure and overturning due to earthquake using the factored load equation presented in Section 2.1.12. The cylindrical walls are reinforced circumferentially with steel hoops and vertically with straight bars.

For the vertical axial load in the cylinder the 1.25 P loading condition governs the design. The axial force in the cylinder due to the pressure loading on the dome is given by Eq. 3.1.5.

The uplift force in the cylinder due to the horizontal earthquake is given by Eqs. 3.1.29 and 3.1.32. The dead weight force in the cylinder is obtained by taking the total weight of the dome as the force acting at the top of the cylinder and the total weight of the dome and cylinder as the force acting at the base. The uplift force in the cylinder due to the pressure loading on the dome and the uplift due to the horizontal earthquake are combined with the dead weight load in the cylinder. The resultant load diagram varies from an uplift force of 330 k/ft at the base of the cylinder to 276 k/ft at the springline.

For the hoop direction, the 1.5P case controls since the dead weight and earthquake effects are zero. The force in the hoop direction is given by Eq. 3.1.6.

The seismic loads were determined as described in Section 3.1.5. To provide for the seismic steel, diagonal bars are placed in the cylinder walls in both directions at an angle of 45°.* Seismic steel reinforcement is as shown in Figures 4.11 and 4.12.

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* The design of the diagonal steel is such that its horizontal component is equal to the maximum value of the shear flow which is equal to twice the average shear on the cross-section. Since the diagonals are assumed to act in diagonal tension only, half of the total area of the 45° diagonal seismic bars as assumed active to resist seismic shear effects at any given instant.

4.4.0 CONTAINMENT DOME

The thickness of the dome is small in comparison with the radius of curvature (1/15) and there are no discontinuities such as sharp bends in the meridional curves, therefore the stresses due to dead weight, pressure, or earthquake, were calculated by considering a uniform distribution across the wall thickness. All membrane tensile stresses are assumed taken by the steel reinforcement and none by the concrete unless they are compressive stresses since the concrete is assumed to have no tensile strength.

The membrane analysis of the hemispherical dome has been performed by the superposition of forces resulting from gravity and accident pressure. The dead weight forces in the dome are computed by using the procedure outlined in the Portland Cement Association Bulletin ST55, "Design of Circular Domes." The total vertical dead load acting downward for a given central angle from the apex is given by Eq. 3.1.4.

The meridional thrust (T) is given by Eq. 3.1.3 and the circumferential thrust (H) is given by Eq. 3.1.2.

The membrane force due to the internal design pressure is equal throughout the dome and is given by Eq. 3.1.5.

Analysis has shown that the earthquake effects are small in the dome, therefore the critical design condition is the 1.5P factored load case. The membrane forces due to the 1.5 factored internal design pressure of 70.5 psi are added vectorily to the membrane forces due to 95 percent of dead weight and the total force per foot is divided by the allowable yield stress of the rebar (57 KSI) to determine the area of steel required. All of the combined direct stresses are developed in the reinforcing steel encased in the concrete.

The vertical steel in the cylindrical concrete wall is extended into the dome such that a continuity between the dome and cylinder is achieved. At an angle of 60° from the springline, the 18S bars come together to a 6 inch spacing. The bars are connected to splice plates by means of Cadwell mechanical splices such that for every two bars coming together there is one 18S bar extending beyond this point. 5A-63

At an angle of 75° from the springline the bars again come together to a 6 inch spacing and are cadwelded to a splice plate to increase the spacing to 12 inches. Similarly, the 18S bars are connected to splice plates at an angle of 83° and 86.5° from the springline. At the apex of the dome 18S bars at a constant spacing of 12 inches connect the splice plates which are 3.5° from the center of the dome as shown on Figure 4.13.

To provide the required earthquake resistance the seismic steel in the cylinder is extended into the dome to a point which is 30° above the springline as shown in Figure 4.14.

Above 30° from the springline the membrane steel in the dome is sufficient to carry the seismic shear . The maximum stress in the rebar due to an earthquake is determined by resolution of the principal tensile stress into components parallel the rebar. In addition, the dome liner has sufficient capacity to carry seismic loads under operation or accident conditions.

4.5.0 CONTAINMENT LINER

4.5.1 PURPOSE OF LINER

The purpose of the steel liner, which is attached to the inside face of the concrete shell, is to ensure a high degree of leak tightness in the event of an accident resulting in the loss of reactor coolant and potential release of radioactive material. The liner is attached to the concrete by means of stud anchors so that it forms an integral part of the entire composite structure under all loadings.

4.5.2 DESIGN LOAD CRITERIA

The loads considered in the design of the containment structure, which can create stresses within the component parts such as the liner, are enumerated in Section 2.1.0. The resultant limiting loads in the liner from these specified load combinations for the typical points specified in Section 3.3.0 are shown in Table 4.1

4.5.3 STRESS CRITERIA

The design stress criteria for the liner is based on the philosophy that no gross deformation beyond the elastic limit occurs for all primary membrane loading conditions defined previously.

In this reinforced concrete structure, the design limits for tension member (i.e., the capacity required for the design loads) are based upon ASTM specified minimum allowable stresses for reinforcing steel.

This reinforcement has also been designed so that it is not subject to average stresses beyond the yield point across any section due to the factored loads.

4.5.3.1 Additional Safety Provisions Regarding Stresses

As an additional safety factor, the allowable stress under any given load is reduced from the values referred to above by a capacity reduction factor, denoted as "∅." This reduction provides for the possibility that small adverse variations in material strength, workmanship, dimensions, control and degree of supervision, while individually within required tolerances and the limits of good practice, occasionally may combine to result in undercapacity.

The values of "∅" used for the liner is 0.95.

Thus, for principal compression and tension, the liner stresses are maintained below 0.95 specified minimum yield at normal operating temperature, i.e., 0.95 x 32,000 = 30,400 psi. For shear, the liner stresses are maintained below 0.6 specified minimum yield at normal temperature. (The actual shear stresses are well below this limit.) The actual proportioning of seismic shear between the liner and the concrete shell is dependent upon the relative stiffness of the two elements. Conservatively assuming only the relative stiffness of the steel reinforcement in the concrete shell versus the stiffness of the liner approximately 30 percent of the seismic shear could be transmitted into the liner.

The limiting case governing the contribution of shear stress to direct stress in the liner to determine the maximum principal compressive stress shows the liner capable of carrying 40 percent of the seismic shear before principal yield compression would be reached. The liner plate material is ASTM A-442, Grade 60.

For the Structural Proof Test, primary membrane stresses are maintained within elastic limits.

4.5.4 MISSILE PROTECTION

High pressure reactor coolant system equipment which could be the source of missiles is suitably shielded from impacting on the liner either by the concrete shield wall enclosing the reactor coolant loops and pressurizer or by the concrete operating floor to block any passage of missiles to the containment walls. A structure is provided for the control rod drive mechanism to block any missiles generated from fracture of the mechanisms.

4.5.5 DESIGN AND STRESS ANALYSIS

The reactor containment is a reinforced concrete shell in the form of a vertical right cylinder with a hemispherical dome and a generally flat base, supported on rock. The inside surface of the structural concrete is lined with steel plate anchored in the concrete shell.

Anchorage of the liner to the concrete shell is effected as shown on Figure 4.15 and described below.

Attachment of the dome liner to the concrete is made by a combination of structural steel tee sections welded to the exterior face of the dome plate in two directions at approximately five foot intervals and Nelson Studs which are provided between the tees. The liner for the cylindrical portion of the concrete shell is anchored by means of Nelson Studs welded to the plate at 14 inches vertical spacing and 24 inches horizontal spacing on the 3/8 inch thick plate and 28 inches by 24 inches on the ½ inch thick plate.

The first course of studs is approximately 18 inches above the base slab. Results of the analysis performed for the base slab preclude the need for anchorage of the bottom horizontal liner plate to the concrete base.

The basic design concept for the liner utilizing stud anchorage ductility assures that the studs fail due to shear, tension or bending stress without the stud connection causing failure or tearing of the liner plate.

The design has also taken into consideration the possibility of daily stress reversals due to ambient temperature changes for the life of the plant. Fatigue limit of the studs, verified by extensive testing of the fatigue life of plates with stud shear connectors will exceed the design requirements. Moreover, to accommodate possible fatigue failure in the plate-to-stud weldment, the depth of weld to the liner plate is controlled to avoid impairment of liner integrity.

In general, the stresses in the liner have been determined assuming deformation compatibility with the containment concrete. The exception to this assumption is the base of the cylindrical wall at the juncture with the base slab. The shear capacity of the studs in the vicinity of the juncture points is less than 10 per cent of the shear capacity required to transfer total discontinuity bending stresses into the liner. For this reason stresses induced in the liner by bending of the concrete shell have been neglected.

The design of the liner takes into consideration buckling of the plate under loading. In order to determine the critical buckling stress, the plate is assumed to be hinged along EFGH as shown in Figure 4.16. This assumption corresponds to buckling mode type III as identified in reference 2. The critical buckling stress for the case of equal bi-axial compression of the assumed hinged plate EFGH is 38.1 ksi. The maximum calculated stresses as shown in Table 4.2 and F are -30.4 ksi vertically and –25.0 ksi horizontally and from a Mohr's circle consideration, the normal stress on the assumed hinged plate is –29.0 ksi and the shear stress 2.34 ksi. The shear stresses on the assumed hinged plate is of such low magnitude that no reduction of normal critical buckling stress results.

Since the maximum applied stress of 29.0 ksi is less than the critical buckling stress of 38.1 ksi, the plate will not buckle.

It will be assumed that during the 115% pressure test of the containment at 54 psig, the liner will contribute to the net overall cross-sectional strength of the structure to resist membrane forces. Since the liner will be anchored to the shell by Nelson Studs at appropriate intervals, elastic stability will be assured and the liner will not be loaded beyond a 95% yield. Results of the calculations for the overpressure test indicate maximum stresses of 30.3 ksi in the liner which are within the allowance of 95% of yield.

The liner will make only a small contribution to the structural capability of the total containment under an accident loading condition. It will tend to expand faster than the concrete at increased temperature and therefore will be stressed first in tension due to pressure build up, and then in compression as a result of temperature rise. Insulation material will be applied to the lower 20 on the inside of the liner cylinder to maintain stresses within the design criteria and to ensure elastic stability.

The maximum liner stresses, computed for this condition, is 30.4 ksi, which is within the design criteria.

The stress values at different points in the liner due to the three loading conditions (a), (b) and (c) on the containment structure, described in Section 2.1.12, are summarized in Table 4.2. The results indicate that the calculated maximum liner stresses are in conformance with the criteria. In determining the final stress state both Poisson ratio effects and elastic deformation of the concrete are considered. In all cases, the seismic loads exceed tornado induced loads hence stresses for load case (d) has not been included.

Columns 3 through 9 of Table 4.2 show the stresses resulting from individual load components of the factored load equations. In column 10 is found total resultant liner stress considering the containment wall rigid, that is neglecting the deformation of the liner due to elastic straining of the concrete shell, and no Poisson ratio effects. The stresses corrected for the interaction between liner and concrete shell are presented in column 11 and final liner stress intensities including Poisson ratio effects are shown in column 12.

4.6.0 PENETRATIONS

In general, a penetration consists of a sleeve embedded in the concrete wall and welded to the containment liner. Piping penetrations pass through on embedded sleeve and the ends of the resulting annulus are closed off, either by welded end plates, bolted flanges or a combination of these.* Provision is made for differential expansion and misalignment between pipe or cartridge, and sleeve. The cartridges, however, have no expansion provisions as they are only connected at one end.

Penetrations are designed with double seals so as to permit continuous pressurization during plant operation to prevent outleakage in the event of a loss-of-coolant accident. In addition, small steel channels are welded over all joints in the containment vessel liner to form chambers which also permit continuous pressurization to demonstrate the integrity of the penetration-to-liner weld joint. Pressurizing connections are provided to continuously demonstrate the integrity of the penetration assemblies. Pressure in the penetrations and liner joint channels is maintained at a minimum pressure greater than the calculated peak accident pressure. This is accomplished by the Containment Penetration Pressurization System. This system also allows introduction of Freon or a similar tracer gas for leak detection as may be required should consumption of pressurizing air be excessive. These provisions, in addition to the Isolation Valve Seal Water System, effectively block all containment leakage paths.

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* Electrical penetrations and the equipment hatch pass through an embedded sleeve but the ends are not closed off outside the containment building.

4.6.1 TYPES

4.6.1.1 Electrical Penetrations

"Cartridge" type penetrations are used for all electrical conductors passing through the containment. The penetrations are provided with a pressure connection to allow continuous pressurization. Ceramic type seals are used to provide a pressure barrier for the conductors. Typical electrical penetrations are shown in Figure 4.17.

4.6.1.2 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. In the case of piping carrying hot fluid, the pipe is insulated and cooling is provided to maintain the concrete temperature adjoining the embedded sleeve at or below 150°F. Typical piping penetrations are shown in Figure 4.18.

4.6.1.3 Equipment and Personnel Access Hatches

An equipment hatch is provided which is fabricated from welded steel and furnished with a double-gasketed flange and 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 to prevent both being open simultaneously and to ensure that one door is completely closed before the opposite door can be opened.

Remote indicating light and annunciator situated in the control room indicate the door position status. An emergency lighting and communication 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 to the outer door, from outside is possible by the use of special door unlatching tools.

4.6.1.4 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 inch stainless steel pipe installed inside a 24 inch 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. The 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 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. The fuel transfer penetration is shown in Figure 4.19.

4.6.1.5 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 remotely opened for containment purging but are automatically closed upon a signal of high containment pressure or high containment radiation level. The space between the valves is pressurized above design pressure while the valves are normally closed during plant operation.

4.6.1.6 Sump Penetrations

The piping penetration in the containment sump area 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.

4.6.1.7 Dome Penetration

An opening is located n the dome at the top of the vessel. This opening is for construction ventilation and will be permanently closed at the conclusion of the construction work.

4.6.1.8 Temporary Construction Openings

There are no temporary construction openings.

All personnel locks and any portion of the equipment access door extending beyond the concrete shall conform in all respects to the requirements of ASME Section VIII Nuclear Vessels Code. 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. Liner reinforcements are designed to support penetrations in the appropriate portion of the liner plate during shop testing, shipping and field erection.

The adequacy of penetrations in retaining strength and ductility while preventing leakage is ensured by the following measures:

1. The materials for all components are selected primarily because of their high ductility.

2. By design, all penetrations can withstand all stresses imposed on the as a result of normal plant operation and the hypothetical loss-of-coolant accident. Specifically, the joint between the penetration sleeve and the building liner plate is reinforced with a thickened plate. The sleeve is anchored to the concrete by means of stud anchors welded to a steel ring which is, in turn, welded to the sleeve. The penetration end plates through which the pipes or electric cable pass are designed to withstand the penetration's internal air pressure during normal operation and also containment internal pressure during the hypothetical loss-of-coolant accident.

3. Load transfer around penetrations is based on maintaining continuity of main reinforcing bars which is accomplished by bending of reinforcing to ensure the transfer of tensions, bending moments and shares. At the equipment access opening, a reinforced concrete boss is provided to carry stresses around the opening and to resist bending and torsional moments created by the load transfer. Again, main reinforcement is bent to maintain continuity of stress to ensure load transfer.

4. The line is basically not a load-carrying member and because of its integral relationship with the reinforced concrete is subjected to the strains which the reinforced concrete imposes upon it. Therefore, the criterion at penetrations is one of consistent deformations rather than transfer of load. Nevertheless, the liner is reinforced at each penetration according to the rules set forth in the ASME Unfired Pressure Vessel Code, Section VIII UG-36. An additional conservatism is that the reinforcing requirements set forth in the ASME Code are based on unequal bi-axial stresses, whereas the liner principal stresses, being dependent on reinforcing bar strains, are essentially equal. For the penetrations the maximum stress at the opening is essentially the same as the average nominal stress of the liner.

5. The weldments of liner to penetration sleeve are of sufficient strength to accommodate the stress raisers around the openings. These welds shall adhere strictly to ASME Section VIII requirements for both type and strength. In addition, each weld has a channel placed over it (for pressurization and ultimate leak testing) which adds strength and stiffness to the welded area and assists in reducing stress in the weld and liner plate.

4.6.2.1 Penetration Loading

The penetration sleeves and end plates are designed to accommodate all loads imposed on them. These loads include the following:

1. Internal pressure
2. Concentrated loads imposed by the sleeve anchors to the concrete as the anchors strain in conjunction with wall movement under both operating and accident conditions.
3. Thermal effects due to both gradient and thermal reactions of the particular item passing through the sleeve.
4. Shear, bending and compression due to accident end pressures.
5. 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 the stress limitations imposed by ASME Code Section VIII.

In addition, pipes which penetrate the containment building wall and which are subject to machinery originated vibratory loadings, such as the reactor coolant pumps, will have their supports spaced in such a manner that the natural frequency of the piping system immediately adjacent to the penetrations will be greater than the dominant frequencies of the pump. Pipe line vibration will be checked during preliminary plant operation; and where necessary vibration dampers will be fitted. This checking and fitting will effectively eliminate vibrating loads as a design consideration.

4.6.2.2 Design Computations

Stresses in the penetration sleeves and the liner to which they are attached is determined by compatibility of deformation between the liner and the sleeve. The radial deformation in a plate subject to biaxial stresses is determined by performing an integration of the tangential strains around the periphery of the hole.

$$\sigma_\theta = S - 2S \cos 2_\theta + \left[S - 2S \cos(2\theta - T) \right] \tag{4.6.1}$$

Where:

σ_θ = tangential stress at the boundary of the hole defined at the angle θ from the horizontal axis

S = horizontal stress in the liner

S' = vertical stress in the liner

The displacements are determined

$$\delta D = \frac{1}{E} \int_o^\pi \left(S - 2S \cos 2\theta + \left[S' - 2S' \cos(2\theta - \pi) \right] \right) \; r \sin\theta d\theta \tag{4.6.2}$$

$$\delta D = \frac{r}{E} \int_o^\pi S \sin\theta d\theta - 2S \int_o^\pi \cos 2\theta \sin\theta d\theta + S' \int_o^\pi \sin\theta d\theta - 2S' \int_o^\pi \cos(2\delta - \pi) \sin\theta d\theta \tag{4.6.3}$$

$$\begin{aligned} \int \cos(2\theta - \pi) \sin\theta d\theta &= -\int \cos 2\theta \sin \theta d\theta \\ &= -\int \left(1 - 2\text{Sin}^2\theta\right)(\text{Sin}\,\theta)\, d\theta \\ &= -\int \left(\text{Sin}\,\theta - 2\text{Sin}^3\theta\right) d\theta \\ &= -\left[(-\cos\theta) - 2\left(\frac{\text{Sin}^2\theta \cos\theta}{3} + \frac{2}{3}\int \sin\theta d\theta \right) \right] \\ &= -\left(-\cos\theta + {}^2\!/_3 \sin^2\theta \cos\theta + \frac{4}{3}\cos\theta \right) \Big] \end{aligned} \tag{4.6.4}$$

$$\int cos(2\theta-\pi)\,sin\,\theta d\theta = \frac{-cos\,\theta}{3} - \frac{2}{3} sin^2\theta cos\,\theta$$

$$\delta = \frac{r}{E}\left[-S\,cos\,\theta - 2S\left(\frac{cos\,\theta + \quad 2}{3 \qquad\quad 3} sin^2\theta cos\,\theta\right) - S' cos\,\theta - 2S'\left(\frac{-cos\,\theta}{3} - \frac{2}{3} sin^2\theta cos\,\theta\right)\right]_0^{\pi} \quad (4.6.5)$$

$$\delta = \frac{r}{E}\left[\left(S + \frac{2}{3}S + S' - \frac{2}{3}S'\right) - \left(-S - \tfrac{2}{3}S - S' + 2/3S'\right)\right]$$

$$\delta = \frac{r}{E}\left[2S + \frac{4}{3}S + 2S' - \frac{4}{3}S'\right]$$

$$\delta = \frac{r}{E}\left[\frac{10}{3}S + \frac{2}{3}S'\right]$$

$$\delta = \frac{2}{3}\frac{r}{E}[5S + S']$$ (For Stresses in the Same Direction) (4.6.6)

$$\delta = \frac{2}{3}\frac{r}{E}(5S - S')$$ (For Stresses in Opposite Directions) (4.6.7)

The composite deformation determined in the plate and the sleeve and the resultant stress in the liner and sleeve is determined:

$$^{\Delta}UN = {}^{\Delta}P \text{ (restrained) } + {}^{\Delta}S$$ (4.6.8)

$$^{\Delta}UN = \frac{S_1}{E}(1-v)R + \frac{S_1(t)R^2\lambda}{2Et_s}$$

$$^{\Delta}UN = \frac{S_1}{E}\left[R(1-v) + \frac{{}^{t}pR^2\lambda}{2 \quad t_s}\right]$$

$$S_1 = \frac{\Delta_{UN} E}{R\left[(1-\nu)+\left(\frac{t_p R\lambda}{2t_s}\right)\right]} \qquad (4.6.9)$$

$$S_s = \frac{S_1 t_p R\lambda}{2t_s} \qquad (4.6.10)$$

$$S_s = \frac{\Delta_{UN} E\, t_p R \lambda}{R\left[(1-\upsilon)+\left(\frac{t_p R\lambda}{2t_s}\right)\right]} \quad 2t_s$$

where:

$$\lambda = \sqrt[4]{\frac{3(1-\nu^2)}{R^2 t_s^2}}$$

Δ_{UN} = unrestrained deflection

Δ_p = deflection of liner plate

Δ_s = deflection of sleeve

S_1 = stress in liner

R = radius of penetration

υ = poissons ration

t_p = thickness of liner plate

t_s = thickness of sleeve

E = modulus of elasticity

A summary of liner and penetration stresses is shown in Table 4.3. The assumptions assumed in design are as follows:

1. The liner alone was designed for stress concentration effects while the cracked concrete was ignored.

2. The unrestrained growth is based on maximum growth from a stress concentration consideration.

3. The main stream and mechanical penetrations have been considered in a non-insulated 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.

4. The allowable stress in the sleeve = 56,700 psi except for the stainless steel fuel transfer penetration = 49,500 psi. These values come from Table N-421 and Figure N-414 of the ASME Nuclear Vessel Code Section III.

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).

4.7.0 CONTAINMENT CYLINDER, BASE AND DOME AT POINTS OF DISCONTINUITY

Discontinuity stresses occur at changes in section or direction of the containment shell. The juncture of the cylinder to the dome is a point of discontinuity since, under the internal pressure and temperature design conditions, the cylinder will tend to increase in diameter somewhat differently than the dome. To compute the unrestrained dimensional changes, the dome and cylinder have been considered as steel membranes equivalent to the area of reinforcing steel in the hoop direction. As shown in Section 3.1.3, the unrestrained radial deformation of the dome and cylinder are nearly equal therefore the discontinuity moments and shears are insignificant and there is no steel required at the dome to cylinder juncture due to the discontinuity effects.

The juncture of the cylindrical wall and the base mat is also a point of discontinuity. In determining the discontinuity moments and shears, the base mat was considered as offering complete fixity, therefore the only discontinuity is that due to the unrestrained radial expansion of the cylinder. As for the dome to cylinder juncture, the unrestrained radial expansion of the cylinder has been computed by considering the cylinder to be a steel membrane equivalent to the area of reinforcing steel in the hoop direction. The method of analysis for the discontinuity moment and shear and its distribution into the cylindrical walls is given in Section 3.1.3.

The maximum discontinuity moment at the base occurring under the 1.5P factored load condition is 1210 K.FT/FT and the maximum discontinuity shear is 157/K/FT. The limiting discontinuity moments and shears are distributed as shown in Figures 2.2., 2.3 and 2.4 of this report. The placement of steel to carry discontinuity shears and moments is shown in Figures 4.20 and 4.21.

The required area of shear reinforcing as determined from Eq. 17-6 of the ACI Code is given in Eq. 3.2.7. The allowable value of f_y used as the basis for f_y in Eq. 3.2.7 is reduced from 60 KSI to 47 KSI since part of the stress is assumed taken by the axial force due to uplift. The point where the minimum web reinforcement required is less than the .15 percent of the area b_s the provisions of ACI-318 Code Article 1706b apply.

The allowable shear which may be taken by the concrete alone is found from Article 1701 e) of the ACI Code and is given by:

$$v_c = 3.5\,ø \quad f_c \ (1 + 0.002N/A_g)$$

where:

v_c = allowable shear stress carried by the concrete
f_c = concrete design compressive strength
N = load normal to the cross section where N is negative for tensile loads
A_g = gross area of cross section

References

(1) Roark, R. J., Formulas for Stress and Strain, 4th Ed. McGraw Hill Book Co., New York, 1965.

(2) Diable Canyon Unit No. 1, Pacific Gas and Electric Company, Docket No. 50-275, Supplement No. 4, Section III.

5. CONTAINMENT MATERIAL PROPERTIES, FABRICATION AND ERECTION PROCEDURES

5.1 CONCRETE

Concrete used in the containment structure was designed to have minimum compressive strengths in 28 days of 3000 psi and 4000 psi. The concrete mixes were designed to produce strengths of fifteen percent above the minimum design strengths as determined by the average strengths of three laboratory tests of the specified design mixes including satisfactory plasticity qualities.

The minimum cement factor specified was 5 sks/cu. yd. for 3000# Class and 6 ¼ sks/cu. yd for 4000# Class. The maximum slump permitted was limited to 5 inches, except in localized regions of extreme congestion where 7 inch slump was permitted. Concrete was prepared in ready mix equipment conforming to ASTM Specifications C94.

5.1.1 CEMENT

The cement used was Portland Cement Type II conforming to ASTM designation C-150. Cement used in the ready mix batch process was stored in weather-proof bins so as to prevent deterioration or contamination.

5.1.2 WATER

Concrete mix water was supplied from the drinking water supply of the city of Verplank, New York, and as such is clean, clear and free of significant impurity.

5.1.3 AGGREGATES

Fine aggregate consisted of sand conforming to the requirements of ASTM Specification C-33.

Typical properties of the sand are as follows:

SIEVE ANALYSIS

Sieve Sizes	% Passing by wt.	ASTM C-33 Specifications
3/8"	100	100
#4	97.8	95-100
#8	84.9	80-100
#16	61.8	50-85
#30	42.7	25-60
#50	18.1	10-30
#100	3.3	2-10
Fineness Modulus	2.91	
Specific Gravity (SSD)	2.67	
Absorption %	0.7	
Clay Lumps %	Negative	1.0 Max.
Coal & Lignite %	Negative	0.5 Max.
Material Finer than		
No. 200 Sieve %	0.6	3.0 Max.
Organic Impurities	Standard	Standard
Soundness 5 Cycles, % Loss	10.9	
Unit wt. (dry-rodded) lbs/ft^3	104.3	

Coarse aggregate consisted of crushed gravel conforming to the requirements of ASTM Specification C-33. Typical properties of the crushed gravel are as follows;

COARSE AGGREGATE

30% — 40% Crushed Gravel

SIEVE ANALYSIS

Sieve Sizes	% Passing by wt.	ASTM C-33 Specifications
1 ½"	100.0	100
1"	97.2	95-100
3/4"	71.5	-
1/2"	30.9	25-60
3/8"	12.4	-
#4	0	0-10
Fineness Modulus	7.16	
Specific Gravity	2.67	
Absorption %	0.7	
Clay Lumps %	Negative	0.25 Max.
Soft Particles %	Negative	5.0 Max.
Unit wt. (dry-rodded) lbs/ft^3	102.2	
Magnesium Sulfate Soundness		
5 Cycles, % loss	14.8	18 Max.
Los Angeles Abrasion, % loss	41.7	50 Max.

5.1.4 ADMIXTURES

The admixtures used in the concrete mix design was a plasticizer "Placewell" manufactured by the Union Carbide Corporation. The plasticizer is provided to increase ease of concrete placement in highly congested areas. Air entraining admixture used was Aircon when specified in concrete design mix.

5.1.5 PLACEMENT AND CURING

Placing and Curing of concrete conform to the provisions of Chapter 6 of the ACI 318-63.

5.2. REINFORCING STEEL

Reinforcing steel used for the dome, cylindrical walls and base mat is high-strength deformed billet steel bars conforming to ASTM Designation A-615 “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. The design limit for a tension member (i.e., the capacity required for the design load) was based upon the yield stress of the reinforcing steel. No steel reinforcement experiences average strains beyond the yield point at the factored load except in local areas when subjected to temperature at accident conditions. The load capacity so determined has been reduced by a capacity reduction factor "ø" 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 numbers, the factor "ø" was 0.95, 0.90 for flexure and 0.85 for diagonal tension, bond and anchorage.

5.2.1 CADWELD SPLICES

All reinforcing bar design to carry membrane tension or in the size range 14S and 18S where jointed by means of mechanical butt splices known as a Cadweld splice which is a standard commercial product manufactured by Erico Products Inc., Cleveland, Ohio. All splices used are designed to develop the specified minimum ultimate strength of the ASTM A-615 reinforcing bar or greater even though the specified requirement on splice strength was set at 125 percent of specified minimum yield (83.3 percent of minimum ultimate).

The mean value of the ultimate strength of splices made during any time period shall be equal (as a minimum) to 75,000 psi, plus the standard deviation in strength from the mean ultimate strength. In addition, the mean value of the ultimate strength and the standard deviation shall show, by statistical analysis, that at least 99.0% of all of the splices will have an ultimate strength of 60,000 psi or greater.

Splices shall be monitored by the following procedure. Any splice which, in the judgment of the inspector, does not pass visual inspection shall be cut out and replaced.

a. Bar ends shall be approximately square. They may be torch-cut, sawed or sheared. The cut faces of both re-bar, when inserted into the sleeve, shall be entirely within the specified limits for the size of the bar.

b. Bar ends shall be cleaned of dirt, oil, moisture, concrete, or heavy rust, to a degree of cleanliness as represented by heating the end of the bar uniformly to a surface temperature of 200°F to 300°F, power wire brushing to bare metal, reheating to the same temperature range, and hand wire brushing to remove any resulting dust and/or loose material.

c. The re-bars shall be assembled with their sleeve immediately after cleaning and properly aligned.

d. Preheating is not generally required; however, if the air temperature is below 40°F and/or the humidity is above 80%, the bar ends and sleeve shall be preheated to 100°F in order to remove moisture.

e. If it is necessary to remove a portion of the longitudinal rib on the re-bar in order to fit it into the splicing sleeve, the metal shall be removed by grinding only. In no case shall the entire rib be removed nor shall there be any under-cutting of the rib into the stock material of the re-bar.

Containment wall splices are staggered as specified on the UE & C drawings.

5.3 FORMWORK

Concrete form work was erected to conform to the shape, lines and dimensions of the concrete elements as called for on the drawing and sufficiently tight to prevent leakage of mortar.

For all permanently exposed surfaces of concrete the form facing was constructed of new unscarred plywood, re-used plywood in good condition or metal pans. Forms were removed in such a manner and at such a time as to insure the complete safety of the structure. No areas of the containment concrete structure are in contact with backfill.

5.4 CONTAINMENT LINER

5.4.1 MATERIAL

The steel liner plate is carbon steel conforming to ASMT Designation A-442 "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 ¼-in. thick at the bottom, ½-in. thick in the first three courses except ¾-in. thick at penetrations and 3/8-in. thick for remaining portion of the cylindrical walls and ½-in. thick in the dome. The liner material was impact tested at a temperature 30°F lower than the minimum operating temperature of the liner material. For the liner steel the factor "ø" was 0.95 for tension and compression.

5.4.2 FABRICATION

The steel liner plate was fabricated from hot rolled plate in the Greeneville, Pennsylvania and New Castle, Delaware shop of the Chicago Bridge and Iron Co. The plate was shop fabricated into approximately 9' by 30' section and rolled to desired curvature. The Nelson stud anchors were welded to the containment liner shell after the plate was erected.

5.4.3 ERECTION

The difference between the minimum and maximum inside diameters at any cross section does not exceed 0.25 percent of the nominal diameter at the cross section under consideration. Maximum diameter 135'-2", minimum diameter 134'-10" below elevation +95. Above +95 tolerance does not exceed .50 percent of the nominal diameter of cross section under consideration. The liner was erected true and plumb not to exceed 1/500 of height at cross section under consideration with allowance for 2" buckling in the plates.

Particular care was taken in matching edges of cylindrical and hemispherical sections to insure that all joints were properly aligned. Maximum permissible offset of completed joints was 25 percent of nominal plate thickness.

5.5 LINER INSULATION

To protect the lower portion of the containment liner from severe temperature changes under accident conditions, the first 18 feet (approximately) of the liner is covered with insulation. The basic insulation selected is 7/8" thick urethane foam covered with a 1/2" thick gold bond fire shield gypsum board and a .019" thick stainless steel jacket and backed with asbestos paper cover on the unexposed side.

The insulation was designed to meet the following operational requirements:

1. Normal operating temperature - 120°F.
2. Under accident conditions rise in liner temperature not to exceed 80°F above ambient.
3. Insulation panels rated non-burning in accordance with ASTM procedure D-1692.

5.6 PENETRATIONS

In general, containment penetrations for pipe, electrical conduit, duct or access hatches consist of sleeves imbedded in the concrete section and welded to the containment liner. The weld to the liner is shrouded by a continuously pressurized channel which is used to assure the leak tightness of the penetration to liner weld joint. Differential expansion between sleeve and pipes passing through is accommodated by bellows type expansion joints between the outer end of the sleeve and the outer plate.

5.6.1 MATERIALS

The materials for penetrations, including the personnel and equipment access hatches together with mechanical and electrical penetrations, will be carbon steel, conform with the requirements of the ASME Nuclear Vessels Code and exhibit ductility and welding characteristics compatible with the main liner material. As required by the Nuclear Vessels Code, the penetration materials were Charpy V-notch impact tested to a minimum of 15 ft-lbs at 50°F.

The stainless steel bellows of the hot penetration expansion joints will be protected from damage in transit and during construction by sheet metal covers fastened in place at the fabricator's shop. These can be left in place permanently if there is no interference with nearby piping or equipment.

The specific materials used in penetrations may be found in Section 2.2.6.

5.6.2 DESIGN

Those portions of penetrations not backed up by concrete are designed to meet the requirements of ASME Code Section VIII. Those portions of penetrations backed up by concrete are designed considering strains and stresses compatible with the deformation of the concrete wall sections and as such have the same governing design criteria as does the containment liner. As such, no primary load strains greater than the guaranteed yield point under factored loads are permitted. However, strains due to stress concentrations and other localized secondary load effects are limited to 0.5 percent strain.

5.6.3 FABRICATION

The qualifications of welding procedures and welders have 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 Para. UW-38 Section VIII "Unfired Pressure Vessels."

6. QUALITY CONTROL METHOD AND PREOPERATIONAL TEST PROCEDURES

6.1 QUALITY CONTROL ORGANIZATION AND CHAIN OF COMMAND

The responsibility for implementation of the on-site quality control program for WEDCO rests with the Manager – Site Quality Control who reports directly to the Reliability Manager who in turn reports directly to the WEDCO executive Vice President.

Reporting directly to the Manager-Site Quality Control at the project site are Quality Control Engineers assigned primarily to a specific discipline (e.g., Concrete, Structural, Mechanical, Electrical, and Piping/Welding), Quality Control Inspectors, Clerks, and subcontracted testing service personnel.

No one in this quality control chain of command is directly responsible for production or construction schedules.

WEDCO Site Quality Control and/or the subcontracted testing service personnel conduct the first level inspection and test of all construction of structural elements of the vapor containment building except for the field fabrication and erection of the containment liner and penetrations where the construction subcontractor to WEDCO has first level responsibility subject to audit and surveillance by the WEDCO Site Quality Control.

In all cases, all quality control activity is audited by the Prime Contractor, Westinghouse Electric Corp., the Owner, Consolidated Edison Co., and the owner's Surveillance Group, United States Testing Laboratories.

All necessary records and documentation are compiled and maintained by the WEDCO Site Quality Control.

6.2. SUMMARY OF MATERIAL TEST RESULTS

6.2.1 CONCRETE

Minimum design strengths of 3000 psi and 4000 psi are specified. To date no test cylinders strengths under 3000 psi or 4000 class 28 day strength have been determined. One set of six test cylinders to include three 7 day and three 28 day test cylinders have been tested per each 100 cubic yards placed. Approximately 10,000 cubic yards of 3000# Class and 2000 cubic yards of 4000# Class of concrete have been placed in the containment structure to date.

6.2.2 REINFORCING STEEL

Material mill test reports are required for each heat of steel received. Results of all tests show conformance with ASTM specification requirements. In addition to the mill test reports, random heats of no's 11, 14 and 18 bars are user tested. All tests have met minimum specified strength requirements.

6.2.3 STRUCTURAL STEEL

Various types of structural steel were furnished and erected. Structural steel was furnished to ASTM Specification in job lots substantiated by mill certification covering each job lot.

6.2.4 INSULATION

Letters of certification covering material requirements substantiated by test results are furnished by the manufacturer.

6.2.5 CONTAINMENT LINER

All heats of steel used in the fabrication of the liner plate are covered by mill test certificates showing chemical analysis, mechanical test results, and Charpy impact test results.

Each liner plate is marked or coded to a specific heat of steel. These heat numbers are recorded on the as-built drawings. Material control (heat number) continuity is maintained by subcontractor and checked by WEDCO.

The same method of heat identification, certification, and recordation is maintained for the penetration material as for the liner plate.

Weld rod control (only E 7018 rod used on liner plate) is maintained by subcontractor and audited by WEDCO Site Quality Control.

Dimensions of erected material are checked by the WEDCO engineers and recorded on marked-up drawings. Any dimension found out of tolerance is reported to the subcontractor, corrected and rechecked by the survey group. The correction procedure is approved by Engineering.

6.3 QUALITY CONTROL TESTS ON FABRICATED ELEMENTS

6.3.1 LINER, PENETRATIONS, LOCKS, AND EQUIPMENT HATCH

Nondestructive testing of these items consists of the following:

Coupon Testing – In locations on the liner where radiography is not possible, such as floor plates, and lower course of the shell where back-up plates are used, the subcontractor welds a 2" long overrun coupon which is broken off, marked for location and given to WEDCO for destructive examination or radiography.

a. Vacuum Box Test

Bottom liner plate welds and all liner plate seam welds in the cylindrical walls utilizing back-up plates are vacuum box tested with at least 5 psi pressure differential by the subcontractor. No leaks are permitted. (If any portion of a weld seam is inaccessible for vacuum box testings other forms of NDT will be applied.)

b. Strength Tests

After successful vacuum box testing or spot radiography all liner plate weld channels (bottom, cylinder, and dome) are welded on the seam weld and the channel welds tested by pressurizing the channel with air at 54 psig for 15 minutes. No leaks are permitted. Strength testing shall be by predetermined zones, and includes channels and gaskets of the personnel locks.

c. Leak Test

After strength tests of liner seam welds and channels, these welds and penetration sleeve weld channels, and personnel lock weld channels are leak tested by pressurization to 47 psig with a 20% by weight Freon-Air mixture. The entire run of plate weld and the channel to plate welds are then traversed with a halogen leak detector.

The sensitivity of the leak detector is 1 x 10^{-9} standard cc per second. Any halogen indication indicates a leak requiring repair and retest. In addition, the zone of channels tested is held at test pressure for at least 2 hours, with no indication of drop in pressure.

The strength and leaks tests are also performed on the gaskets and seals on the lock penetrations by pressurizing the space between the gaskets and seals as above.

6.3.2 CADWELDS

All Cadwelds are visually inspected by WEDCO Site Quality Control or its Q.C. Sub-contractor site. Details of Cadwelding operations, operator qualification criteria, testing frequencies and criteria, inspection procedures and acceptance standards are included in the UE &C procedure - "Recommended Procedure for the Testing of Mechanical Type Splices for Concrete" reinforcing bars. Each splice shall be visually inspected in accordance with the following procedure:

a. Properly made splices will have filler metal visible at both ends of the sleeve and at the tap hole in the center of the sleeve.

b. Filler metal will not flow to the very edge of the sleeve due to the gasket action of the asbestos wicking used to seal in the molten filler metal. A recess less than 1/2" will not be cause for rejection.

c. As a result of the Cadweld process, a shrinkage bubble may be visible at the tap hole where the molten metal is introduced and shrinkage fissures and pinholes may be visible at the top of splices. These casting flaws do not adversely affect the physical performance of the splice and, therefore, do not constitute cause for rejection.

Bars or splices which do not meet the requirements above shall be rejected and removed from the structure.

Before any Cadweld crew can be assigned to production work, they shall demonstrate their ability to produce splices meeting the specification requirements.

Each new crew shall be qualified using approved materials and procedures by making five splices of each type and tested to destruction.

Each Cadweld crew shall be qualified to do specific work only to the extent or having performed satisfactory qualification splices, for each category of crew can make only this type of splice. Any crew having prior qualification for the four types of splices (horizontal-straight, horizontal-reducing, vertical-straight, vertical-reducing) shall be deemed capable of making any type of splice required by the project.

Each crew shall be assigned an identification number and this number shall not be re-issued during the life of the project.

For purposes of this work a crew is defined as an operator who has been qualified in accordance with the above procedure and who shall be assigned a competent helper.

Cadweld splices shall be capable of developing tension at least 125 per cent of the specified yield strength of the reinforcing bar, in accordance with the requirements of ACI 318-63, Section 805-d.

Individual splices which do not meet 125% of yield shall be rejected.

6.3.3 STUD ANCHORS ON THE LINER

A procedure is set up whereby after qualification, the first stud welded each day by each welder is tested by cold bending the stud to an angle of 45°. This is repeated after the lunch break.

6.4 PREOPERATIONAL PERFORMANCE TESTING

6.4.1 STRUCTURAL INTEGRITY TEST

After completion of the vapor containment structure the building will be pressurized with air to 54 psig (115% of the design pressure of 47 psig). At pressure levels of 12, 21, 41, and 54 psig, gross deformations are determined and visual inspections are performed. Crack pattern and spacing measurements will be made to determine deformation behavior of the containment. These results will be correlated with the results obtained from the structural test behavior of Unit No. 2 as the test proceeds to ensure that structural behavior of Unit No. 3 is comparable to the successful testing of Unit No. 2.

Instrumentation will consist of invar wire extensometers inside the containment capable of measurement of movement of ± 0.01 in. and mechanical feeler gages used to measure crack width with ± 0.002 in. accuracy.

The range of strains and deformations expected vary from 0 to the following expected maximums:

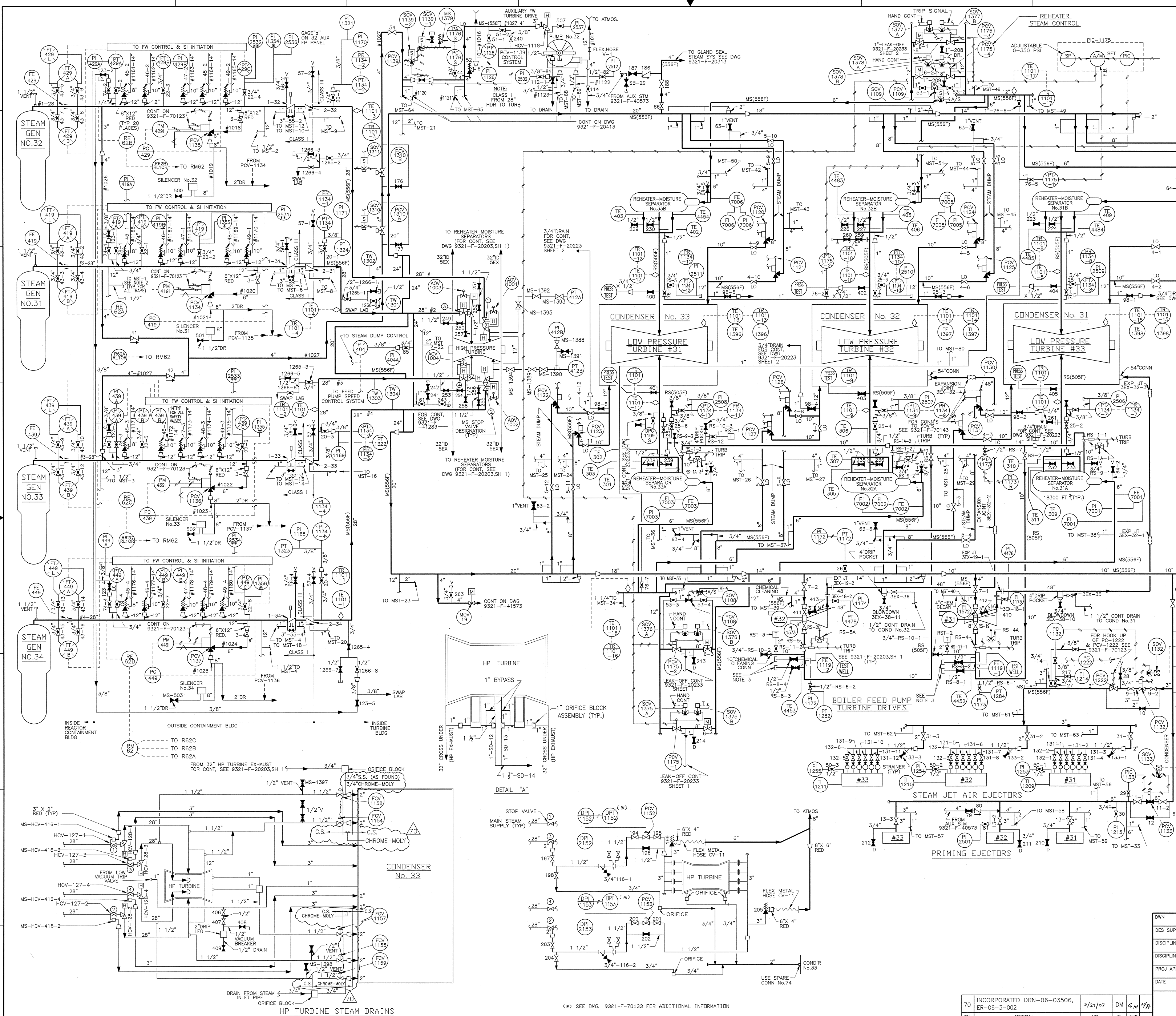
	In.
Vertical elongation (top of mat to top of dome)	1.5
Increase in Diameter	2.0
Crack Width	1/16
Uniform Strain	.002 in/in.

In the quadrant of "boss" around the equipment hatch and personnel lock and in the 10' wide by 5' high areas of the base wall intersection, at the mid-height of the wall, and at the wall dome intersection, concrete surface will be sandblasted or acid etched and detailed measurements of crack width and spacing shall be recorded prior to pressurization at 54 psig, and immediately following depressurization.

6.4.2 CONTAINMENT INTEGRATED LEAK RATE TEST

Following the completion of the Structural Integrity Test (at 54 psig), the containment pressure will be lowered to 50% (min) of the calculated peak accident pressure for purposes of performing the reduced pressure Integrated Leak Rate Test. The 24-hour test will be conducted with the Weld Channel and Penetration Pressurization System depressurized and open to the inside of the containment building. The containment pressure will then be raised to 100% (min) of the calculated peak accident pressure and leakage again determined under the same conditions established during the 50% integrated leak test to correlate leakage rates for purposes of in-service testing.

In addition, a Sensitive Leak Rate Test will be performed to assess the capability of the Weld Channel and Penetration Pressurization System to limit building outleakage. This leak test will be conducted with the containment building at ambient conditions, and the Weld Channel and Penetration Pressurization System pressurized at a pressure greater than the calculated peak accident pressure. Leakage will be limited to less than equivalent 0.2% of the building free volume per day during accident conditions.



CLASS I PIPING

1. MAIN STEAM HEADERS FROM EACH STEAM GENERATOR THROUGH MAIN STEAM (NON-RETURN) CHECK VALVES INCLUDING ALL CONNECTED VENTS,DRAINS, TRAP PIPING TO & INCLUDING TRAP & TRAP BY-PASS VALVES,INSTRUMENTATION.
2. STEAM GENERATOR SAFETY RELIEF VALVES MS-45 THROUGH MS-49.
3. ATMOSPHERIC DUMP VALVES PCV-1134 THROUGH PCV 1137,INCLUDING DISCHARGE PIPING THEREFROM.
4. STEAM SUPPLY TO TURBINE DRIVEN AUXILIARY FEEDWATER PUMP INCLUDING ALL CONNECTED VENTS,DRAINS;TRAP PIPING TO & INCLUDING TRAP & TRAP BY-PASS VALVES,INSTRUMENTATION, RELIEF VALVE WITH DISCHARGE PIPING & TURBINE EXHAUST PIPING.

REFERENCE DWGS

9321-D-20163	FLOW DIA SYMBOLS
9321-F-20493	MAIN STM PIPING SH 1
9321-F-20823	TURBINE DRAIN PIPING
9321-F-21253	AUX BFP TURBINE STM
9321-F-21293	MAIN STM TRAP PIPING
9321-F-22253	BOILER FEED PUMP TURBINE EXHAUST PIPING
9321-C-23003	CONVENTIONAL LINE SCH,CL 1

NOTES:

1. ALL VALVE NUMBERS ARE PRECEDED BY MS UNLESS OTHERWISE NOTED.
2. FOR ALL STEAM TRAP FLOW DIAG SEE DWG'S 9321-F-20413 & 9321-F-20423.
3. FLOW INSTRUMENTS AND ORIFICES WERE REMOVED BY DCP 01-3-013 RS.

** AT ATMOSPHERIC LOCAL CONTROL STATIONS
* AT AUX FW PUMPS LOCAL CONTROL STATION.

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SAFETY RELATED

INDIAN POINT NO. 3 NUCLEAR POWER PLANT

FLOW DIAGRAM
MAIN STEAM

DWN | CHK'D | DES SUPV | DISCIPLINE ENG | DISCIPLINE MGR. | PROJ APPROVAL | DATE

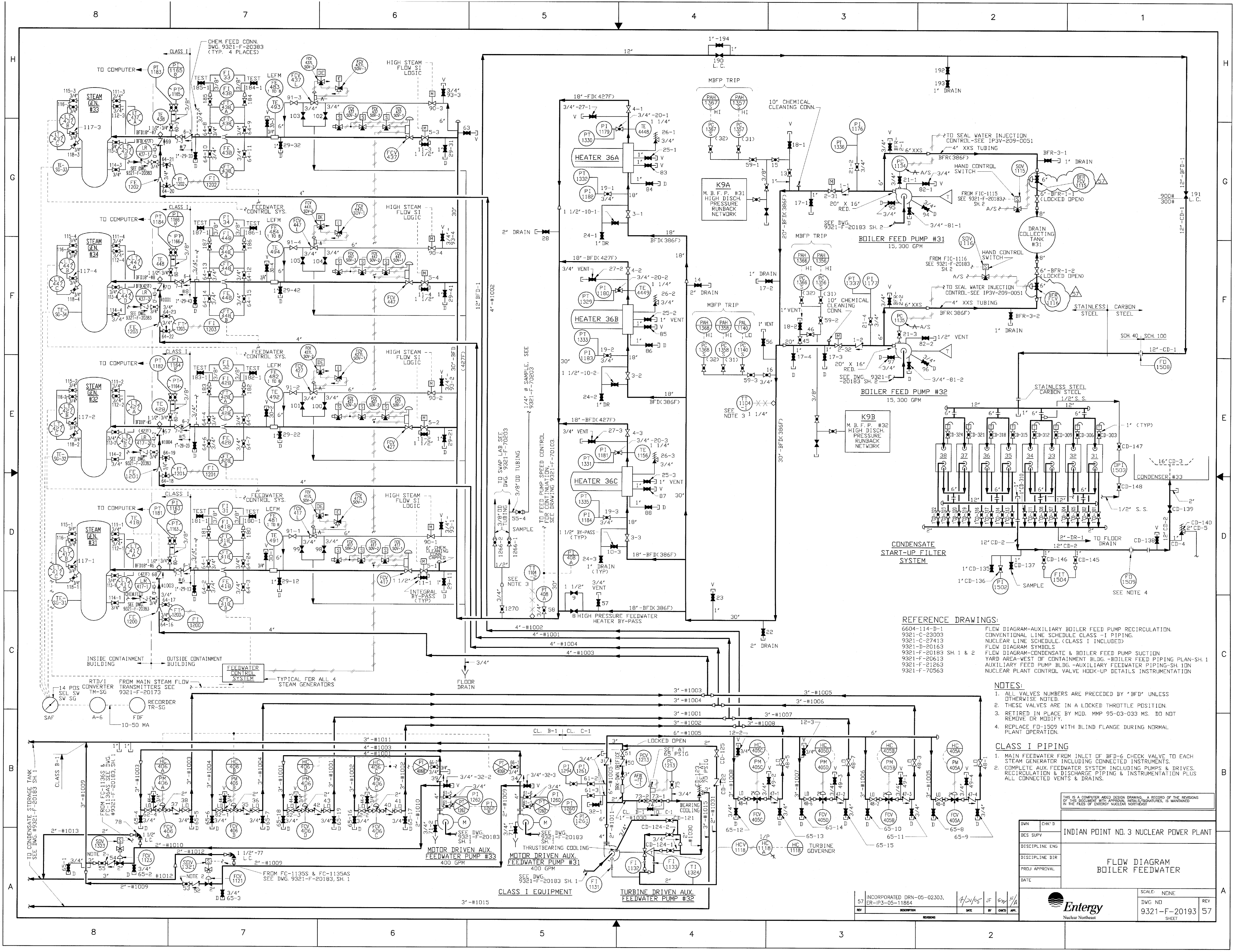
REV	DESCRIPTION	DATE	BY	CHK'D	APP.
70	INCORPORATED DRN-06-03506, ER-06-3-002	3/27/07	DM	GN	N/A

REVISIONS

Entergy
Nuclear Northeast

SCALE: NONE
DWG NO 9321-F-20173 REV 70
SHEET

1 TYPE A/ISI/FSAR



STEAM GEN. #33
STEAM GEN. #34
STEAM GEN. #32
STEAM GEN. #31
TO COMPUTER
FEEDWATER CONTROL SYS.
HIGH STEAM FLOW SI LOGIC
CHEM. FEED CONN. DWG. 9321-F-20383 (TYP. 4 PLACES)
HEATER 36A
HEATER 36B
HEATER 36C
HIGH PRESSURE FEEDWATER HEATER BY-PASS
MBFP TRIP
K9A
M. B. F. P. #31 HIGH DISCH. PRESSURE RUNBACK NETWORK
K9B
M. B. F. P. #32 HIGH DISCH. PRESSURE RUNBACK NETWORK
BOILER FEED PUMP #31
15,300 GPM
BOILER FEED PUMP #32
15,300 GPM
DRAIN COLLECTING TANK #31
TO SEAL WATER INJECTION CONTROL-SEE IP3V-209-0051
CONDENSATE START-UP FILTER SYSTEM
CONDENSER #33
STAINLESS STEEL
CARBON STEEL
INSIDE CONTAINMENT BUILDING
OUTSIDE CONTAINMENT BUILDING
FEEDWATER CONTROL SYSTEM
TYPICAL FOR ALL 4 STEAM GENERATORS
FROM MAIN STEAM FLOW TRANSMITTERS SEE 9321-F-20173
14 POS SEL SW SW SG
RTD/I CONVERTER TM-SG
RECORDER TR-SG
FLOOR DRAIN
TO CONDENSATE STORAGE TANK SEE DWG. #9321-F-20183 SH. 1
FROM FC-1135S & FC-1135AS SEE DWG. 9321-F-20183, SH. 1
MOTOR DRIVEN AUX. FEEDWATER PUMP #33
400 GPM
MOTOR DRIVEN AUX. FEEDWATER PUMP #31
400 GPM
TURBINE DRIVEN AUX. FEEDWATER PUMP #32
CLASS I EQUIPMENT
THRUSTBEARING COOLING
TURBINE GOVERNOR

REFERENCE DRAWINGS:
6604-114-D-1 FLOW DIAGRAM-AUXILIARY BOILER FEED PUMP RECIRCULATION.
9321-C-23003 CONVENTIONAL LINE SCHEDULE CLASS -I PIPING.
9321-C-27413 NUCLEAR LINE SCHEDULE. (CLASS I INCLUDED)
9321-D-20163 FLOW DIAGRAM SYMBOLS
9321-F-20183 SH. 1 & 2 FLOW DIAGRAM-CONDENSATE & BOILER FEED PUMP SUCTION
9321-F-20613 YARD AREA-WEST OF CONTAINMENT BLDG. -BOILER FEED PIPING PLAN-SH. 1
9321-F-21263 AUXILIARY FEED PUMP BLDG. -AUXILIARY FEEDWATER PIPING-SH. 1ON
9321-F-70563 NUCLEAR PLANT CONTROL VALVE HOOK-UP DETAILS INSTRUMENTATION

NOTES:
1. ALL VALVES NUMBERS ARE PRECEDED BY "BFD" UNLESS OTHERWISE NOTED.
2. THESE VALVES ARE IN A LOCKED THROTTLE POSITION.
3. RETIRED IN PLACE BY MOD. MMP 95-03-033 MS. DO NOT REMOVE OR MODIFY.
4. REPLACE FO-1509 WITH BLIND FLANGE DURING NORMAL PLANT OPERATION.

CLASS I PIPING
1. MAIN FEEDWATER FROM INLET OF BFD-6 CHECK VALVE TO EACH STEAM GENERATOR INCLUDING CONNECTED INSTRUMENTS.
2. COMPLETE AUX. FEEDWATER SYSTEM INCLUDING PUMPS & DRIVES. RECIRCULATION & DISCHARGE PIPING & INSTRUMENTATION PLUS ALL CONNECTED VENTS & DRAINS.

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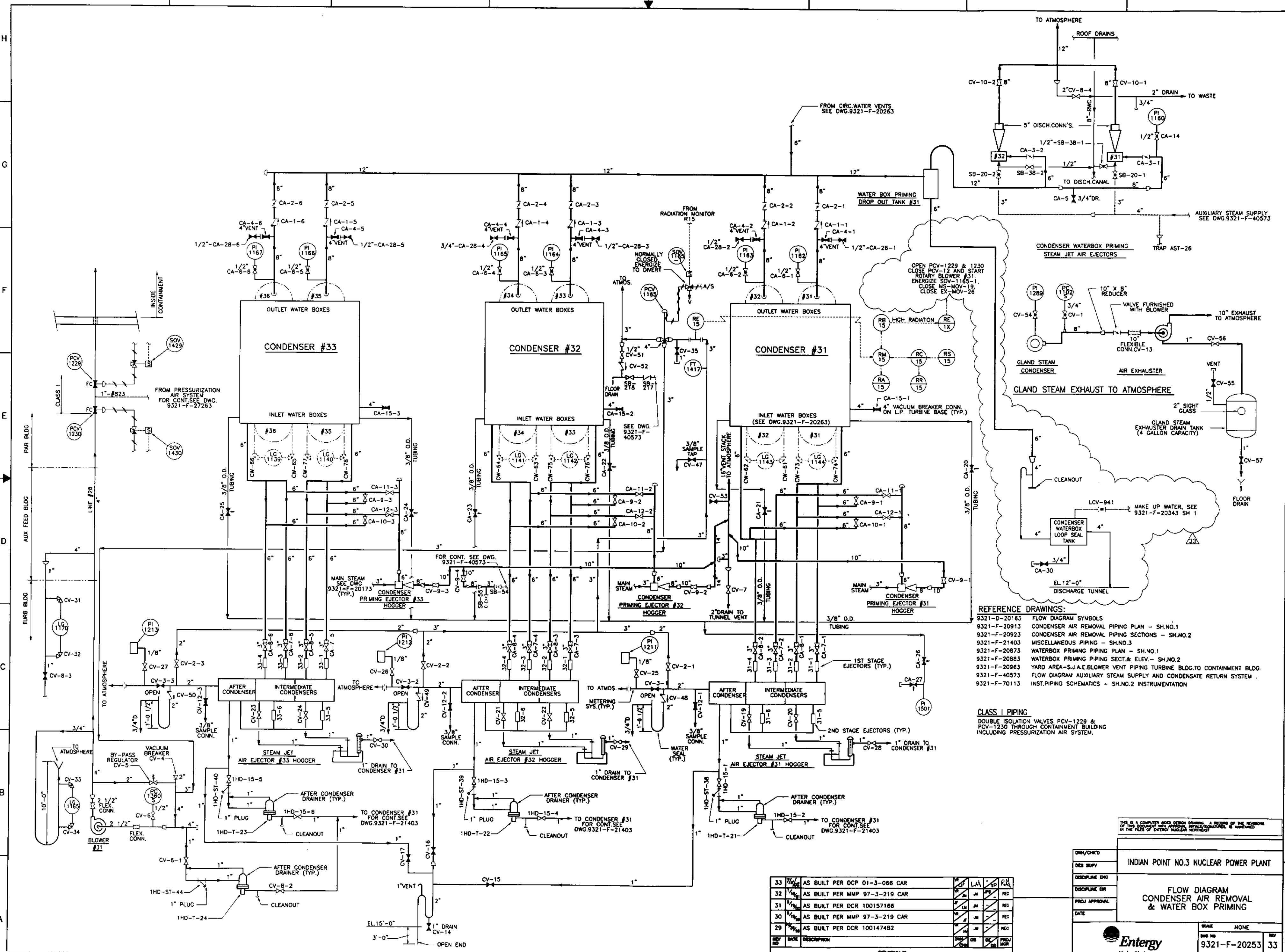
57 INCORPORATED DRN-05-02303, ER-IP3-05-11864 4/21/05 JF N/A
REV DESCRIPTION DATE BY CHK'D APP.
REVISIONS
DWN CHK'D
DES SUPV
DISCIPLINE ENG
DISCIPLINE DIR
PROJ APPROVAL
DATE
INDIAN POINT NO. 3 NUCLEAR POWER PLANT
FLOW DIAGRAM BOILER FEEDWATER
Entergy
Nuclear Northeast
SCALE: NONE
DWG NO 9321-F-20193
SHEET
REV 57



REV	DATE	DESCRIPTION
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32	[illegible]	AS BUILT PER MMP 97-3-219 CAR
31	[illegible]	AS BUILT PER DCR 100157166
30	[illegible]	AS BUILT PER MMP 97-3-219 CAR
29	[illegible]	AS BUILT PER DCR 100147482

REVISIONS

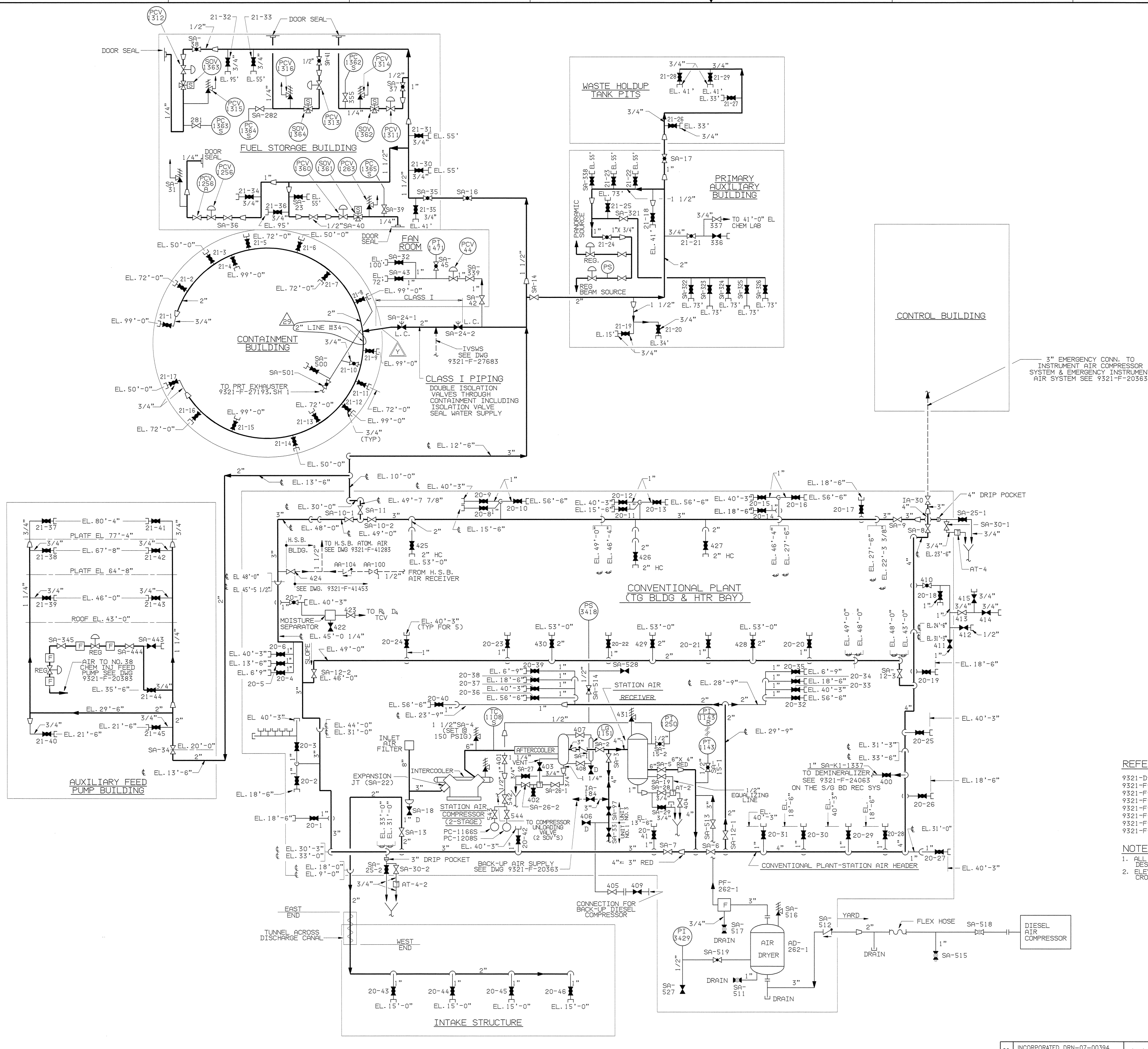
INDIAN POINT NO.3 NUCLEAR POWER PLANT

FLOW DIAGRAM
CONDENSER AIR REMOVAL
& WATER BOX PRIMING

Entergy Nuclear Northeast

SCALE NONE
DWG NO 9321-F-20253 REV 33

TYPE A/FSAR



REFERENCE DRAWINGS:

9321-D-20163 FLOW DIAGRAM SYMBOLS.
9321-F-21133 YARD AREA-STA & INSTR AIR PIPING - SH 1.
9321-F-21143 TURB BLDG & HTR BAY-STA & INSTR AIR PIPING - SH 2.
9321-F-21153 TURB BLDG & HTR BAY-STA & INSTR AIR PIPING - SH 3.
9321-F-26553 HOLDUP TANK PIT COMPOSITE PIPING - SH 1.
9321-F-26843 PRIMARY AUXILIARY BLDG-SERVICE AIR & CITY WATER PIPING.
9321-F-26873 CONTAINMENT BUILDING-SERVICE AIR & CITY WATER PIPING
9321-F-26953 FUEL STORAGE BUILDING-SERVICE AIR & CITY WATER PIPING

NOTES:

1. ALL VALVE NO'S. ARE PRECEDED BY AN "SA" SYSTEM DESIGNATION UNLESS OTHERWISE NOTED.
2. ELEVATIONS ARE SHOWN ON THIS DWG AS AN AID IN CROSS REFERENCE TO THE PHYSICAL DWG 9321-F-21143.

THIS IS A COMPUTER AIDED DESIGN DRAWING. A RECORD OF THE REVISIONS OF THIS DOCUMENT WITH APPROVAL INITIALS/SIGNATURES, IS MAINTAINED IN THE FILES OF ENTERGY NUCLEAR NORTHEAST.

REV	DESCRIPTION	DATE	BY	CHK'D	APP.
29	INCORPORATED DRN-07-00394 ER IP3-07-11661	1/25/07	RM		

REVISIONS

DWN/CHK'D
DES SUPV
DISCIPLINE ENG
DISCIPLINE DIR
PROJ APPROVAL
DATE

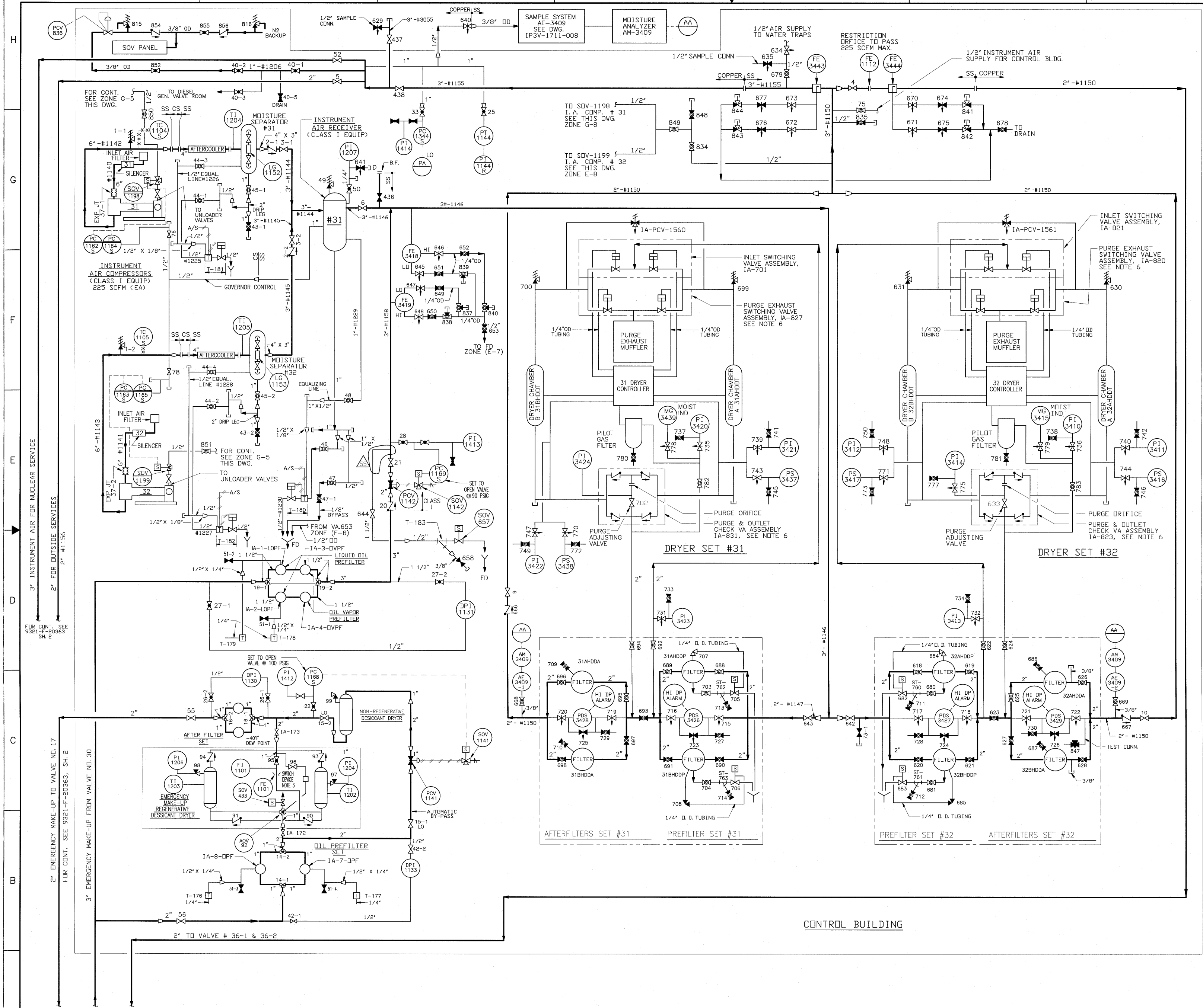
INDIAN POINT NO. 3 NUCLEAR POWER PLANT

FLOW DIAGRAM
STATION AIR

Entergy
Nuclear Northeast

SCALE: NONE
DWG NO 9321-F-20353
REV 29
SHEET

TYPE A/ISI/FSAR



NOTES:

1. ALL 1/2" SOURCE VALVES ARE NO. IA-32.
2. ALL 1" SOURCE VALVES ARE NO. IA-40.
3. SWITCH DEVICE CONTROLLED BY MULTIPLE SOLENOID VALVE.
4. ALL VALVE NO'S. ARE PRECEDED BY 'IA' AS THE SYSTEM DESIGNATION EXCEPT AS NOTED.
5. WORK THIS DWG. WITH 9321-F-20363, SH. 2
6. ASSEMBLY INCLUDES VALVES INTERNAL TO ASSEMBLY, REFER TO IP3V-1659-0013.

REFERENCE DWGS:

9321-D-20163	FLOW DIAGRAM SYMBOLS.
9321-F-21133	STATION AND INSTR AIR PIPING SH NO. 1
9321-F-21143	STATION AND INSTR AIR PIPING SH NO. 2
9321-F-21153	STATION AND INSTR AIR PIPING SH NO. 3
9321-F-21163	STATION AND INSTR AIR PIPING SH NO. 4
9321-F-70083	INSTRUMENT AIR SUPPLY, SH 1
9321-F-70093	INSTRUMENT AIR SUPPLY, SH 2
9321-F-22313	CONTROL BUILDING, RESTRAINT & SUPPORT DESIGN LINE #1146, THRU 1155 & 1158, SH 1
9321-H-20119	INTAKE STRUCTURE BLDG. SLUICE GATE VALVE INSTRUMENT AIR PARTIAL PLAN, SECTIONS & DETAILS.
9321-F-70483	INSTR PIPING SCHEMATICS SH NO.7 INSTRUMENTATION

CLASS I PIPING

1. INSTRUMENT AIR FROM INLET AIR FILTER TO NUCLEAR SERVICES. PENETRATION & WELD CHANNEL PRESSURIZATION SYSTEM.
2. TO SECONDARY PLANT SERVICES THROUGH RESTRICTION ORIFICE ONLY.
3. EMERGENCY MAKE-UP TO PENETRATIONS & WELD CHANNEL PRESSURIZATION SYSTEM FROM INLET OF CHECK VALVE IA-17.
4. INSTRUMENT AIR FOR OUTSIDE SERVICES TO AUX. FEED PUMP HOUSE.
5. EMERGENCY STATION AIR FROM INLET OF IA-20 TO INSTRUMENT AIR HEADER.
6. INSTRUMENT AIR SUPPLY TO THE SERVICE WATER PUMP SCREEN 5.

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REV	DESCRIPTION	DATE	BY	CHK'D	APP.
55	INCORPORATED DRN-05-01229, ER-IP3-05-13651	3-11-05	JF	GH	N/A

REVISIONS

DWN | CHK'D | DES SUPV | DISCIPLINE ENG | DISCIPLINE MGR. | PROJ APPROVAL | DATE

INDIAN POINT NO.3 NUCLEAR POWER PLANT

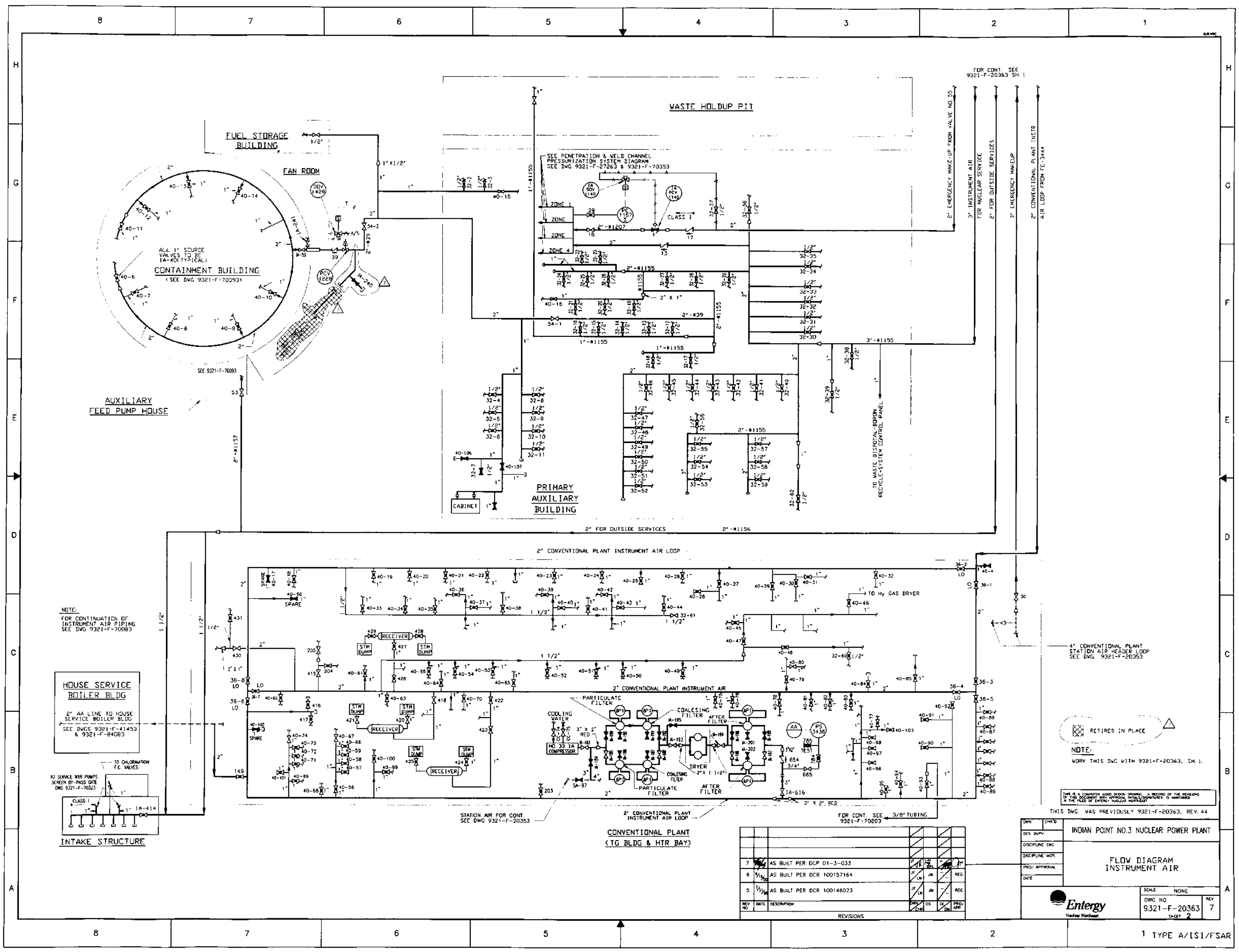
FLOW DIAGRAM
INSTRUMENT AIR

Entergy Nuclear Northeast

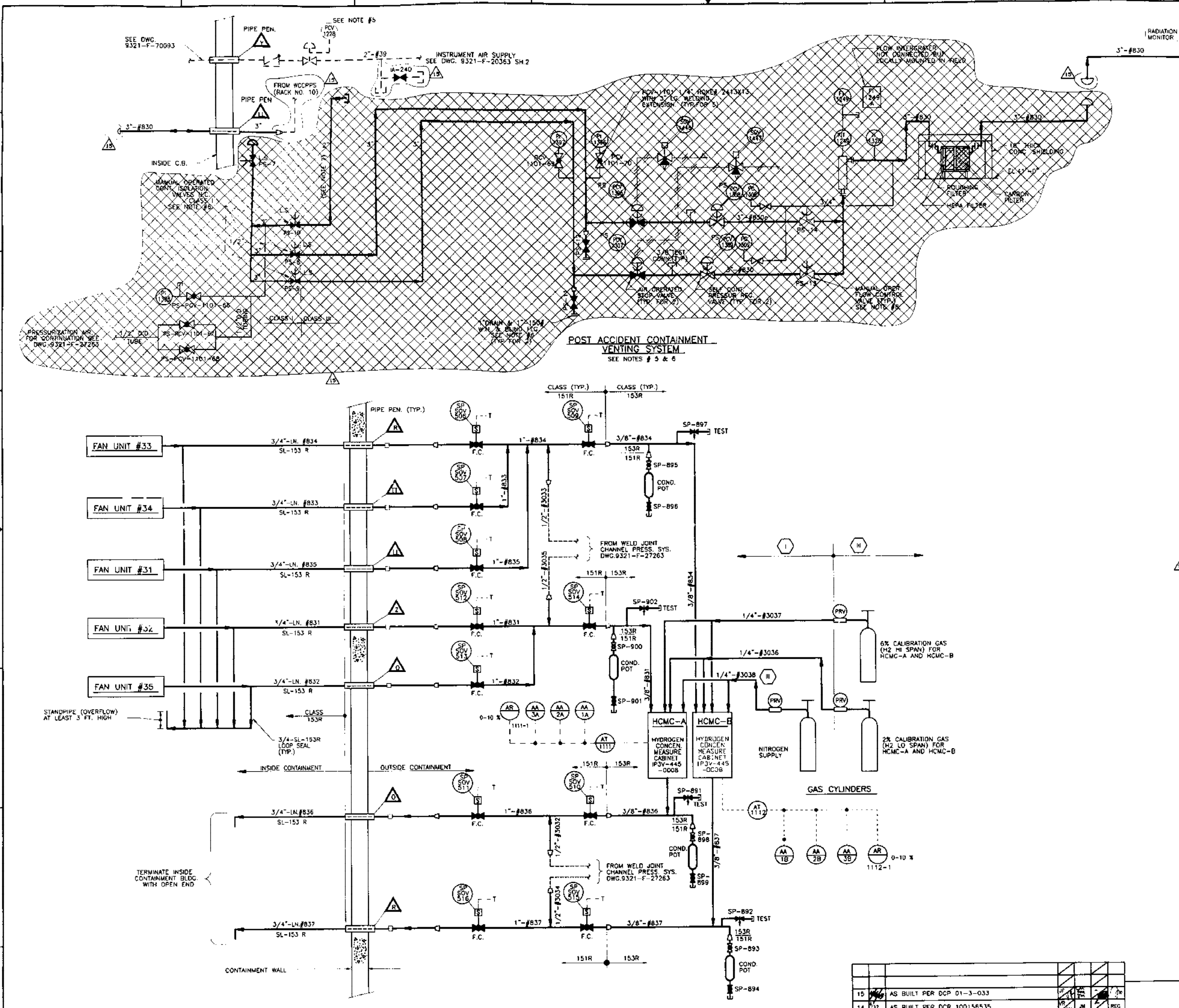
SCALE: NONE

DWG NO 9321-F-20363 SHEET 1 REV 55

TYPE A/FSAR



WASTE HOLDUP PIT
FUEL STORAGE BUILDING
FAN ROOM
ALL 1" SOURCE VALVES TO BE IA-40(TYPICAL)
CONTAINMENT BUILDING
(SEE DWG 9321-F-70093)
AUXILIARY FEED PUMP HOUSE
SEE PENETRATION & WELD CHANNEL PRESSURIZATION SYSTEM DIAGRAM SEE DWG 9321-F-27263 & 9321-F-70353
PRIMARY AUXILIARY BUILDING
CABINET
TO WASTE DISPOSAL-BORON RECYCLE-SYSTEM CONTROL PANEL
FOR CONT. SEE 9321-F-20363 SH 1
2" EMERGENCY MAKE-UP FROM VALVE NO.55
3" INSTRUMENT AIR FOR NUCLEAR SERVICE
2" FOR OUTSIDE SERVICES
3" EMERGENCY MAKEUP
2" CONVENTIONAL PLANT INSTR AIR LOOP FROM FC-3444
2" FOR OUTSIDE SERVICES
2" CONVENTIONAL PLANT INSTRUMENT AIR LOOP
TO H2 GAS DRYER
NOTE: FOR CONTINUATION OF INSTRUMENT AIR PIPING SEE DWG 9321-F-70083
HOUSE SERVICE BOILER BLDG
2" AA LINE TO HOUSE SERVICE BOILER BLDG SEE DWGS 9321-F-41453 & 9321-F-84083
TO CHLORINATION F.C. VALVES
INTAKE STRUCTURE
2" CONVENTIONAL PLANT INSTRUMENT AIR
4" CONVENTIONAL PLANT STATION AIR HEADER LOOP SEE DWG. 9321-F-20353
RECEIVER
STM DUMP
PARTICULATE FILTER
COALESCING FILTER
AFTER FILTER
DRYER
COOLING WATER
NO 33 IA COMPRESSOR
SPARE
RETIRED IN PLACE
NOTE: WORK THIS DWG WITH 9321-F-20363, SH.1.
STATION AIR FOR CONT. SEE DWG 9321-F-20353
2" CONVENTIONAL PLANT INSTRUMENT AIR LOOP
FOR CONT. SEE 9321-F-70203
3/8" TUBING
CONVENTIONAL PLANT (TG BLDG & HTR BAY)
THIS DWG. WAS PREVIOUSLY 9321-F-20363, REV. 44
INDIAN POINT NO.3 NUCLEAR POWER PLANT
FLOW DIAGRAM INSTRUMENT AIR
Entergy
SCALE NONE
DWG NO 9321-F-20363
REV 7
SHEET 2
7 AS BUILT PER DCP 01-3-033
6 AS BUILT PER DCR 100157164
5 AS BUILT PER DCR 100146023
REVISIONS
1 TYPE A/ISI/FSAR



GENERAL NOTES:

1- ALL PIPING IS IN ACCORDANCE WITH SPEC 6604-104-248-4 CLASS 151R & 153R AS NOTED.

2- △ DENOTES CONTAINMENT PENETRATION.

3-

4- S =CONTAINMENT ISOLATION SIGNAL.

5- HAND OPERATOR SHALL BE ADDED TO AIR SUPPLY VALVE PCV-1228 FOR LOCATION SEE DWG. 9321-F-70093.

6- MANUAL OPERATED VALVES SHALL BE AS PER WESTINGHOUSE SPECIFICATION NO.676281 TAG NO. AS SHOWN.

7- AIR SUPPLY LINE BETWEEN LINE #39 AND INLET TO VALVE PS-10 IS CARBON STEEL.

REFERENCE DRAWINGS:

9321-F-27263	FLOW DIAGRAM PENETRATION & LINER WELD JOINT CHANNEL PRESSURIZATION SYSTEM
9321-F-26863	CONT.BLDG.- POST ACCIDENT CONT.AIR SAMPLING SYSTEM
9321-F-40793	POST ACCIDENT CONTAINMENT VENTING SYSTEM-PLANS,SECTIONS & DETAILS

LEGEND:

SUPPLY AIR

CONTROL AIR

15 RETIRED IN PLACE

NOTE:
ALL PIPING ON THIS DWG. IS SEISMIC CLASS I EXCEPT AS NOTED

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REV NO	DATE	DESCRIPTION	DWN/CHK	DS	DE/DM	PROJ APP.
15	[illegible]	AS BUILT PER DCP 01-3-033	[illegible]	[illegible]	[illegible]	[illegible]
14	4/12/00	AS BUILT PER DCR 100156535	VR/LM	JM		REG
13	2/11/98	AS BUILT PER DCR 100143086	IS/RM	JM		REG

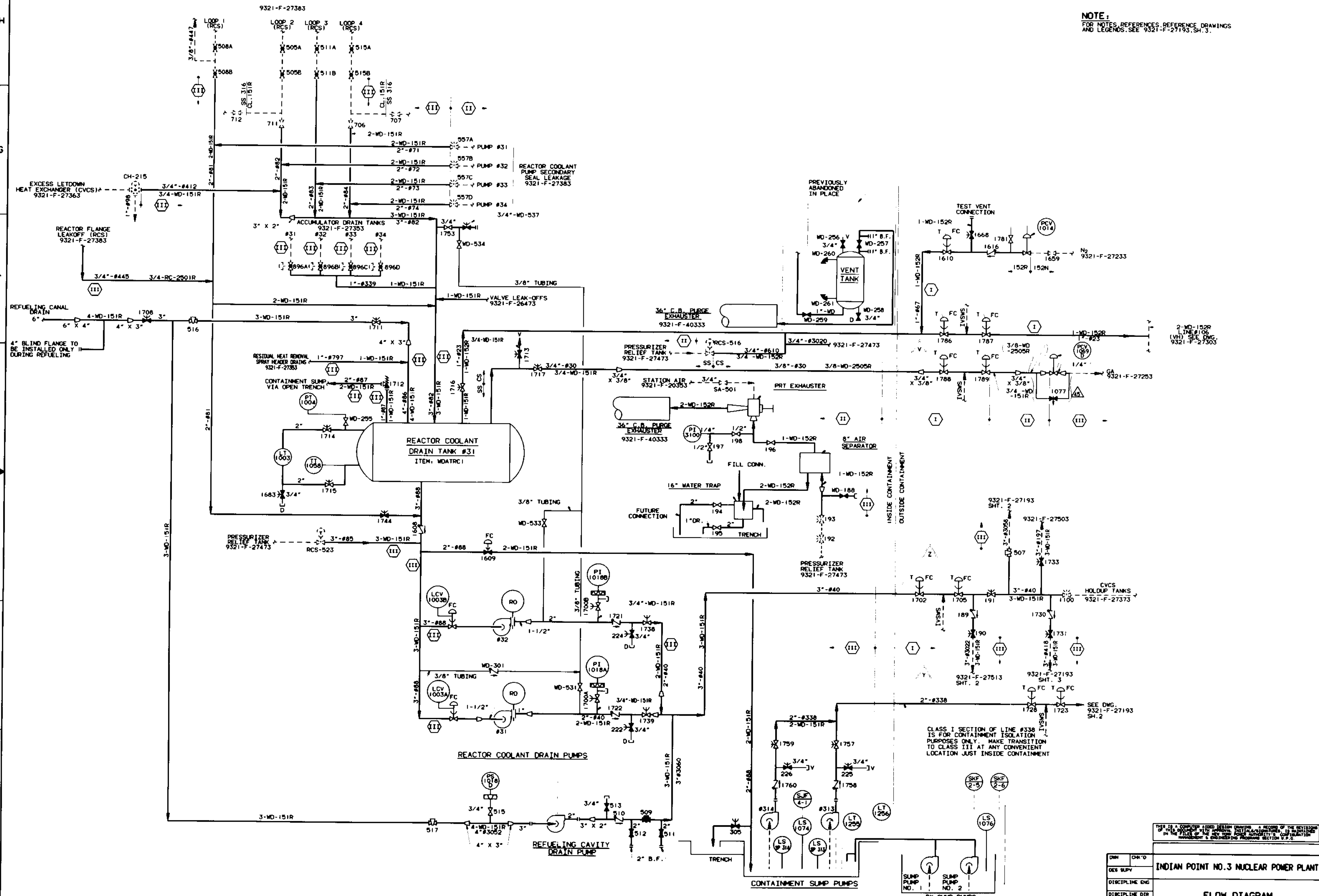
REVISIONS

INDIAN POINT NO.3 NUCLEAR POWER PLANT

FLOW DIAGRAM
CONTAINMENT HYDROGEN CONCENTRATION MEASUREMENT & POST ACCIDENT CONTAINMENT VENTING SYSTEM

Entergy
Nuclear Northeast

SCALE: NONE
DWG NO 9321-F-26533
REV 15
TYPE A/ISI



45	9/3/97	AS BUILT PER MMP 95-3-325 GMD					42		AS BUILT PER DCR'S 100124492		MBH		REG
44		AS BUILT PER DCR 100139985		JH		REG	41		AS BUILT PER DCR'S 100118181 & 100123787		MBH		-
43		AS BUILT PER DCR 100028276		MBH		-	40		AS BUILT PER DCR'S 100121636, 100121638, & 100121851		MBH		REG
REV NO	DATE	DESCRIPTION	DWN/CHK	DS	DE/DD	PROJ MGR	REV NO	DATE	DESCRIPTION	DWN/CHK	DS	DE/DD	PROJ MGR

REVISIONS

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DWN | CHK'D | DES SUPV | DISCIPLINE ENG | DISCIPLINE DIR | PROJ APPROVAL | DATE

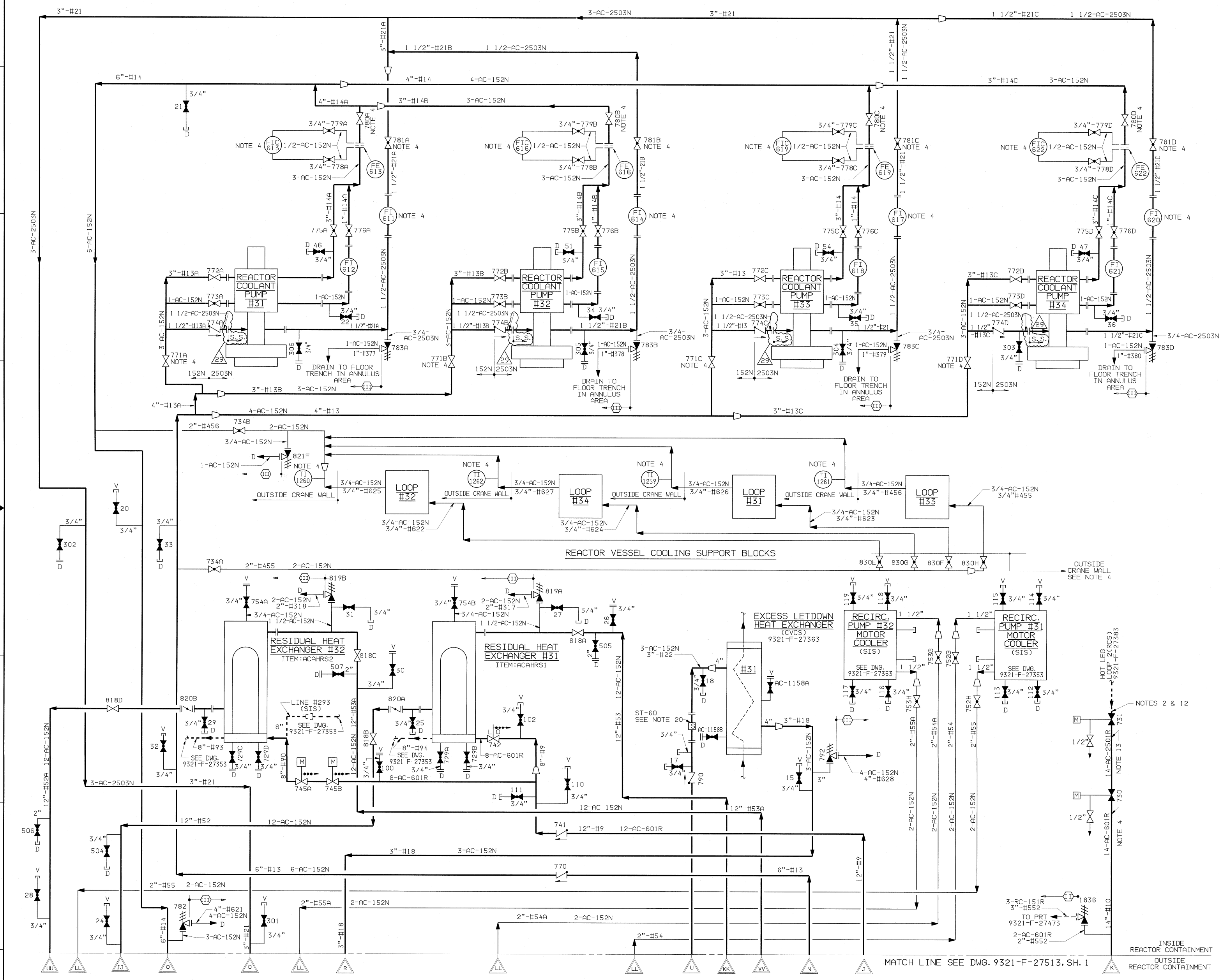
INDIAN POINT NO.3 NUCLEAR POWER PLANT

FLOW DIAGRAM
WASTE DISPOSAL SYSTEM
SHEET NO.1-CONTAINMENT

NewYorkPower Authority

SCALE NONE
DWG NO 9321-F-27193 SHEET 1
REV 45

TYPE A/ISI/FSAR



REFERENCES:

1. PROCESS FLOW DIAGRAM SHEET #1 Ⓦ DWG. 540-F-889
2. DEFINITION OF SYMBOLS
   Ⓦ E. SPEC. G675176 REV. 2 AND
   Ⓦ E. SPEC. G676164 REV. 0
3. INSTALLATION OF INSTRUMENTATION
   Ⓦ PROC. SPEC. CAP 294367 REV. 1
4. MATERIAL SPEC. PIPE AND FITTINGS
   NYPA SPEC. TS-MS-024
   Ⓦ E. SPEC. G676398 REV. 0

REFERENCE DRAWINGS:

9321-F-27513	AC	AUXILIARY COOLANT SYSTEM SHEET 1 & 2
9321-F-27193	WD	WASTE DISPOSAL SYSTEM SHEET 1-CONT.
9321-F-27303	WD	WASTE DISPOSAL SYSTEM-GAS
9321-F-27353	SI	SAFETY INJECTION SYSTEM SHEET 1
9321-F-27503	SI	SAFETY INJECTION SYSTEM SHEET 2
9321-F-27383	RC	REACTOR COOLANT SYSTEM SHEET 1
9321-F-27283	NED	NUCLEAR EQUIPMENT DRAINS
9321-F-27243	PW	PRIMARY MAKE-UP WATER SYSTEM
9321-F-27453	SP	SAMPLING SYSTEM
9321-F-27463	IV	ISOLATION VALVE SEAL WATER SYSTEM
9321-C-27413		NUCLEAR LINE SCHEDULE

NOTES:

1. VALVES ARE NORMALLY INSTALLED WITH FLOW UNDER SEAT: EXCEPTIONS ARE VALVES, 831A & 831B.
2. SUCTION CONNECTION IS LOCATED IN BOTTOM HALF OF REACTOR COOLANT PIPING ON 45° ANGLE TO VERTICAL.
3. SPRING LOADED CHECK VALVE
4. LOCATED IN ANNULAR RING BETWEEN SECONDARY SHIELD AND CONTAINMENT WALL.
5. VALVE IS LOCATED CLOSE TO FUEL PIT WALL.
6. VACUUM BREAKER SUPPLIED WITH TANK.
7. PRESSURE TAP LOCATED PER Ⓦ I & C STANDARDS.
8. CENTERLINE OF WALL PENETRATION FOR THE SPENT FUEL PIT PUMP SUCTION LINE IS SIX (6) FEET BELOW THE TOP OF THE SPENT FUEL PIT WALL.
9. SPENT FUEL PIT PUMP DISCHARGE PIPE TERMINATES IN THE POOL AT ELEVATION 74'-4 3/4"
10. ADDITIONAL VENTS & DRAINS MAY BE INSTALLED. SEE PIPING LAYOUTS.
12. REACTOR COOLANT PRESSURE INTERLOCK PREVENTS ISOLATION VALVE OPENING ABOVE SET PRESSURE.
13. SCHEDULE 140 PIPE
15. ELBOW FLOW METER.
16. ONLY IF PLOCAP IS INSTALLED.
17. EMERGENCY SERVICE WATER CONNECTION.
18. PIPE RUN SUBMERGED IN SPENT FUEL PIT HAS SIPHON HOLE TO MAINTAIN A SPECIFIC WATER LEVEL AND PREVENT WATER FROM BEING SIPHONED OFF.
19. ALL VALVE NO'S. ARE PRECEDED BY "AC" AS THE SYSTEM DESIGNATION. EXCEPT AS NOTED.
20. TEMPORARY STAINER (ST-19, ST-21, ST-22, ST-23, ST-60, ST-61, ST-62, ST-63, ST-64, ST-65, ST-66 & ST-67) IS PLACED IN THE FLANGED SPOOL PIECE DURING INITIAL FLUSHING OPERATIONS. STRAINER MUST BE REMOVED BEFORE PLANT START-UP. CAPPED LINE IS CONNECTED TO PRESSURE GAUGE DURING INITIAL FLUSHING FOR STRAINERS ST-65 & ST-66 USE EXISTING DRAIN AND FLUSH CONNECTIONS FOR PRESSURE GAUGES. STRAINER BASKET/ SCREEN HAS BEEN REMOVED FROM ST-19, ST-21, ST-22, ST-23, ST-60, ST-65, ST-66 & ST-67. ST-20, ST-54, ST-55 & ST-56 HAVE BEEN REMOVED PER MWR93-10104, 93-10105, 93-10106, & 93-10107; ST-61, ST-62, ST-63 & ST-64 HAVE BEEN REMOVED PER MWR93-10168, 93-10163, 93-10167, & 93-10161.
21. RCV-017A & RCV-017B HAVE HAD AIR & CONTROL SIGNAL REMOVED AND ARE NORMALLY CLOSED.
22. FLANGE AND ELBOW DOWNSTREAM OF AC-803 MAY HAVE INTERNAL SURFACE HARDENED BY DIFFUSION ALLOYING PROCESS.

LEGEND:

RWST	REFUELING WATER STORAGE TANK
ECC	EMERGENCY COOLING CONNECTION
DH	DRAIN HEADER
LO	LOCKED OPEN
FO	FAIL OPEN
FC	FAIL CLOSED
T	TRIP OR CLOSE ON HIGH CONTAINMENT PRESSURE
V	LOCAL VENT
D	LOCAL DRAIN
S	SAFETY INJECTION SIGNAL
ST	TEMPORARY STRAINER
△	CONTAINMENT PENETRATION
⬡	SEISMIC CLASSIFICATION

NOTE:
ALL PIPING ON THIS DWG. IS SEISMIC CLASS I EXCEPT AS NOTED

NOTE:
WORK THIS DWG WITH DWG'S 9321-F-27513, SH 1 & 2.

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REDRAWN FROM Ⓦ DWG. 685J428 & 110E364

REV	DESCRIPTION	DATE	BY	CHK'D	APP.
29	INCORPORATED DRN-05-03377, ER IP3-05-17496	6/20/05	GH	Rui	C/A

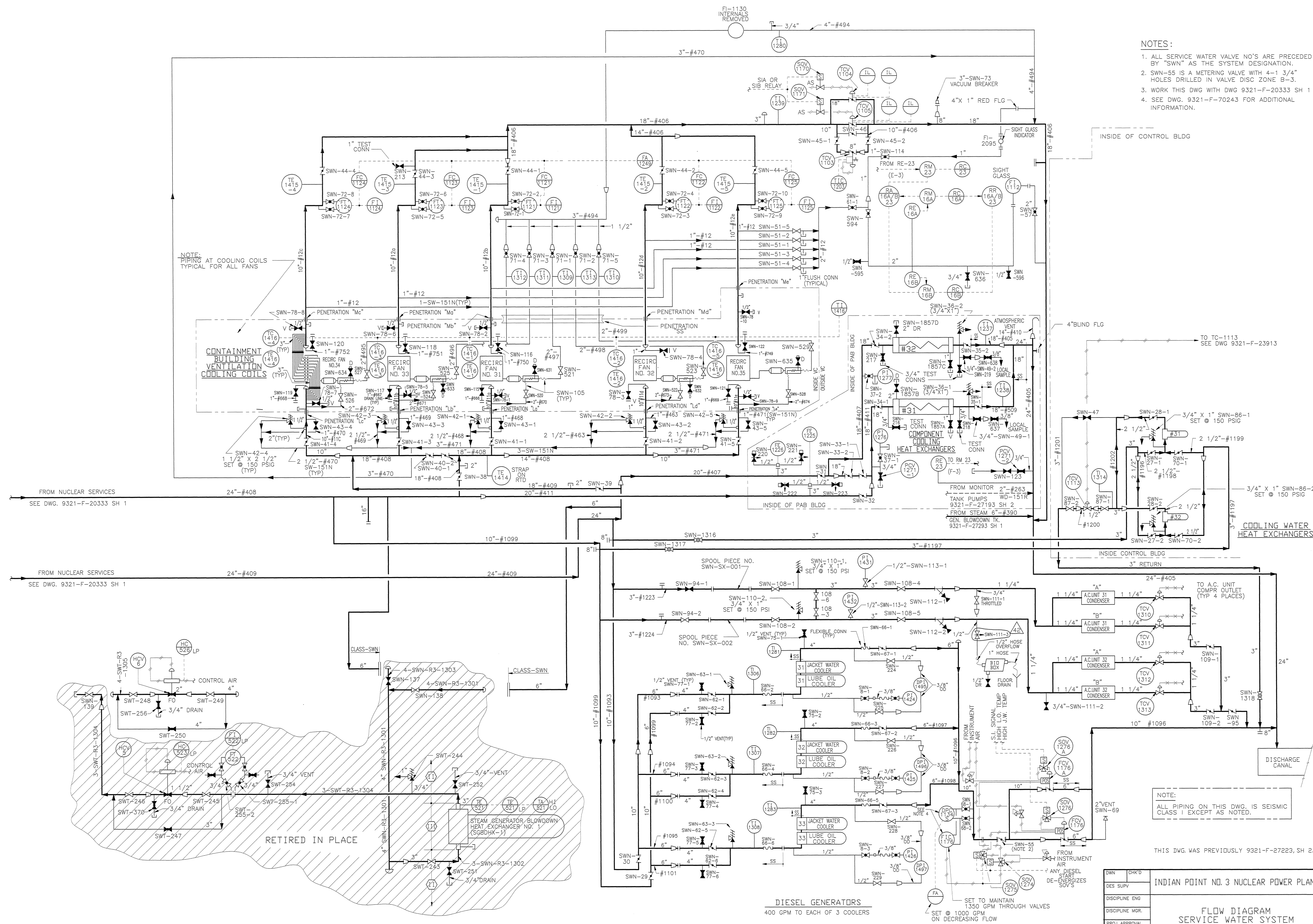
REVISIONS

DWN/CHK'D	INDIAN POINT NO. 3 NUCLEAR POWER PLANT
DES SUPV	
DISCIPLINE ENG	FLOW DIAGRAM AUXILIARY COOLANT SYSTEM INSIDE CONTAINMENT
DISCIPLINE DIR	
PROJ APPROVAL	
DATE	

Entergy Nuclear Northeast

SCALE NONE
DWG NO 9321-F-27203
SHEET
REV 29

IP3.SCAN1



NOTES:

1. ALL SERVICE WATER VALVE NO'S ARE PRECEDED BY "SWN" AS THE SYSTEM DESIGNATION.
2. SWN-55 IS A METERING VALVE WITH 4-1 3/4" HOLES DRILLED IN VALVE DISC ZONE B-3.
3. WORK THIS DWG WITH DWG 9321-F-20333 SH 1
4. SEE DWG. 9321-F-70243 FOR ADDITIONAL INFORMATION.

NOTE:
ALL PIPING ON THIS DWG. IS SEISMIC CLASS I EXCEPT AS NOTED.

THIS DWG. WAS PREVIOUSLY 9321-F-27223, SH 2.

REV	DESCRIPTION	DATE	BY	CHK'D	APP.
42	INCORPORATED DRN-06-01413, ER IP3-06-14923	6/26/06	RM	RL	

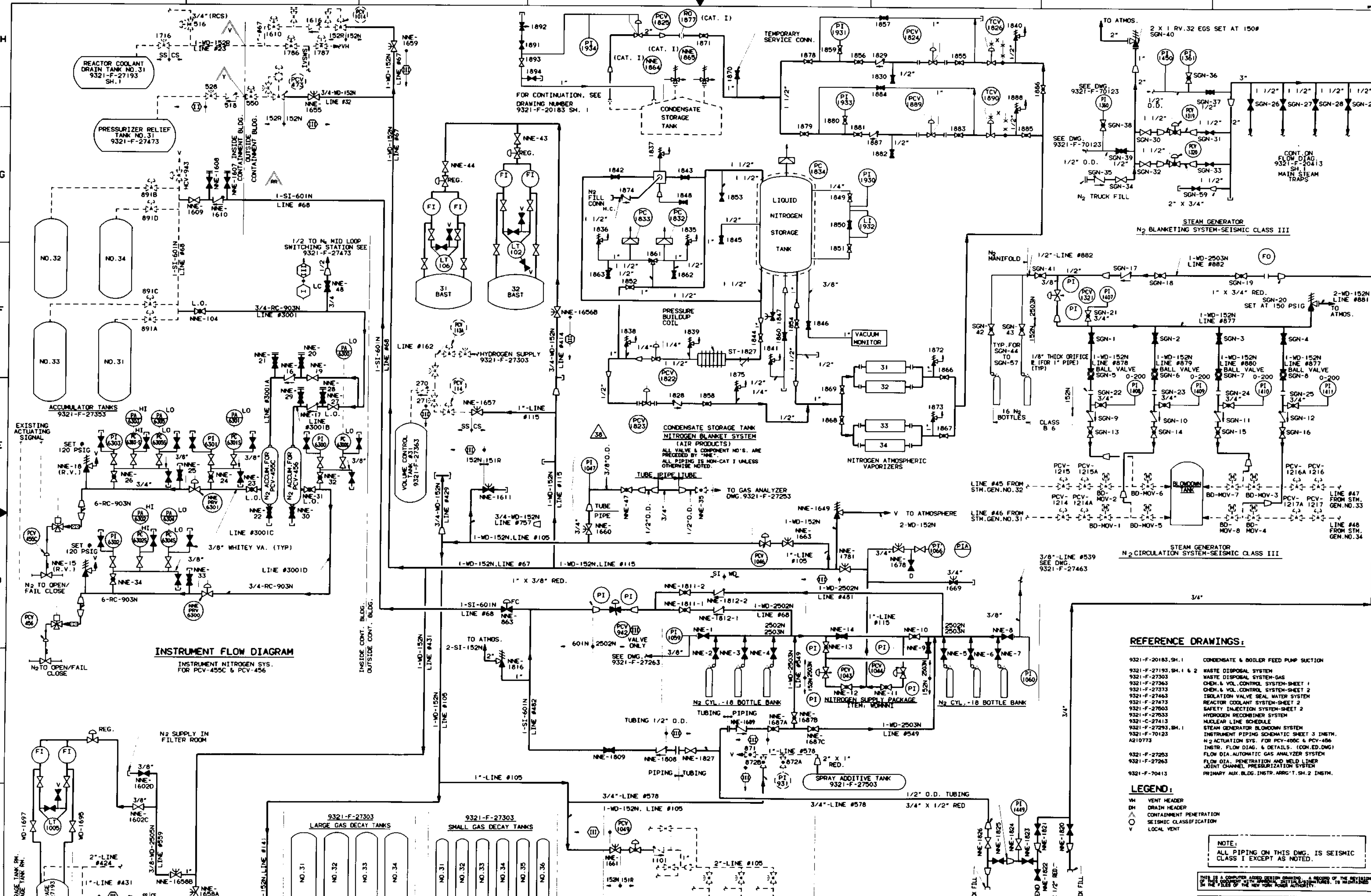
INDIAN POINT NO. 3 NUCLEAR POWER PLANT

FLOW DIAGRAM
SERVICE WATER SYSTEM
NUCLEAR STEAM SUPPLY PLANT

Entergy
Nuclear Northeast

SCALE: NONE
DWG NO 9321-F-27223
REV 42

TYPE A/ISI
FSAR



REFERENCE DRAWINGS:

9321-F-20183,SH.1	CONDENSATE & BOILER FEED PUMP SUCTION
9321-F-27193,SH.1 & 2	WASTE DISPOSAL SYSTEM
9321-F-27303	WASTE DISPOSAL SYSTEM-GAS
9321-F-27363	CHEM.& VOL.CONTROL SYSTEM-SHEET 1
9321-F-27373	CHEM.& VOL.CONTROL SYSTEM-SHEET 2
9321-F-27463	ISOLATION VALVE SEAL WATER SYSTEM
9321-F-27473	REACTOR COOLANT SYSTEM-SHEET 2
9321-F-27503	SAFETY INJECTION SYSTEM-SHEET 2
9321-F-27533	HYDROGEN RECOMBINER SYSTEM
9321-C-27413	NUCLEAR LINE SCHEDULE
9321-F-27293,SH.1	STEAM GENERATOR BLOWDOWN SYSTEM
9321-F-70123	INSTRUMENT PIPING SCHEMATIC SHEET 3 INSTM.
A210773	N_2 ACTUATION SYS. FOR PCV-455C & PCV-456 INSTR. FLOW DIAG. & DETAILS. (CON.ED.DWG)
9321-F-27253	FLOW DIA.AUTOMATIC GAS ANALYZER SYSTEM
9321-F-27263	FLOW DIA. PENETRATION AND WELD LINER JOINT CHANNEL PRESSURIZATION SYSTEM
9321-F-70413	PRIMARY AUX.BLDG.INSTR.ARRG'T.SH.2 INSTM.

LEGEND:

VH	VENT HEADER
DH	DRAIN HEADER
△	CONTAINMENT PENETRATION
○	SEISMIC CLASSIFICATION
V	LOCAL VENT

NOTE:
ALL PIPING ON THIS DWG. IS SEISMIC CLASS I EXCEPT AS NOTED.

THIS IS A COMPUTER AIDED DESIGN DRAWING. A RECORD OF THE REVISIONS OF THIS DOCUMENT WITH APPROVAL INITIALS/SIGNATURES, IS MAINTAINED IN THE FILES OF THE NEW YORK POWER AUTHORITY.

REV NO	DATE	DESCRIPTION	DWN/CHK	DS	DE/DD	PROJ MGR
38	4/2/98	AS BUILT PER DCR 100144516	JF/DA		-	
37	1/20/98	AS BUILT PER DCR 100143874	VR/LM	JM	-	REG
36	8/21/97	AS BUILT PER MOD 97-3-133 N2	IS/RM	JM	-	REG
35	5/1/97	AS BUILT PER DCR 100135467	IS/RM	JM	-	REG
34	10/2/96	AS BUILT PER DCR 100128512	IS/MM	MGM	-	REG

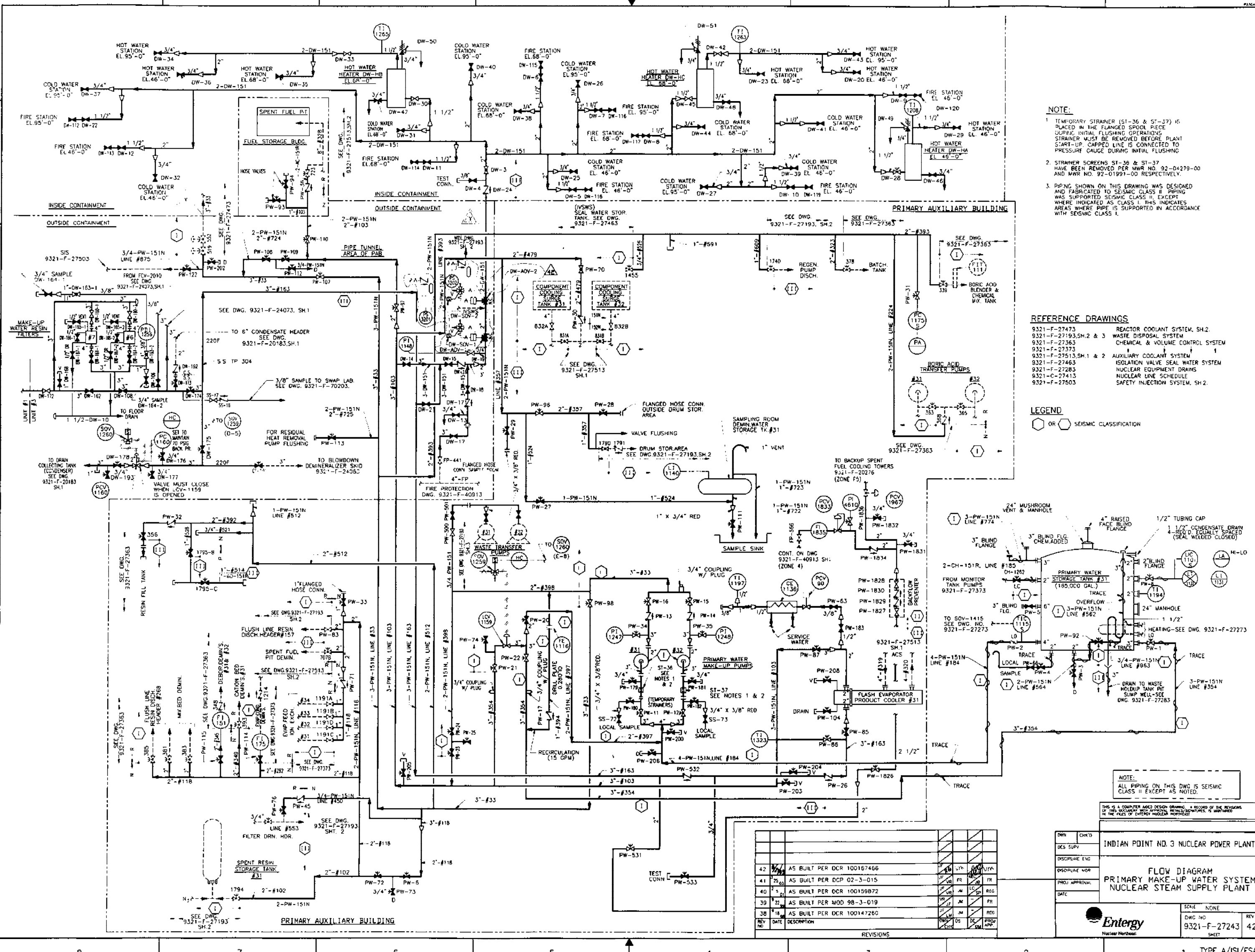
REVISIONS

DWN/CHK'D	INDIAN POINT NO.3 NUCLEAR POWER PLANT
DES SUPV	
DISCIPLINE ENG	FLOW DIAGRAM NITROGEN TO NUCLEAR EQUIPMENT
DISCIPLINE DIR	
PROJ APPROVAL	
DATE	

New York Power Authority

SCALE NONE
DWG NO 9321-F-27233
REV 38
SHEET

TYPE A/ISI/FSAR



NOTE:

1. TEMPORARY STRAINER (ST-36 & ST-37) IS PLACED IN THE FLANGED SPOOL PIECE DURING INITIAL FLUSHING OPERATIONS STRAINER MUST BE REMOVED BEFORE PLANT START-UP. CAPPED LINE IS CONNECTED TO PRESSURE GAUGE DURING INITIAL FLUSHING
2. STRAINER SCREENS ST-36 & ST-37 HAVE BEEN REMOVED PER MWR NO. 92-04279-00 AND MWR NO. 92-01991-00 RESPECTIVELY.
3. PIPING SHOWN ON THIS DRAWING WAS DESIGNED AND FABRICATED TO SEISMIC CLASS II PIPING WAS SUPPORTED SEISMIC CLASS II, EXCEPT WHERE INDICATED AS CLASS I. THIS INDICATES AREAS WHERE PIPE IS SUPPORTED IN ACCORDANCE WITH SEISMIC CLASS I.

REFERENCE DRAWINGS

9321-F-27473	REACTOR COOLANT SYSTEM, SH.2.
9321-F-27193,SH.2 & 3	WASTE DISPOSAL SYSTEM
9321-F-27363	CHEMICAL & VOLUME CONTROL SYSTEM
9321-F-27373	" " " "
9321-F-27513,SH.1 & 2	AUXILIARY COOLANT SYSTEM
9321-F-27463	ISOLATION VALVE SEAL WATER SYSTEM
9321-F-27283	NUCLEAR EQUIPMENT DRAINS
9321-C-27413	NUCLEAR LINE SCHEDULE
9321-F-27503	SAFETY INJECTION SYSTEM, SH.2.

LEGEND

OR SEISMIC CLASSIFICATION

NOTE:
ALL PIPING ON THIS DWG IS SEISMIC CLASS II EXCEPT AS NOTED.

THIS IS A COMPUTER AIDED DESIGN DRAWING. A RECORD OF THE REVISIONS OF THIS DOCUMENT WITH APPROVAL INITIALS/SIGNATURES, IS MAINTAINED IN THE FILES OF ENTERGY NUCLEAR NORTHEAST

REV NO	DATE	DESCRIPTION	DWN / CHK	DS	DE / DM	PROJ APP.
42	[illegible]	AS BUILT PER DCR 100167466	JF	VIK	[illegible]	VIK
41	3/25/03	AS BUILT PER DCP 02-3-015	JF / VR	FR	AEC / JB	FR
40	2/5/01	AS BUILT PER DCR 100159872	JF / VR	JM	LC / SP	REG
39	9/22/[illegible]	AS BUILT PER MOD 98-3-019	VR / JF	JM	-	FR
38	9/18/[illegible]	AS BUILT PER DCR 100147260	JF / LM	JM	-	REG

REVISIONS

INDIAN POINT NO. 3 NUCLEAR POWER PLANT

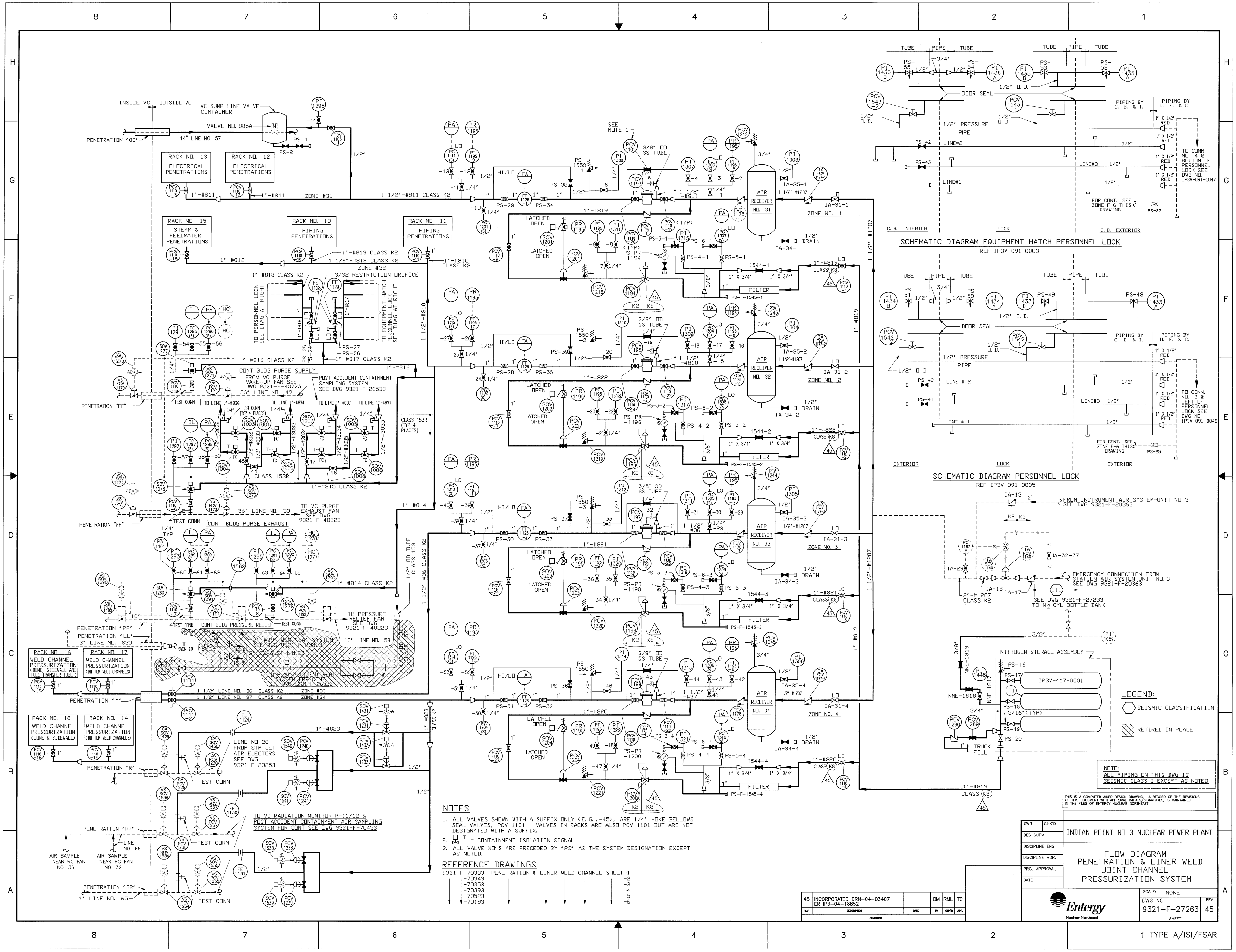
FLOW DIAGRAM
PRIMARY MAKE-UP WATER SYSTEM
NUCLEAR STEAM SUPPLY PLANT

DWN | CHK'D | DES SUPV | DISCIPLINE ENG | DISCIPLINE MGR | PROJ APPROVAL | DATE

Entergy
Nuclear Northeast

SCALE NONE
DWG NO 9321-F-27243 SHEET
REV 42

TYPE A/ISI/FSAR



NOTES:

1. ALL VALVES SHOWN WITH A SUFFIX ONLY (E.G., -45), ARE 1/4" HOKE BELLOWS SEAL VALVES, PCV-1101. VALVES IN RACKS ARE ALSO PCV-1101 BUT ARE NOT DESIGNATED WITH A SUFFIX.
2. ⊡-T = CONTAINMENT ISOLATION SIGNAL
3. ALL VALVE NO'S ARE PRECEDED BY "PS" AS THE SYSTEM DESIGNATION EXCEPT AS NOTED.

REFERENCE DRAWINGS:

9321-F-70333 PENETRATION & LINER WELD CHANNEL-SHEET-1
-70343 -2
-70353 -3
-70393 -4
-70523 -5
-70193 -6

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REV	DESCRIPTION	DATE	BY	CHK'D	APP.
45	INCORPORATED DRN-04-03407 ER IP3-04-18852		DM	RML	TC

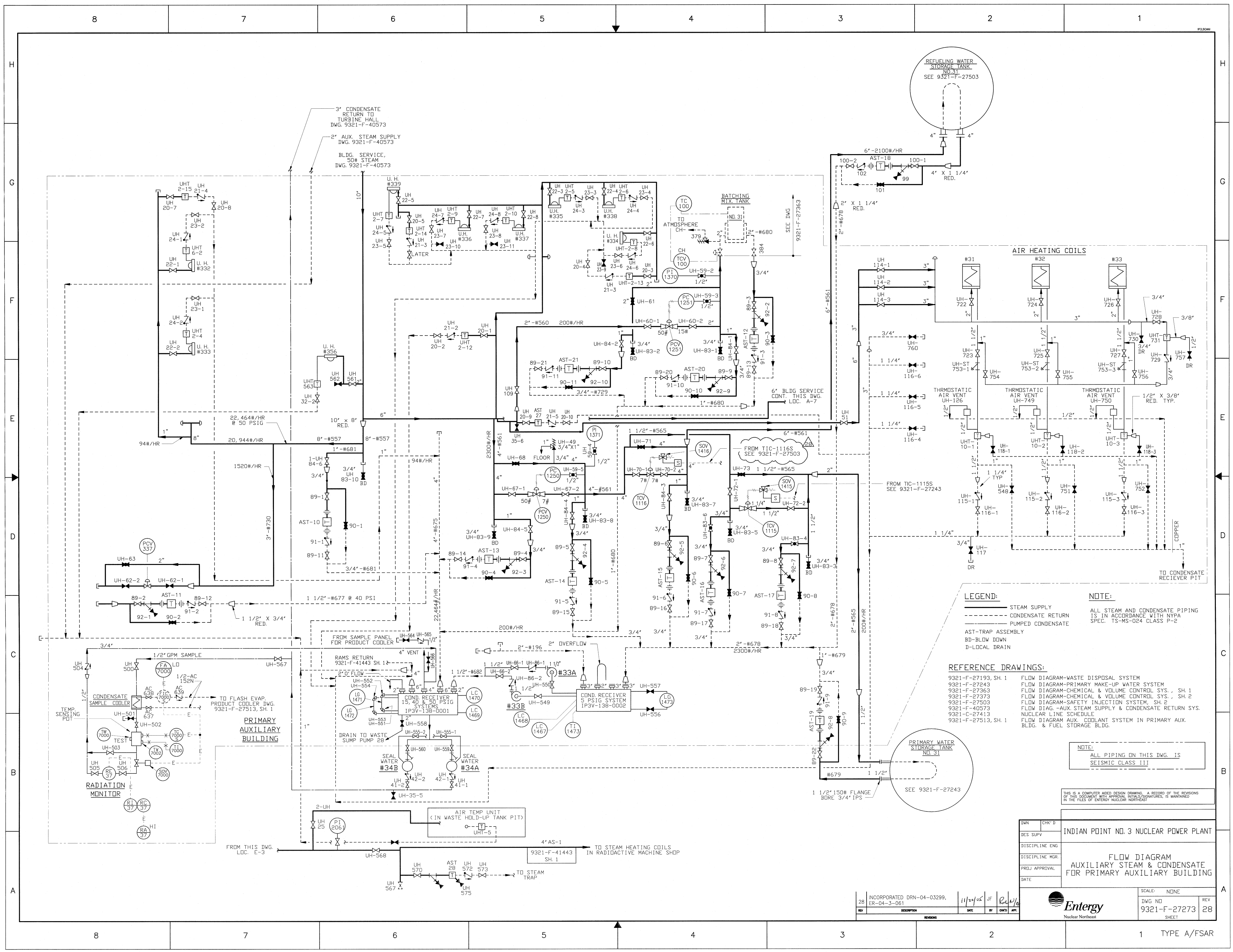
INDIAN POINT NO. 3 NUCLEAR POWER PLANT

FLOW DIAGRAM PENETRATION & LINER WELD JOINT CHANNEL PRESSURIZATION SYSTEM

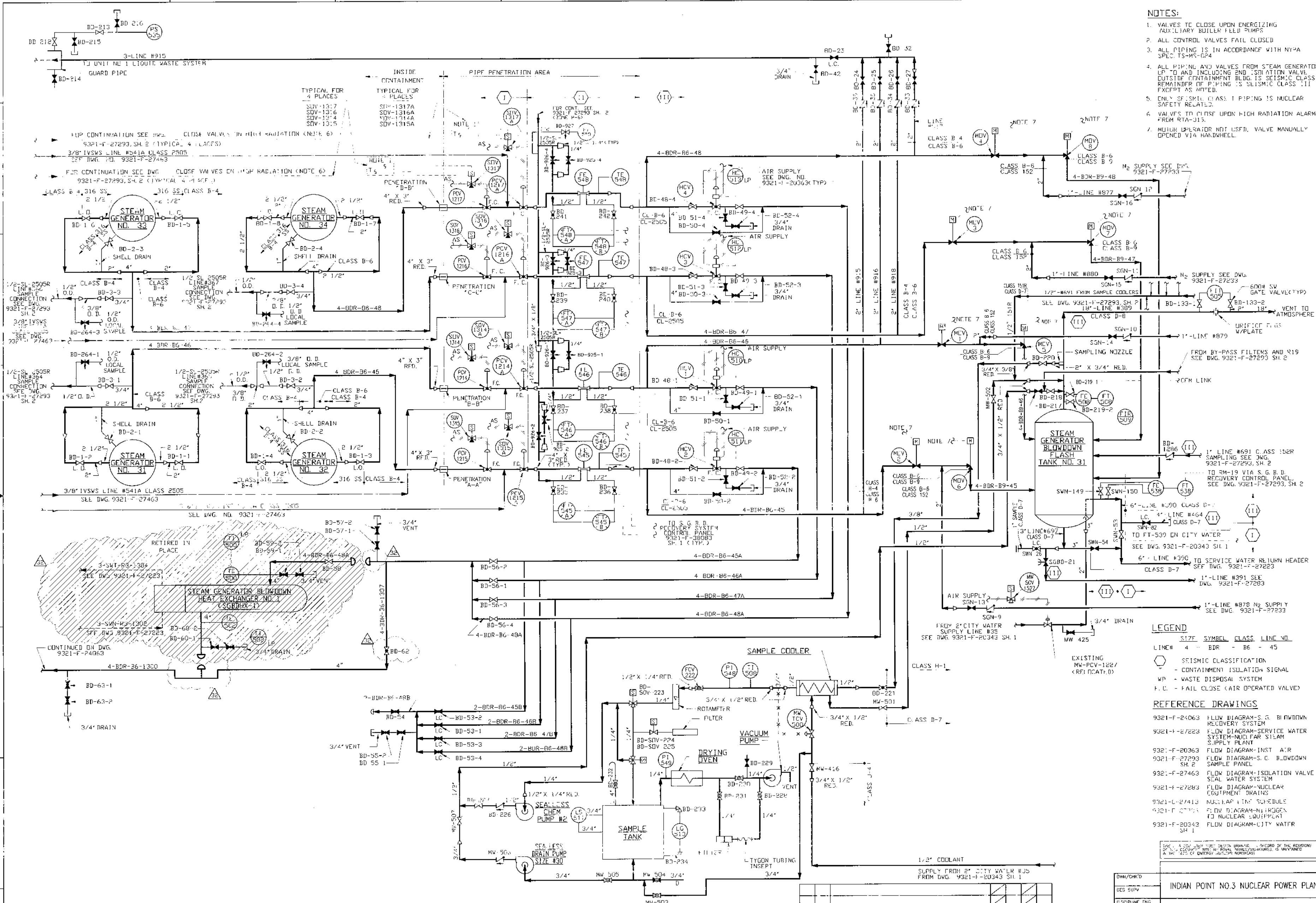
Entergy Nuclear Northeast

SCALE: NONE
DWG NO 9321-F-27263 REV 45

1 TYPE A/ISI/FSAR



REFUELING WATER STORAGE TANK NO.31 SEE 9321-F-27503
3" CONDENSATE RETURN TO TURBINE HALL DWG. 9321-F-40573
2" AUX. STEAM SUPPLY DWG. 9321-F-40573
BLDG. SERVICE, 50# STEAM DWG. 9321-F-40573
BATCHING MIX. TANK
AIR HEATING COILS
6" BLDG SERVICE CONT. THIS DWG. LOC. A-7
FROM TIC-1116S SEE 9321-F-27503
FROM TIC-1115S SEE 9321-F-27243
TO CONDENSATE RECIEVER PIT
PRIMARY AUXILIARY BUILDING
RADIATION MONITOR
CONDENSATE SAMPLE COOLER
TO FLASH EVAP. PRODUCT COOLER DWG. 9321-F-27513, SH. 1
FROM SAMPLE PANEL FOR PRODUCT COOLER
RAMS RETURN 9321-F-41443 SH. 1
COND. RECEIVER 15, 40 & 50 PSIG SYSTEMS IP3V-138-0001
COND. RECEIVER 5 PSIG SYSTEM IP3V-138-0002
DRAIN TO WASTE SUMP PUMP 28
AIR TEMP UNIT (IN WASTE HOLD-UP TANK PIT)
FROM THIS DWG. LOC. E-3
TO STEAM HEATING COILS IN RADIOACTIVE MACHINE SHOP
9321-F-41443 SH. 1
TO STEAM TRAP
PRIMARY WATER STORAGE TANK NO. 31 SEE 9321-F-27243
1 1/2" 150# FLANGE BORE 3/4" IPS
LEGEND:
STEAM SUPPLY
CONDENSATE RETURN
PUMPED CONDENSATE
AST-TRAP ASSEMBLY
BD-BLOW DOWN
D-LOCAL DRAIN
NOTE:
ALL STEAM AND CONDENSATE PIPING IS IN ACCORDANCE WITH NYPA SPEC. TS-MS-024 CLASS P-2
REFERENCE DRAWINGS:
9321-F-27193, SH. 1 FLOW DIAGRAM-WASTE DISPOSAL SYSTEM
9321-F-27243 FLOW DIAGRAM-PRIMARY MAKE-UP WATER SYSTEM
9321-F-27363 FLOW DIAGRAM-CHEMICAL & VOLUME CONTROL SYS., SH. 1
9321-F-27373 FLOW DIAGRAM-CHEMICAL & VOLUME CONTROL SYS., SH. 2
9321-F-27503 FLOW DIAGRAM-SAFETY INJECTION SYSTEM, SH. 2
9321-F-40573 FLOW DIAG.-AUX. STEAM SUPPLY & CONDENSATE RETURN SYS.
9321-C-27413 NUCLEAR LINE SCHEDULE
9321-F-27513, SH. 1 FLOW DIAGRAM AUX. COOLANT SYSTEM IN PRIMARY AUX. BLDG. & FUEL STORAGE BLDG.
NOTE: ALL PIPING ON THIS DWG. IS SEISMIC CLASS III
THIS IS A COMPUTER AIDED DESIGN DRAWING. A RECORD OF THE REVISIONS OF THIS DOCUMENT WITH APPROVAL INITIALS/SIGNATURES, IS MAINTAINED IN THE FILES OF ENTERGY NUCLEAR NORTHEAST
28 INCORPORATED DRN-04-03299, ER-04-3-061 11/29/05 JF
INDIAN POINT NO. 3 NUCLEAR POWER PLANT
FLOW DIAGRAM AUXILIARY STEAM & CONDENSATE FOR PRIMARY AUXILIARY BUILDING
Entergy Nuclear Northeast
SCALE: NONE
DWG NO 9321-F-27273
REV 28
TYPE A/FSAR



NOTES:

1. VALVES TO CLOSE UPON ENERGIZING AUXILIARY BOILER FEED PUMPS
2. ALL CONTROL VALVES FAIL CLOSED
3. ALL PIPING IS IN ACCORDANCE WITH NYPA SPEC. TS-MS-024
4. ALL PIPING AND VALVES FROM STEAM GENERATORS UP TO AND INCLUDING 2ND ISOLATION VALVE OUTSIDE CONTAINMENT BLDG. IS SEISMIC CLASS I REMAINDER OF PIPING IS SEISMIC CLASS III EXCEPT AS NOTED.
5. ONLY SEISMIC CLASS I PIPING IS NUCLEAR SAFETY RELATED.
6. VALVES TO CLOSE UPON HIGH RADIATION ALARM FROM RIA-015.
7. MOTOR OPERATOR NOT USED. VALVE MANUALLY OPENED VIA HANDWHEEL.

LEGEND

	SIZE	SYMBOL	CLASS	LINE NO.
LINE#	4 -	BDR -	B6 -	45

SEISMIC CLASSIFICATION
I - CONTAINMENT ISOLATION SIGNAL
WD - WASTE DISPOSAL SYSTEM
F.C. - FAIL CLOSE (AIR OPERATED VALVE)

REFERENCE DRAWINGS

9321-F-24063	FLOW DIAGRAM-S.G. BLOWDOWN RECOVERY SYSTEM
9321-F-27223	FLOW DIAGRAM-SERVICE WATER SYSTEM-NUCLEAR STEAM SUPPLY PLANT
9321-F-20363	FLOW DIAGRAM-INST AIR
9321-F-27293 SH. 2	FLOW DIAGRAM-S.G. BLOWDOWN SAMPLE PANEL
9321-F-27463	FLOW DIAGRAM-ISOLATION VALVE SEAL WATER SYSTEM
9321-F-27283	FLOW DIAGRAM-NUCLEAR EQUIPMENT DRAINS
9321-C-27413	NUCLEAR LINE SCHEDULE
9321-F-27233	FLOW DIAGRAM-NITROGEN TO NUCLEAR EQUIPMENT
9321-F-20343 SH 1	FLOW DIAGRAM-CITY WATER

THIS IS A COMPUTER MOST DESIGN DRAWING. A RECORD OF THE REVISIONS OF THIS DOCUMENT WITH APPROVAL INITIALS/SIGNATURES, IS MAINTAINED IN THE FILES OF ENTERGY NUCLEAR NORTHEAST

REVISIONS

REV NO	DATE	DESCRIPTION	DWN / CHK	DS	DE / DC	PROJ MGR
32	4/29/04	INCORP. CLN 9321-F-27223 SH.1 -30C DCP 02-3-018	JF / GH	TC	-	TC
31	11/15/07	AS BUILT PER DCP 98-3 074	JF / VR	JM	NS / TB	FR
30	6/3/02	AS BUILT PER DCR 100164198	JF / VR	JM	RSC	REG
29	5/19/00	AS BUILT PER DCR 100157087	JF / JM	JM	-	REG
28	4/12/00	AS BUILT PER MMP 98-3-074	VR / JM	JM	-	REG

DWN/CHK'D
DES SUPV
DISCIPLINE ENG
DISCIPLINE DIR
PROJ APPROVAL
DATE

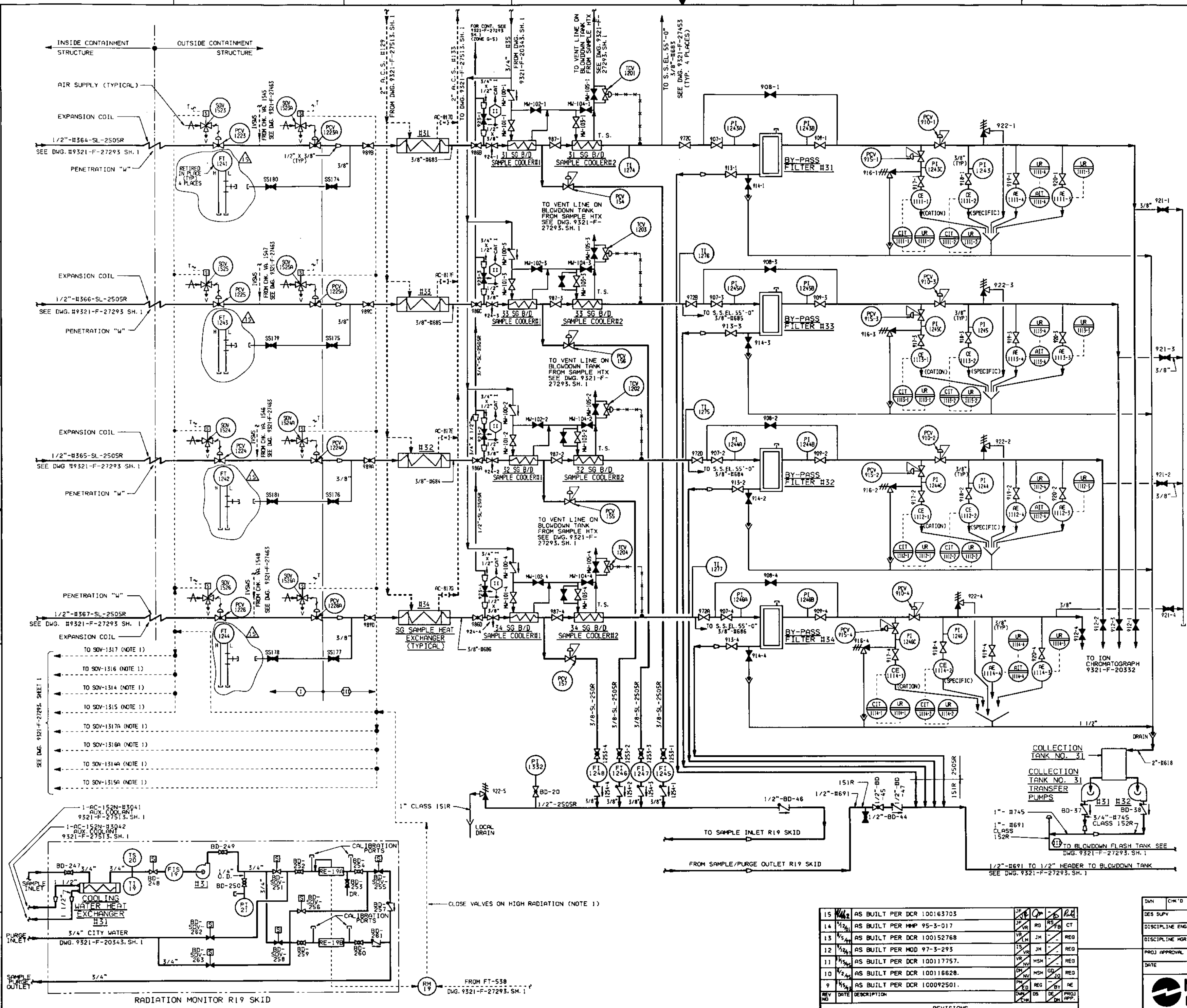
INDIAN POINT NO.3 NUCLEAR POWER PLANT

FLOW DIAGRAM STEAM GENERATOR BLOWDOWN SYSTEM

Entergy
Nuclear Northeast

SCALE NONE
DWG NO 9321-F-27293 REV 32
SHEET 1

TYPE A/ISI/FSAR



NOTES:

1. VALVES TO CLOSE UPON HIGH RADIATION ALARM.
2. ALL VALVE NO'S. ARE PRECEDED BY A "BD" SYSTEM DESIGNATION UNLESS OTHERWISE NOTED.

REFERENCE DRAWINGS:

9321-F-27293, SH.1-STEAM GENERATOR BLOWDOWN SYSTEM-FLOW DIAGRAM.

9321-F-27453-SAMPLING SYSTEM-FLOW DIAGRAM.

9321-F-27463-ISOLATION VALVE SEAL WATER SYSTEM-FLOW DIAGRAM.

9321-F-27513, SH.1-AUX COOLANT SYSTEM IN PAB & FSB-FLOW DIAGRAM.

9321-F-27513, SH.2-AUX COOLANT SYSTEM IN PAB & FSB-FLOW DIAGRAM.

9321-F-20332-ION CHROMATOGRAPHY ANALYZER

LEGEND:

IVSWS	ISOLATION VALVE SEAL WATER SYSTEM
L.O.	LOCKED OPEN VALVE
L.C.	LOCKED CLOSED VALVE
T	CONTAINMENT ISOLATION SIGNAL
F.C.	FAIL CLOSE (AIR OPERATED VALVE)
⬡	SEISMIC CLASSIFICATION
⊖	LOCAL CONTROL PANEL

THIS IS A COMPUTER AIDED DESIGN DRAWING. A RECORD OF THE REVISIONS OF THIS DOCUMENT WITH APPROVAL INITIALS/SIGNATURES IS MAINTAINED IN THE FILES OF THE NEW YORK POWER AUTHORITY.

REV NO	DATE	DESCRIPTION	DWN/CHK	DS	DE/DM	PROJ APP.
15	[illegible]	AS BUILT PER DCR 100163703	JF/JF	[illegible]	-/[illegible]	[illegible]
14	4/12/01	AS BUILT PER MMP 95-3-017	JF/VR	RG	RS/FB	CT
13	6/5/99	AS BUILT PER DCR 100152768	VR/LN	JH	-/-	REG
12	9/12/97	AS BUILT PER MOD 97-3-293	IS/VR	JH	-/-	REG
11	11/15/95	AS BUILT PER DCR 100117757.	VR/NV	MSM	-/-	REG
10	6/2/95	AS BUILT PER DCR 100116628.	DM/NV	MSM	GD/JG	REG
9	11/15/93	AS BUILT PER DCR 100092501.	PM/ED	REG	-/BY	AE

REVISIONS

DWN / CHK'D / DES SUPV / DISCIPLINE ENG / DISCIPLINE MGR. / PROJ APPROVAL / DATE

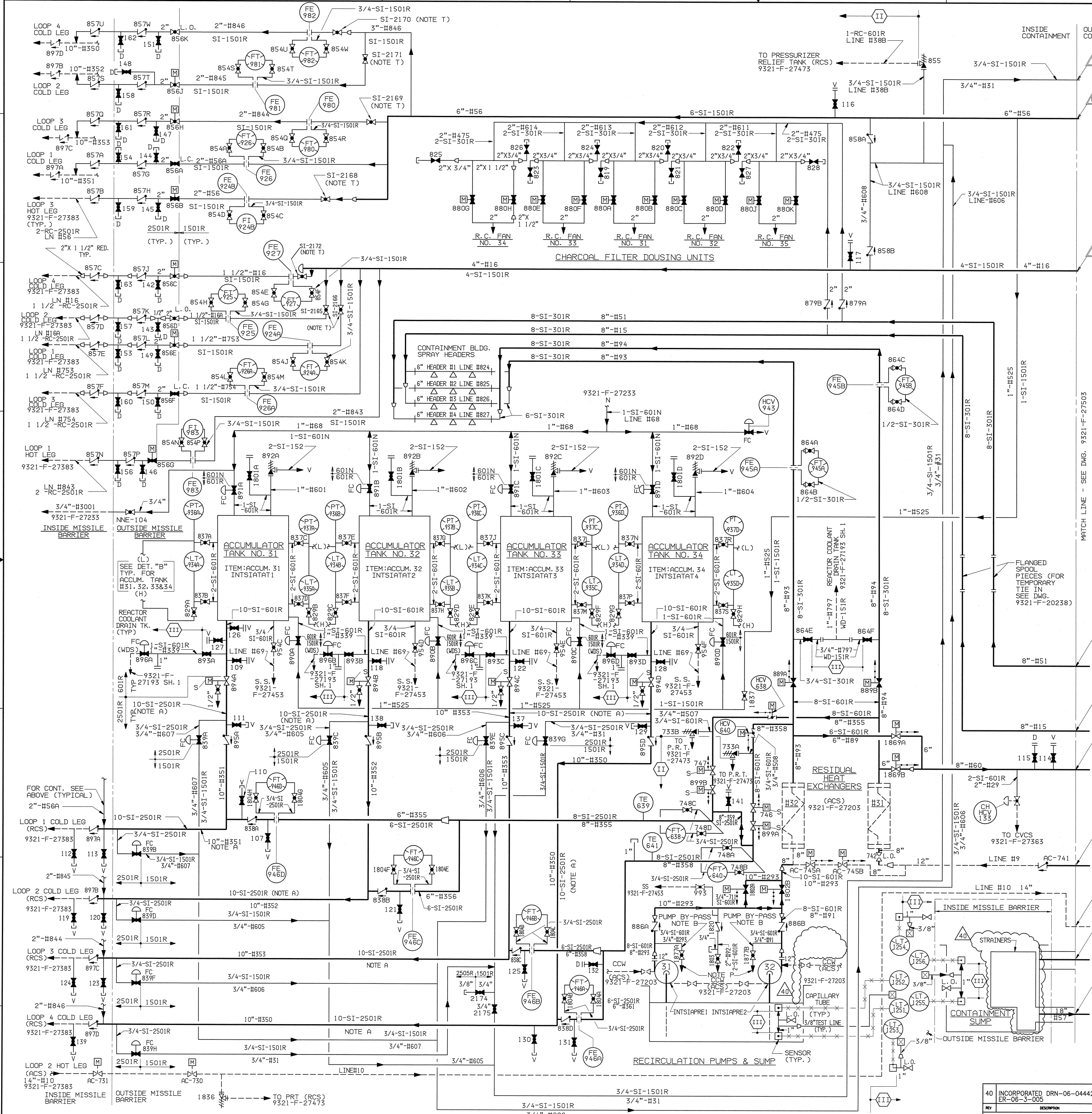
INDIAN POINT NO. 3 NUCLEAR POWER PLANT

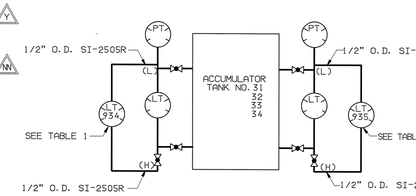
FLOW DIAGRAM
STEAM GENERATOR
BLOWDOWN
SAMPLE PANEL

New York Power Authority

SCALE: NONE
DWG NO 9321-F-27293 SHEET 2
REV 15

TYPE A/ISI/FSAR





DETAIL "B"

TYP. FOR 4 ACCUM. TANKS

(FOR PHYSICAL HOOK-UP SEE DWG. A210198)

TABLE 1

TANK NO.	LT NO.
31	LT934A
	LT935A
32	LT934B
	LT935B
33	LT934C
	LT935C
34	LT934D
	LT935D

REFERENCES:

1. PROCESS FLOW DIAGRAM Ⓦ DWG. 541-F-424
2. DEFINITION OF SYMBOLS
   Ⓦ E. SPEC. G-675176 REV. 3
   Ⓦ E. SPEC. G-675164 REV. 0
3. INSTALLATION OF INSTRUMENTATION
   Ⓦ PROC. SPEC. CAP. 294367 REV. 1
4. MATERIAL SPEC. PIPE AND FITTINGS
   NYPA SPEC. TS-MS-024
   Ⓦ E. SPEC. G-676398 REV. 3

LEGEND:

S	TEMPORARY STRAINER
	ELECTRICAL HEAT TRACING
LLT	TRIP ON LOW LEVEL IN BORON INJECTION TANK
	CONTAINMENT PENETRATION
	SEISMIC CLASSIFICATION
	QUICK DISCONNECT FOR TEST
L.O.	LOCKED OPEN
L.C.	LOCKED CLOSED
F.O.	FAIL OPEN
F.C.	FAIL CLOSED
V	LOCAL VENT
D	LOCAL DRAIN
	SENSOR W/ CAPILLARY

NOTES:

A. PIPING IS SCHEDULE 140
B. LOCATED IN HIGH SIDE OF PIPE IN LOCAL HIGH POINT
C.
D. ORIFICE IS SUPPLIED WITH PUMP PACKAGE
E. MOUNTED WITH STEM HORIZONTAL
F.
G.
H. PROVISION FOR A SECOND SUMP LINE
J.
K. BALANCE PISTON FLOW LINE FOR S.I. PUMP 32 IS INSIDE VALVES 887A AND 887B
L. LOCATED AT LOW POINT OF PIPE WORK
M. VALVES INSTALLED FOR VERTICAL DISCHARGE
N. FLUSH LINE IS INSTALLED AS CLOSE TO VALVE INLET AS POSSIBLE
P. LINES ARE DIRECTED TO THIRD SECTION OF THE SUMP (SECTION CONTAINING RECIRC. PUMPS)
R. IVSWS OR LEAKOFF CONNECTIONS ON THE FOLLOWING VALVES 850A, 850B, 850C, 885A, 866A & B ARE PLUGGED OR CAPPED
S. ALL VALVE NO'S. ARE PRECEDED BY "SI" AS THE SYSTEM DESIGNATION EXCEPT AS NOTED
T. NORMAL POSITION OF VALVES SI-2165, -2166, -2168, -2169, -2170, -2171, -2172 IS LOCK-THROTTLED, AS DETERMINED BY A SYSTEM FLOW BALANCE. VALVE MIN. STOP POSITION SET AT 0.25" TO ALLOW VALVE 855 TO PROTECT DOWNSTREAM PIPING.

REFERENCE DRAWINGS:

9321-F-27503	SIS	SAFETY INJECTION SYSTEM SHEET 2
27193	WDS	WASTE DISPOSAL SYSTEM SHEET 1 - CONTAINMENT
27203	ACS	AUXILIARY COOLANT SYSTEM - INSIDE CONTAINMENT
27513	ACS	AUXILIARY COOLANT SYSTEM IN PAB & FSB - SHEET 2
27363	CVCS	CHEMICAL & VOLUME CONTROL SYSTEM SHEET 1
27373	CVCS	CHEMICAL & VOLUME CONTROL SYSTEM SHEET 2
27383	RCS	REACTOR COOLANT SYSTEM SHEET 1
27473	RCS	REACTOR COOLANT SYSTEM SHEET 2
27233	N	NITROGEN TO NUCLEAR EQUIPMENT
27243	PW	PRIMARY MAKE-UP WATER SYSTEM
27453	SS	SAMPLING SYSTEM
27463	IVSWS	ISOLATION VALVE SEAL WATER SYSTEM
27273	AS	AUXILIARY STEAM & COND FOR PRIMARY AUX. BLDG.
9321-C-27413		NUCLEAR LINE SCHEDULE

WORK THIS DWG. WITH DWG. 9321-F-27503

NOTE:
ALL PIPING ON THIS DWG. IS SEISMIC CLASS I EXCEPT AS NOTED

REDRAWN FROM Ⓦ DWG. 685J427 & 110E364

REV	DESCRIPTION	DATE	BY	CHK'D	APP.
40	INCORPORATED DRN-06-04442 ER-06-3-005	3/28/07	GH	[illegible]	[illegible]

REVISIONS

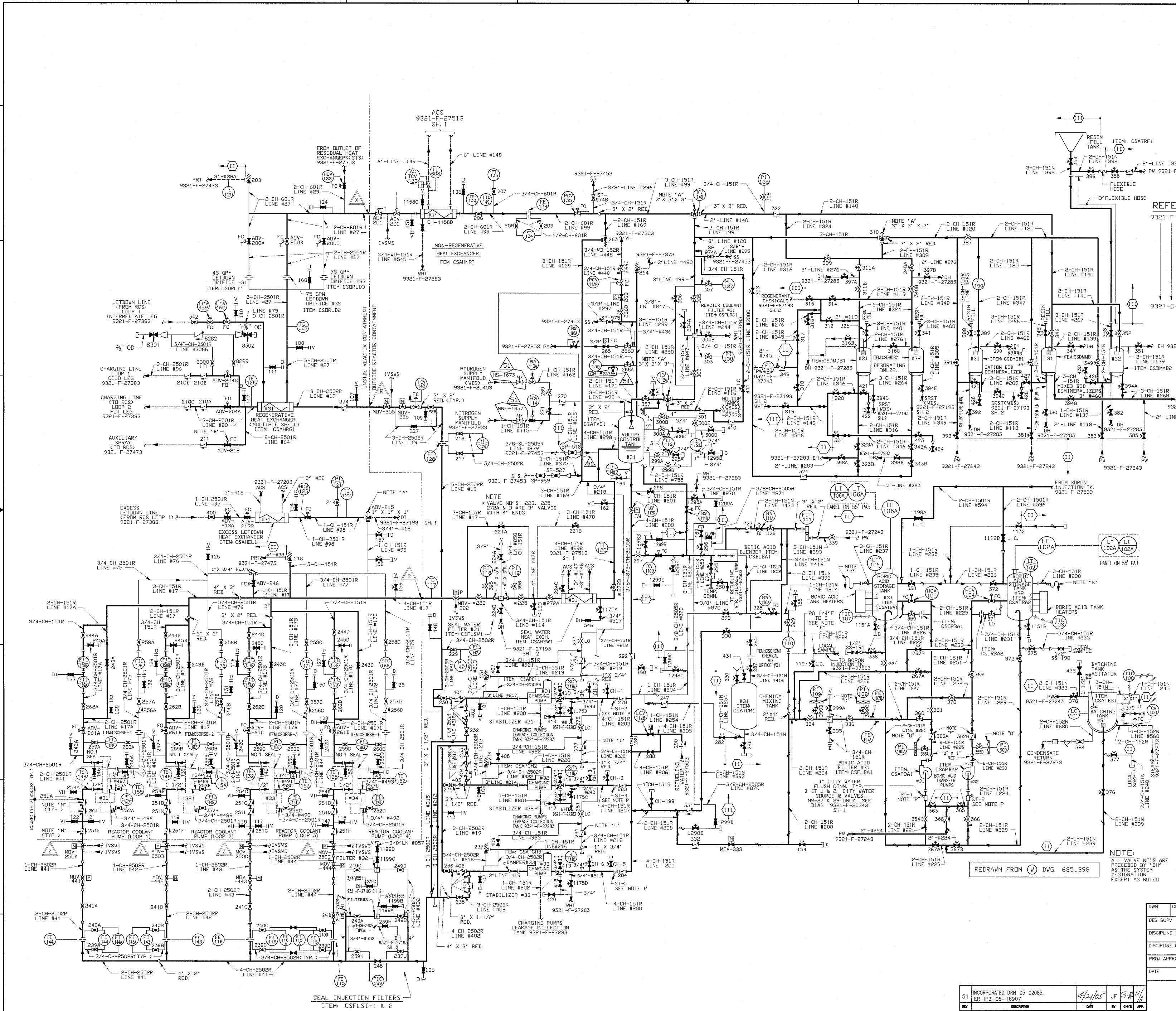
DWN/CHK'D	INDIAN POINT NO. 3 NUCLEAR POWER PLANT
DES SUPV	
DISCIPLINE ENG	FLOW DIAGRAM SAFETY INJECTION SYSTEM SHEET NO. 1
DISCIPLINE DIR	
PROJ APPROVAL	
DATE	

Entergy
Nuclear Northeast

SCALE : NONE

DWG NO 9321-F-27353 SHEET — REV 40

TYPE A/ISI/FSAR



REFERENCES:

1. PROCESS FLOW DIAGRAM (W) DWG.
2. DEFINITION OF SYMBOLS (W) E SPEC. 6675176 REV. 2 AND (W) E SPEC. 6675164 REV. 0
3. INSTALLATION OF INSTRUMENTATION (W) PROC. SPEC. CAP 294367 REV. 1
4. MATERIAL SPEC. PIPE AND FITTINGS (W) E. SPEC. G569866 REV. 2 AND (W) E. SPEC G676398 REV. 0

REFERENCE DRAWINGS:

9321-F-27193 WDS -WASTE DISPOSAL SYSTEM (SH. 1 & SH. 2)
27303 WDS -WASTE DISPOSAL SYSTEM - GAS
27383 RCS -REACTOR COOLANT SYSTEM
27473 RCS -REACTOR COOLANT SYSTEM
27353 SIS -SAFETY INJECTION SYSTEM
27503 SIS -SAFETY INJECTION SYSTEM
27203 ACS -AUXILIARY COOLANT SYSTEM
27513 ACS -AUXILIARY COOLANT SYSTEM (SHT. 1)
27373 CVCS -CHEMICAL & VOLUME CONTROL SYSTEM
27233 N -NITROGEN TO NUCLEAR EQUIP.
27243 PW -PRIMARY MAKE-UP WATER SYSTEM.
27283 DH -NUCLEAR EQUIPMENT DRAINS
27453 SS -SAMPLING SYSTEM
27273 AS -AUX. STEAM & CONDENSATE FOR P. A. BLDG.
27253 GA -AUTOMATIC GAS ANALYZER SYSTEM
9321-C-27413 -NUCLEAR LINE SCHEDULE

LEGEND:

V -LOCAL VENT
PW -PRIMARY MAKE-UP WATER
VH -VENT HEADER (WDS)
DH -DRAIN HEADER (WDS)
T -CONTAINMENT ISOLATION SIGNAL
D -LOCAL DRAIN
DT -REACTOR COOLANT DRAIN TANK (WDS)
HSC -HAND SPEED CONTROL
GA -GAS ANALYZER (WDS)
FAI -FAIL AS IS
FC -FAIL CLOSED
FO -FAIL OPEN
LO -LOCKED OPEN
LC -LOCKED CLOSED
ATM -ATMOSPHERE
IVSWS -ISOLATION VALVE SEAL WATER SYSTEM
PRT -PRESSURIZER RELIEF TANK
WHT -WASTE HOLDUP TANK
ST -TEMPORARY STRAINER
SRST -SPENT RESIN STORAGE TANK
HUT -HOLDUP TANK
WGCP -WASTE GAS COMPRESSOR PACKAGE (WDS)
△ -CONTAINMENT PENETRATIONS
⬡ -SEISMIC CLASSIFICATION

NOTES:

A. VALVE FAILS WITH FLOW TO VOLUME CONTROL TANK
B. VALVE FUNCTIONS AS ISOLATION VALVE ONLY. OVER PRESSURE PROTECTION FOR THE REGENERATIVE HEAT EXCHANGER PROVIDED BY CH-8282.
C. SPECIAL SPRING LOADED CHECK VALVE
D. DIAPHRAGM SEAL
E. ELECTROMAGNETIC-METER LOCATED IN VERT. PIPE RUN
F. .06" DIA. HOLE IN CLAPPER OR CUP
G. ADDITIONAL VENTS & DRAINS MAY BE INSTALLED SEE PIPING LAYOUT.
H. GLOBE VALVES ARE NORMALLY INSTALLED WITH FLOW UNDER THE SEAT. EXCEPTIONS ARE VALVES NO. 230, 232, 233, 235, 236, 238, 240A THRU D INCL, AND 333
J.
K. LOOP SEAL EXTENDS 12" BELOW AND ABOVE OVERFLOW NOZZLE
M. CHECK VALVES 251E, F, G, ARE LOCATED NEAR CONTAINMENT WALL
N. CHECK VALVES 251A, B, C, ARE LOCATED NEAR REACTOR COOLANT PUMPS
P. TEMPORARY (START-UP) STRAINER ST-1, ST-2, ST-3, ST-4 & ST-5 WERE INSTALLED FOR INITIAL FLUSH. STRAINER BASKETS HAVE BEEN REMOVED UNDER MWR'S 93-10097, 93-10098, 93-10101, 93-10102, & 93-10103.

NOTE:
ALL VALVE NO'S ARE PRECEDED BY "CH" AS THE SYSTEM DESIGNATION EXCEPT AS NOTED

REDRAWN FROM (W) DWG. 685J398

NOTE:
ALL PIPING ON THIS DRAWING IS SEISMIC CLASS I EXCEPT AS NOTED

THIS IS A COMPUTER AIDED DESIGN DRAWING. A RECORD OF THE REVISIONS OF THIS DOCUMENT WITH APPROVAL INITIALS/SIGNATURES, IS MAINTAINED IN THE FILES OF ENTERGY NUCLEAR NORTHEAST

REV	DESCRIPTION	DATE	BY	CHK'D	APP.
51	INCORPORATED DRN-05-02085, ER-IP3-05-16907	4/21/05	JF	GH	N/A

REVISIONS

DWN | CHK'D
DES SUPV
DISCIPLINE ENG
DISCIPLINE MGR.
PROJ APPROVAL
DATE

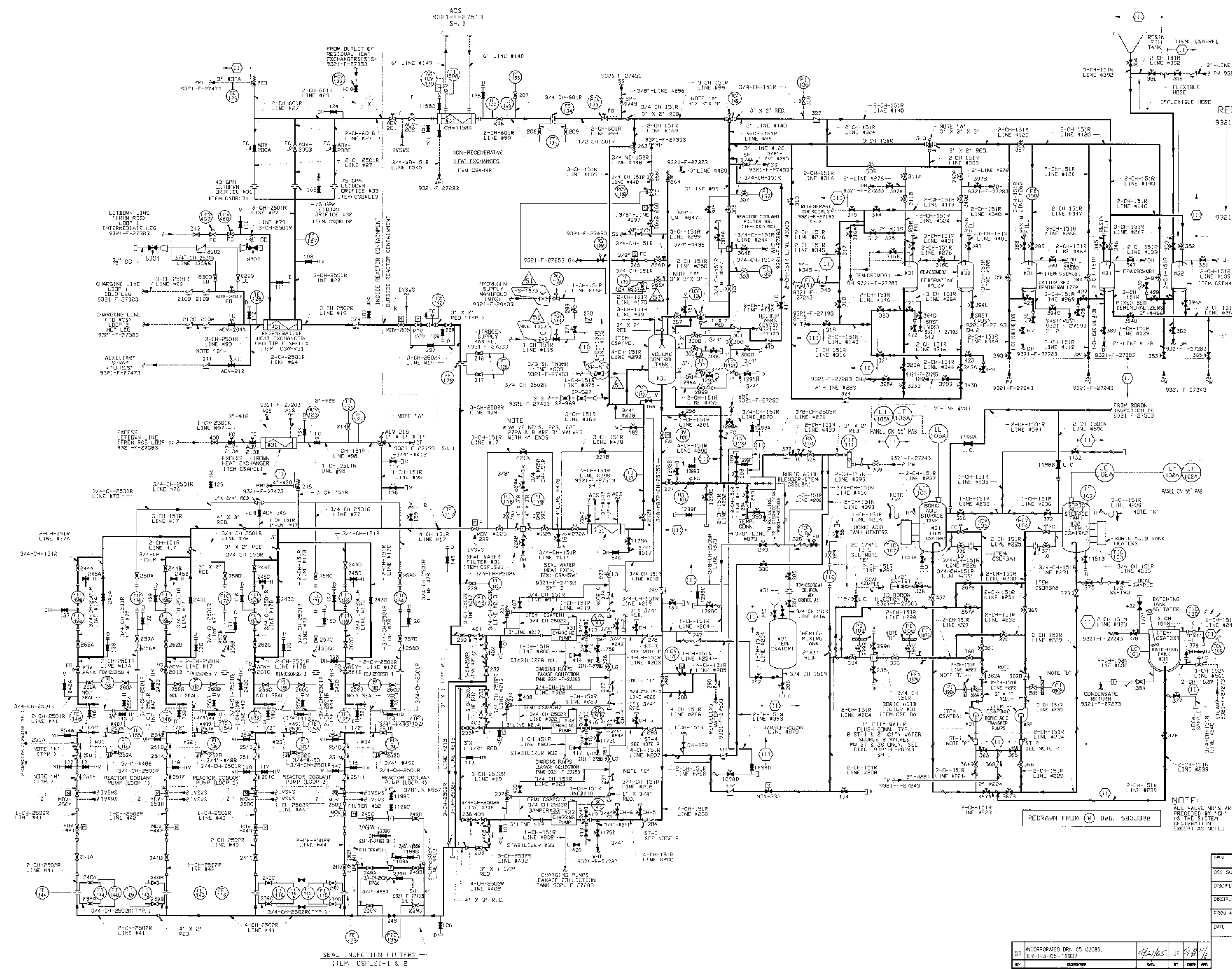
INDIAN POINT NO.3 NUCLEAR POWER PLANT

FLOW DIAGRAM
CHEMICAL & VOLUME CONTROL SYSTEM
SHEET NO.1

Entergy
Nuclear Northeast

SCALE: NONE
DWG NO 9321-F-27363
REV 51
SHEET

TYPE A/ISI/FSAR



REDRAWN FROM (W) DWG. 685J398

NOTE:
ALL VALVE NO'S ARE PRECEDED BY 'CH' AS THE SYSTEM DESIGNATION EXCEPT AS NOTED

REFERENCES:

1. PROCESS FLOW DIAGRAM (W) DWG.
2. DEFINITION OF SYMBOLS (W) F SPEC. 6675176 REV. 2 AND (W) E SPEC. 6675164 REV. 0
3. INSTALLATION OF INSTRUMENTATION (W) PROC. SPEC. CAP 294367 REV. 1
4. MATERIAL SPEC. PIPE AND FITTINGS (W) E. SPEC. G569866 REV. 2 AND (W) E. SPEC G676398 REV. 0

REFERENCE DRAWINGS:

9321-F-27193 WDS -WASTE DISPOSAL SYSTEM (SH. 1 & SH. 2)
27303 WDS -WASTE DISPOSAL SYSTEM GAS
27383 RCS -REACTOR COOLANT SYSTEM
27473 RCS -REACTOR COOLANT SYSTEM
27353 SIS -SAFETY INJECTION SYSTEM
27503 SIS -SAFETY INJECTION SYSTEM
27203 ACS -AUXILIARY COOLANT SYSTEM
27513 ACS -AUXILIARY COOLANT SYSTEM (SHT. 1)
27373 CVCS -CHEMICAL & VOLUME CONTROL SYSTEM
27233 N -NITROGEN TO NUCLEAR EQUIP.
27243 PW -PRIMARY MAKE-UP WATER SYSTEM
27283 DH -NUCLEAR EQUIPMENT DRAINS
27453 SS -SAMPLING SYSTEM
27273 AS -AUX. STEAM & CONDENSATE FOR P. A. BLDG.
27253 GA -AUTOMATIC GAS ANALYZER SYSTEM
9321 C 27413 NUCLEAR LINE SCHEDULE

LEGEND:

V -LOCAL VENT
PW -PRIMARY MAKE-UP WATER
VH -VENT HEADER (WDS)
DH -DRAIN HEADER (WDS)
T -CONTAINMENT ISOLATION SIGNAL
D LOCAL DRAIN
DT -REACTOR COOLANT DRAIN TANK (WDS)
HSC -HAND SPEED CONTROL
GA -GAS ANALYZER (WDS)
FAI -FAIL AS IS
FC -FAIL CLOSED
FO -FAIL OPEN
LO -LOCKED OPEN
LC -LOCKED CLOSED
ATM -ATMOSPHERE
IVSWS -ISOLATION VALVE SEAL WATER SYSTEM
PRT -PRESSURIZER RELIEF TANK
WHT -WASTE HOLDUP TANK
ST -TEMPORARY STRAINER
SRST -SPENT RESIN STORAGE TANK
HUT -HOLDUP TANK
WGCP -WASTE GAS COMPRESSOR PACKAGE (WDS)
-CONTAINMENT PENETRATIONS
-SEISMIC CLASSIFICATION

NOTES:

A. VALVE FAILS WITH FLOW TO VOLUME CONTROL TANK
B. VALVE FUNCTIONS AS ISOLATION VALVE ONLY. OVER PRESSURE PROTECTION FOR THE REGENERATIVE HEAT EXCHANGER PROVIDED BY CH-8282.
C. SPECIAL SPRING LOADED CHECK VALVE
D. DIAPHRAGM SEAL
E. ELECTROMAGNETIC-METER LOCATED IN VERT. PIPE RUN
F. .06" DIA. HOLE IN CLAPPER OR CUP
G. ADDITIONAL VENTS & DRAINS MAY BE INSTALLED SEE PIPING LAYOUT.
H. GLOBE VALVES ARE NORMALLY INSTALLED WITH FLOW UNDER THE SEAT. EXCEPTIONS ARE VALVES NO. 230, 232, 233, 235, 236, 238, 240A THRU I INCL, AND 333
J.
K. LOOP SEAL EXTENDS 12" BELOW AND ABOVE OVERFLOW NOZZLE
M. CHECK VALVES 251E, F, G, ARE LOCATED NEAR CONTAINMENT WALL
N. CHECK VALVES 251A, B, C, ARE LOCATED NEAR REACTOR COOLANT PUMPS
P. TEMPORARY (START UP) STRAINER ST-1, ST-2, ST-3, ST-4 & ST-5 WERE INSTALLED FOR INITIAL FLUSH. STRAINER BASKETS HAVE BEEN REMOVED UNDER MWR'S 93 10097, 93-10098, 93-10101, 93-10102, & 93-10103.

NOTE:
ALL PIPING ON THIS DRAWING IS SEISMIC CLASS I EXCEPT AS NOTED

THIS IS A COMPUTER AIDED DESIGN DRAWING. A RECORD OF THE REVISIONS OF THIS DOCUMENT WITH APPROVAL INITIALS/SIGNATURES IS MAINTAINED IN THE FILES OF ENTERGY NUCLEAR NORTHEAST

REV	DESCRIPTION	DATE	BY	CHK'D	APP.
51	INCORPORATED DRN-05 02085, ER-IP3-05-16907	4/21/05	JF		

REVISIONS

DWN | CHK'D | DES SUPV | DISCIPLINE ENG | DISCIPLINE MGR. | PROJ APPROVAL | DATE

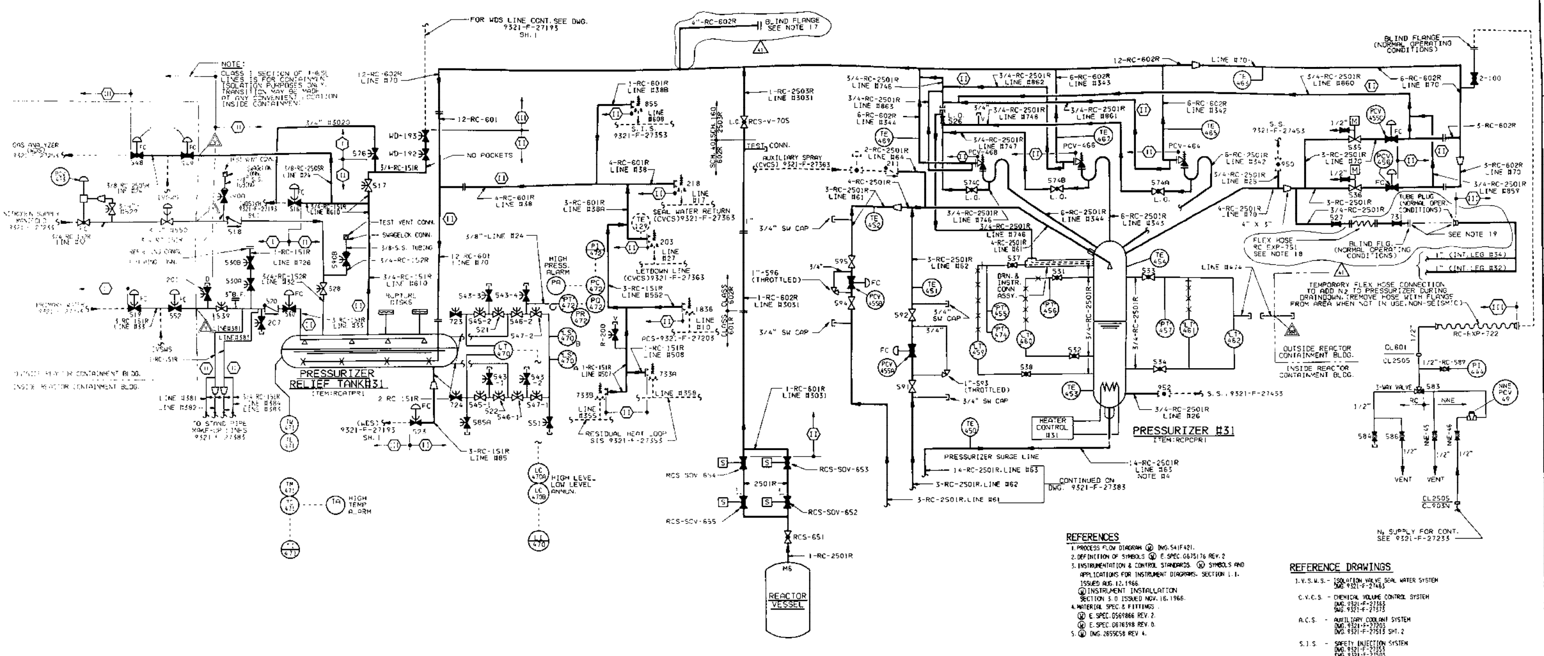
INDIAN POINT NO.3 NUCLEAR POWER PLANT

FLOW DIAGRAM
CHEMICAL & VOLUME CONTROL SYSTEM
SHEET NO.1

Entergy Nuclear Northeast

SCALE: NONE
DWG NO 9321-F-27363
REV 51
SHEET

TYPE A/ISI/FSAR



REFERENCES

1. PROCESS FLOW DIAGRAM (W) DWG. 541F421.
2. DEFINITION OF SYMBOLS (W) E. SPEC. G675176 REV. 2
3. INSTRUMENTATION & CONTROL STANDARDS. (W) SYMBOLS AND 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.
5. (W) DWG. 2655C58 REV 4.

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-LOCATED OUTSIDE SECONDARY SHIELD WALL.
VALVE NO. 507 IS FROM INNER O-RING.
VALVE NO. 502 IS FROM OUTER O-RING.
7-SPRAY LINE SCOOP.
8-ELBOW FLOW METERS.
9-7FT. LG., 6" DIA. SCHEDULE 10S STAND PIPE. BOTH ENDS CAPPED.
10-PIPE SLOPED DOWNHILL TO DRAIN TANK.
11-ALL SCOOPS LOCATED IN THE SAME CROSS-SECTIONAL PLANE.
12-ROOT VALVE IS LOCATED ABOVE ELEVATION OF REACTOR VESSEL NOZZLES.
13-VESSEL VENT LINE AND GASKET MONITORING LINES FURNISHED WITH VESSEL.
14-VALVES 539, 540, 541 & 542 ARE LOCATED BELOW HOT LEG PIPING.
15-FOR SEAL LEAKOFF SEE DWG. 9321-F-26473 FOR VALVES 535, 536, 455A & B.
16-ALL VALVE NO'S. ARE PRECEDED BY "RC" UNLESS OTHERWISE NOTED.
17-THIS CONNECTION SHALL BE USED FOR THE RCS VACUUM REFILL CONNECTION. SEE OPERATING PROCEDURE NO. SOP-RCS-017
18-FLEX HOSE RC-EXP-751 IS A SPECIALTY ITEM NOT GOVERNED BY PIPE CLASS 2501 AND IS RATED FOR THE PRESSURIZER DESIGN CONDITIONS.
19-THESE CONNECTIONS ARE USED WITH LT-390 DURING REDUCED INVENTORY OPERATIONS.

41

REFERENCE DRAWINGS

I.V.S.W.S. - ISOLATION VALVE SEAL WATER SYSTEM DWG. 9321-F-27463
C.V.C.S. - CHEMICAL VOLUME CONTROL SYSTEM DWG. 9321-F-27363, DWG. 9321-F-27373
A.C.S. - AUXILIARY COOLANT SYSTEM DWG. 9321-F-27203, DWG. 9321-F-27513 SHT. 2
S.I.S. - SAFETY INJECTION SYSTEM DWG. 9321-F-27353, DWG. 9321-F-27503
S.S. - SAMPLING SYSTEM DWG. 9321-F-27453
W.D.S. - WASTE DISPOSAL SYSTEM DWG. 9321-F-27193 SHT. 1, DWG. 9321-F-27505
R.C.S. - REACTOR COOLANT SYSTEM DWG. 9321-F-27383
NUCLEAR LINE LIST DWG. 9321-C-27413
REACTOR HEAD VENT SYSTEM DWG. 6604-171-F-1

LEGEND

D T - DRAIN TANK (WDS)
V - VENT TO ATMOSPHERE
F O - FAIL OPEN
F C - FAIL CLOSED
T - TRIP ON CONTAINMENT ISOLATION SIGN
V H - VENT HEADER
D - LOCAL DRAIN
CONTAINMENT PENETRATION
SEISMIC CLASSIFICATION

NOTE:
ALL PIPING ON THIS DWG. IS SEISMIC CLASS I EXCEPT AS NOTED.

REDRAWN FROM (W) DWG. 685J413

REV NO	DATE	DESCRIPTION	DWN/CHK	DS	DE/DD	PROJ MGR
41	[illegible]	AS BUILT PER DCP 00-3-010 RC	[illegible]	[illegible]	[illegible]	[illegible]
40	2/5/01	AS BUILT PER DCR 100159871	JF/VR	JM	LC/SP	REG
39	1/9/00	AS BUILT PER DCR 100159105	JF/VR	JM		REG
38	3/1/00	AS BUILT PER DCR 100155808	VR/LH	JM		REG
37	1/28/99	AS BUILT PER DCR 100149154	VR/LH	JM		REG

REVISIONS

DWN/CHK'D
DES SUPV
DISCIPLINE ENG
DISCIPLINE DIR
PROJ APPROVAL
DATE

INDIAN POINT NO. 3 NUCLEAR POWER PLANT

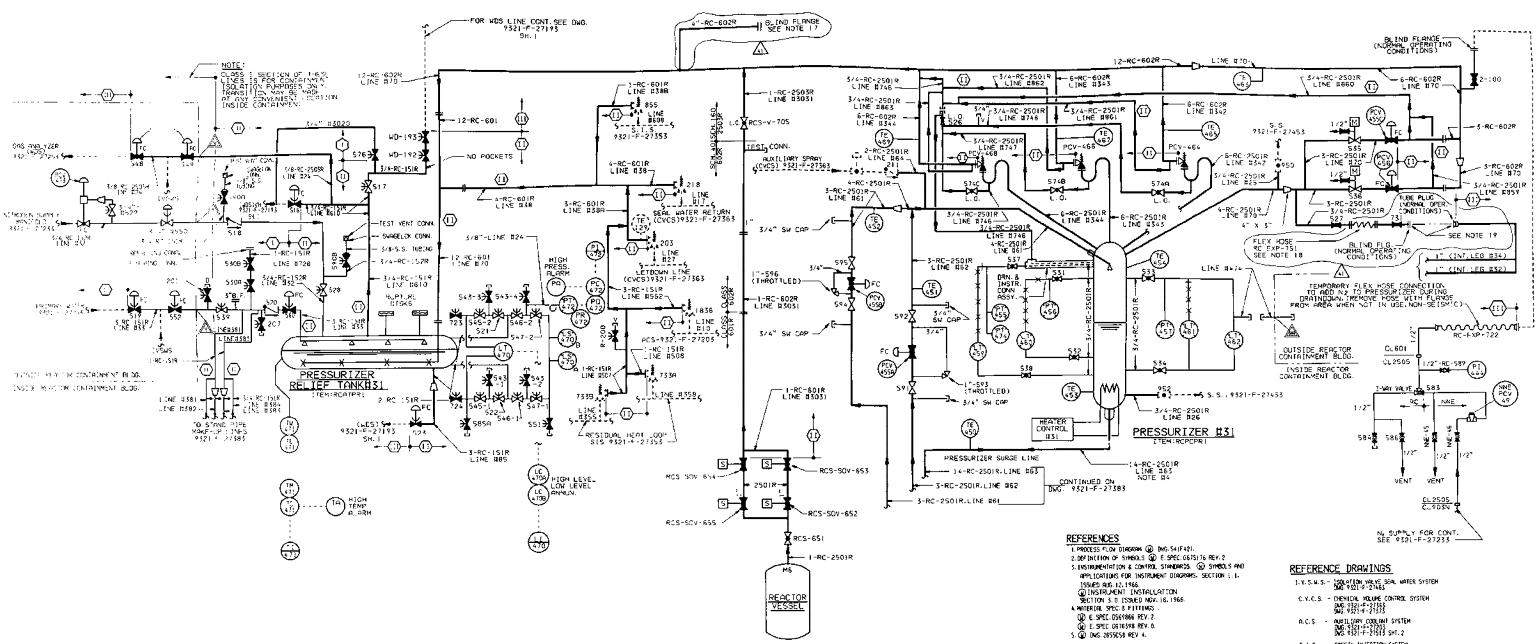
FLOW DIAGRAM
REACTOR COOLANT SYSTEM
SHEET NO. 2

New York Power Authority

SCALE NONE
DWG NO 9321-F-27473
REV 41
SHEET

TYPE A/ISI/FSAR

THIS IS A COMPUTER AIDED DESIGN DRAWING. A RECORD OF THE REVISIONS OF THIS DOCUMENT WITH APPROVAL INITIALS/SIGNATURES, IS MAINTAINED IN THE FILES OF THE NEW YORK POWER AUTHORITY.



REFERENCES

1. PROCESS FLOW DIAGRAM (W) DWG. 541F421.
2. DEFINITION OF SYMBOLS (W) E. SPEC. G675176 REV. 2
3. INSTRUMENTATION & CONTROL STANDARDS. (W) SYMBOLS AND 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.
5. (W) DWG. 2655C58 REV 4.

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-LOCATED 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.
9-7FT. LG., 6" DIA. SCHEDULE 10S STAND PIPE. BOTH ENDS CAPPED.
10-PIPE SLOPED DOWNHILL TO DRAIN TANK.
11-ALL SCOOPS LOCATED IN THE SAME CROSS-SECTIONAL PLANE.
12-ROOT VALVE IS LOCATED ABOVE ELEVATION OF REACTOR VESSEL NOZZLES.
13-VESSEL VENT LINE AND GASKET MONITORING LINES FURNISHED WITH VESSEL.
14-VALVES 539, 540, 541 & 542 ARE LOCATED BELOW HOT LEG PIPING.
15-FOR SEAL LEAKOFF SEE DWG. 9321-F-26473 FOR VALVES 535, 536, 455A & B.
16-ALL VALVE NO'S. ARE PRECEDED BY "RC" UNLESS OTHERWISE NOTED.
17-THIS CONNECTION SHALL BE USED FOR THE RCS VACUUM REFILL CONNECTION. SEE OPERATING PROCEDURE NO. SOP-RCS-017
18-FLEX HOSE RC-EXP-751 IS A SPECIALTY ITEM NOT GOVERNED BY PIPE CLASS 2501 AND IS RATED FOR THE PRESSURIZER DESIGN CONDITIONS.
19-THESE CONNECTIONS ARE USED WITH LT-390 DURING REDUCED INVENTORY OPERATIONS.

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REFERENCE DRAWINGS

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A.C.S. - AUXILIARY COOLANT SYSTEM DWG. 9321-F-27203 DWG. 9321-F-27513 SHT. 2
S.I.S. - SAFETY INJECTION SYSTEM DWG. 9321-F-27353 DWG. 9321-F-27503
S.S. - SAMPLING SYSTEM DWG. 9321-F-27453
W.D.S. - WASTE DISPOSAL SYSTEM DWG. 9321-F-27193 SHT. 1 DWG. 9321-F-27505
R.C.S. - REACTOR COOLANT SYSTEM DWG. 9321-F-27383
NUCLEAR LINE LIST DWG. 9321-C-27413
REACTOR HEAD VENT SYSTEM DWG. 6604-171-F-1

LEGEND

D T - DRAIN TANK (WDS)
V - VENT TO ATMOSPHERE
F O - FAIL OPEN
F C - FAIL CLOSED
T - TRIP ON CONTAINMENT ISOLATION SIGN
V H - VENT HEADER
D - LOCAL DRAIN
CONTAINMENT PENETRATION
SEISMIC CLASSIFICATION

NOTE:
ALL PIPING ON THIS DWG. IS SEISMIC CLASS I EXCEPT AS NOTED.

REDRAWN FROM (W) DWG. 685J413

REV NO	DATE	DESCRIPTION	DWN/CHK	DS	DE/DD	PROJ MGR
41	[illegible]	AS BUILT PER DCP 00-3-010 RC	VR/[illegible]	[illegible]	[illegible]	[illegible]
40	2/5/01	AS BUILT PER DCR 100159871	JF/VR	JN	LC/SP	REG
39	1/1/00	AS BUILT PER DCR 100159105	JF/VR	JN	-/-	REG
38	5/1/00	AS BUILT PER DCR 100155808	VR/LH	JH	-/-	REG
37	1/28/99	AS BUILT PER DCR 100149154	VR/LH	JH	-/-	REG

REVISIONS

DWN/CHK'D	INDIAN POINT NO. 3 NUCLEAR POWER PLANT
DES SUPV	
DISCIPLINE ENG	FLOW DIAGRAM REACTOR COOLANT SYSTEM SHEET NO. 2
DISCIPLINE DIR	
PROJ APPROVAL	
DATE	

New York Power Authority

SCALE NONE

DWG NO 9321-F-27473 SHEET

REV 41

TYPE A/ISI/FSAR

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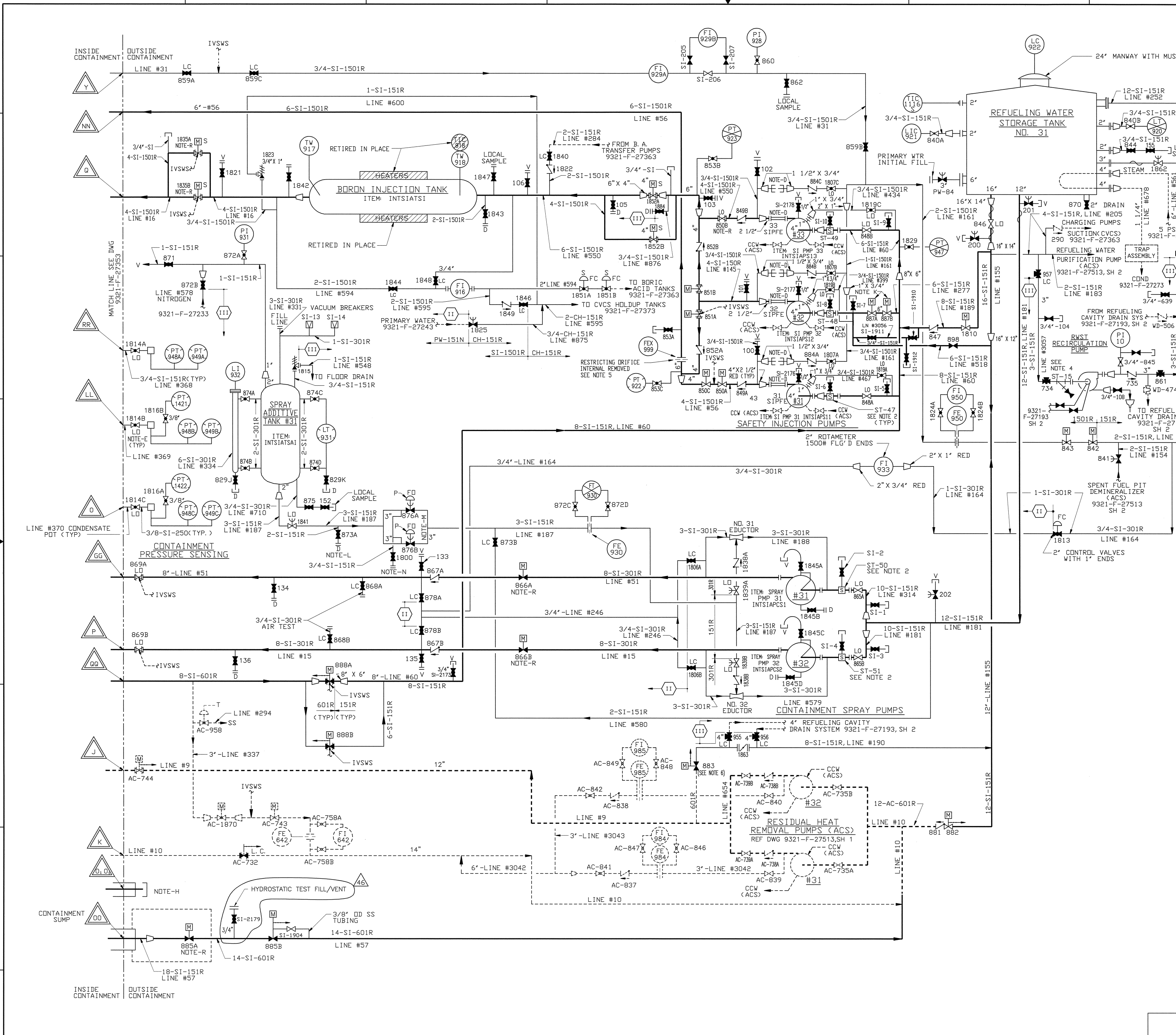


NOTES:

1. ALL PIPING ON THIS DRAWING IS SEISMIC CLASS I EXCEPT AS NOTED.
2. TEMPORARY STRAINERS ST-47 THRU ST-51 ARE PLACED IN THE FLANGED SPOOL PIECE DURING INITIAL FLUSHING OPERATIONS. STRAINERS MUST BE REMOVED BEFORE PLANT START-UP. SCREEN FOR ST-50 HAS BEEN REMOVED PER MWR 93-10142-00, SCREENS FOR ST-47, ST-48, & ST-49 HAVE BEEN REMOVED PER MWR'S 93-10134, 93-10135, & 93-10136. SCREEN FOR ST-51 WAS REMOVED PER MWR 93-10143.
3. ALL VALVE NO'S. ARE PRECEDED BY 'SI' AS THE SYSTEM DESIGNATION EXCEPT AS NOTED.
4. SCREEN FOR ST-15 HAS BEEN REMOVED PER MWR 94-03447
5. REPLACED THE FLOW RESTRICTING ORIFICE PLATE FEX-999 WITH A NON-RESTRICTING ORIFICE PLATE.
6. VALVE 883 HAS A PRESSURE LOCKING RELIEF HOLE IN VALVE INLET DISC.

LEGEND:

SEISMIC CLASSIFICATION

WORK THIS DWG. WITH DWG. 9321-F-27353. FOR ADDITIONAL NOTES AND REFERENCE DWGS SEE 9321-F-27353

THIS IS A COMPUTER AIDED DESIGN DRAWING. A RECORD OF THE REVISIONS OF THIS DOCUMENT WITH APPROVAL INITIALS/SIGNATURES, IS MAINTAINED IN THE FILES OF ENTERGY NUCLEAR NORTHEAST

REDRAWN FROM Ⓦ DWG. 685J429 & 110E364

REV	DESCRIPTION	DATE	BY	CHK'D	APP.
46	INCORPORATED DRN-06-04419, ER-06-3-005	3/21/07	DM		

REVISIONS

INDIAN POINT NO. 3 NUCLEAR POWER PLANT

FLOW DIAGRAM SAFETY INJECTION SYSTEM SHEET NO. 2

DWN | CHK'D | DES SUPV | DISCIPLINE ENG | DISCIPLINE MGR. | PROJ APPROVAL | DATE

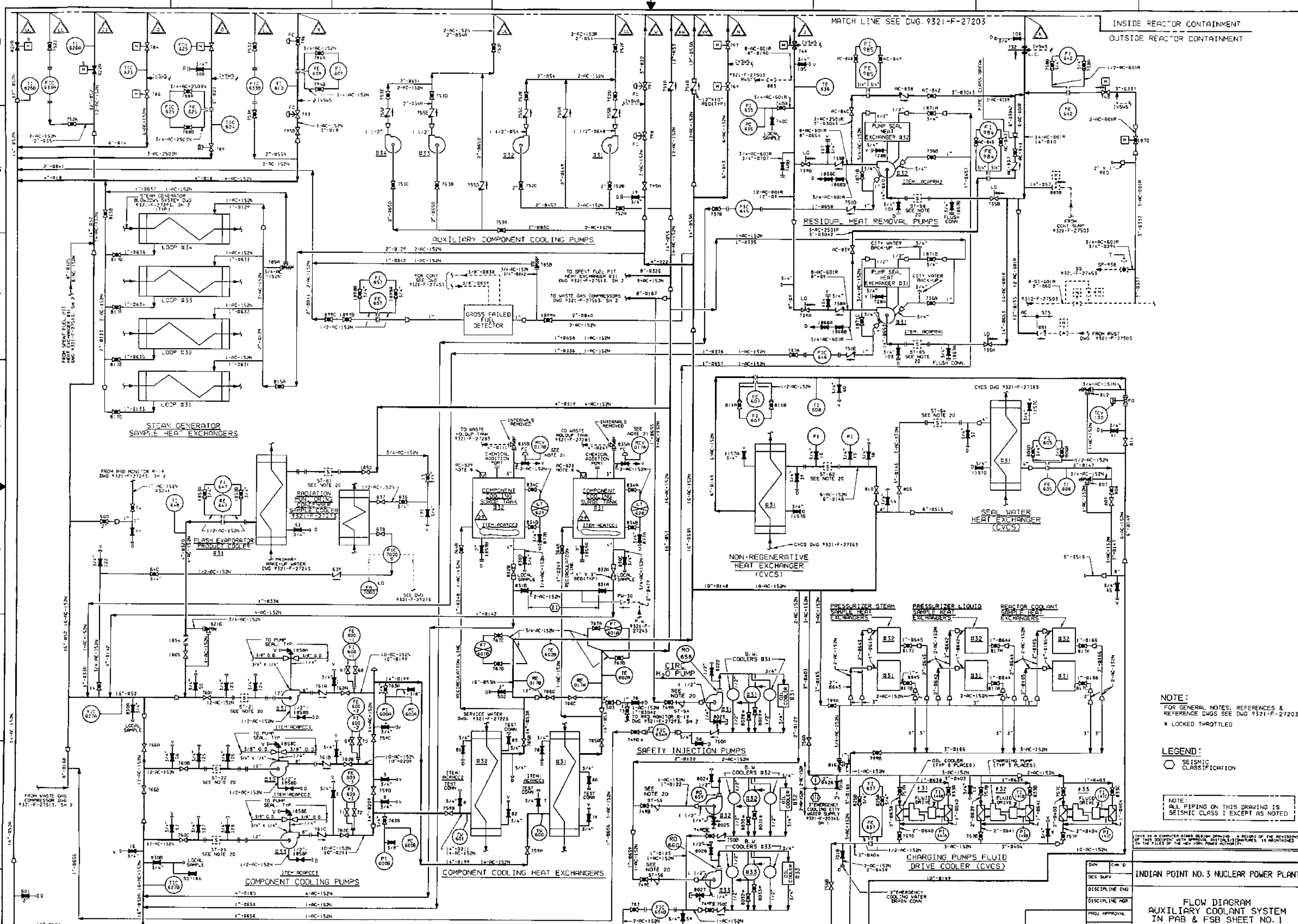
Entergy Nuclear Northeast

SCALE NONE

DWG NO 9321-F-27503 REV 46

SHEET

TYPE A/ISI/FSAR



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REV NO	DATE	DESCRIPTION
29	5/12	AS BUILT PER DCR 100156892
28	3/26/98	AS BUILT PER MMP 95-3-250 CCW
27	3/4/98	AS BUILT PER MMP 95-3-250 CCW
26	1/9/98	AS BUILT PER DCR 100142750
25	7/1/97	AS BUILT PER MOD 96-3-146 CCW
24	1/9/97	AS BUILT PER MMP 95-3-250 CCW
23	3/13/96	AS BUILT PER DCR 100028276
22	9/26/95	AS BUILT PER DCR 100122072
21	9/20/95	AS BUILT PER DCR 100121744
20	6/2/95	AS BUILT PER DCR 100106395 & 100105254

REVISIONS

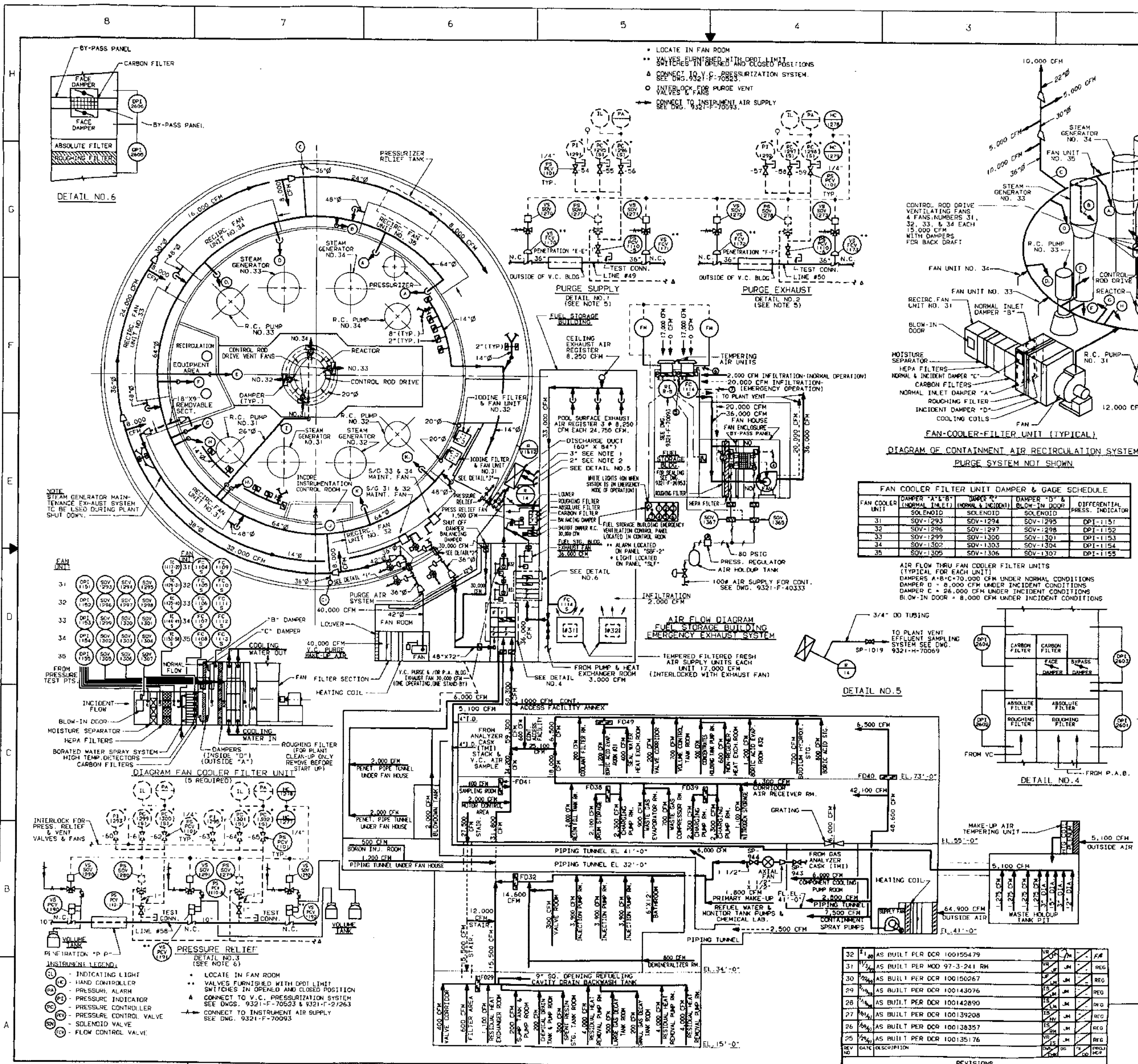
INDIAN POINT NO. 3 NUCLEAR POWER PLANT

FLOW DIAGRAM
AUXILIARY COOLANT SYSTEM
IN PAB & FSB SHEET NO. 1

New York Power Authority

SCALE: NONE
DWG NO 9321-F-27513 SHEET 1
REV 29

TYPE A/ISI/FSAR



FAN COOLER FILTER UNIT DAMPER & GAGE SCHEDULE				
FAN COOLER UNIT	DAMPER "A" & "B" (NORMAL INLET) SOLENOID	DAMPER "C" (NORMAL & INCIDENT) SOLENOID	DAMPER "D" & BLOW-IN DOOR SOLENOID	DIFFERENTIAL PRESS. INDICATOR
31	SOV-1293	SOV-1294	SOV-1295	DPI-1151
32	SOV-1296	SOV-1297	SOV-1298	DPI-1152
33	SOV-1299	SOV-1300	SOV-1301	DPI-1153
34	SOV-1302	SOV-1303	SOV-1304	DPI-1154
35	SOV-1305	SOV-1306	SOV-1307	DPI-1155

AIR FLOW THRU FAN COOLER FILTER UNITS
(TYPICAL FOR EACH UNIT)
DAMPERS A+B+C=70,000 CFM UNDER NORMAL CONDITIONS
DAMPER D = 8,000 CFM UNDER INCIDENT CONDITIONS
DAMPER C = 26,000 CFM UNDER INCIDENT CONDITIONS
BLOW-IN DOOR = 8,000 CFM UNDER INCIDENT CONDITIONS

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 R.C. PUMP & STM. GEN. AREA
- (E)= 17,500 CFM ACROSS REFUELING POOL AREA
- (F)= 10,000 CFM TO RECIRCULATION 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 TO 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

- AIR FLOW DURING ACCIDENT
- NORMAL AIR FLOW
- ROUGHING FILTER
- HEPA FILTER
- CARBON FILTER
- AIRFOIL BLADE DAMPERS
- DAMPERS
- LIMIT SWITCH
- FD__ - FIRE DAMPERS
- (M) AIR OPERATED MOTOR
- (A) HIGH RADIATION ALARM
- N.O. NORMALLY OPEN
- N.C. NORMALLY CLOSED
- ELECT. SUPPLY
- ELECT. SIGNAL
- S.V. SOLENOID VALVE
- (FC) FLOW SWITCH
- (FM) FAN MOTOR
- SW SWITCH FOR MANUALLY PLACING SYSTEM IN EMERGENCY MODE OF OPERATION.
- (R.I.) RADIATION INDICATOR
- PNEUMATIC SIGNAL
- INST. AIR SUPPLY

NOTES

1. 3" FROM POST ACCIDENT CONTAINMENT VENTING SYSTEM
2. 2" #131, FROM GAS DECAY TANKS
3. ALL VALVE NO'S ARE PRECEDED BY "VS" AS THE SYSTEM DESIGNATION EXCEPT AS NOTED.
4. ALL HVAC SYSTEMS ARE SEISMIC CLASS I EXCEPT PAB SUPPLY HEATING COILS, WASTE HOLD-UP AIR TEMPERING UNIT, FSB STEAM SUPPLY AND CONDENSATE.
5. CONTAINMENT ISOLATION VALVES FCV-1170 & FCV-1171 (PURGE SUPPLY) AND FCV-1172 & FCV-1173 (PURGE EXHAUST) ARE CLOSED DURING NORMAL OPERATIONS.
6. CONTAINMENT ISOLATION VALVES PCV-1190, PCV-1191 & PCV 1192 (PRESSURE RELIEF) ARE CLOSED DURING NORMAL OPERATIONS. THESE VALVES MAY BE OPENED INTERMITTENTLY DURING NORMAL OPERATIONS FOR PRESSURE RELIEF.
7. THE VC PURGE AIR SYSTEM WAS DESIGNED FOR 40,000 CFM. THE MEASURED FLOW RATE WAS APPROXIMATELY 28,000 CFM.

SPECIFICATION:

9321-05-45-24 PLANT HEATING, VENTILATING & AIR CONDITIONING SYSTEMS. (REFERENCE ONLY)

REV NO	DATE	DESCRIPTION	DWN/CHK	DS	DE/DO	PROJ MGR
32	[illegible]	AS BUILT PER DCR 100155479	VR	JM		F/M
31	11/3/99	AS BUILT PER MOD 97-3-241 RM	VR/JF	JM		REG
30	3/22/99	AS BUILT PER DCR 100150267	JF/LM	JM		REG
29	2/19/98	AS BUILT PER DCR 100143076	IS/JM	JM		REG
28	1/16/98	AS BUILT PER DCR 100142890	IS/LM	JM		REG
27	8/01/97	AS BUILT PER DCR 100139208	IS/NV	JM		REG
26	7/09/97	AS BUILT PER DCR 100138357	IS/RH	JM		REG
25	4/29/97	AS BUILT PER DCR 100135176	VR/IS	JM		REG

REVISIONS

INDIAN POINT NO.3 NUCLEAR POWER PLANT

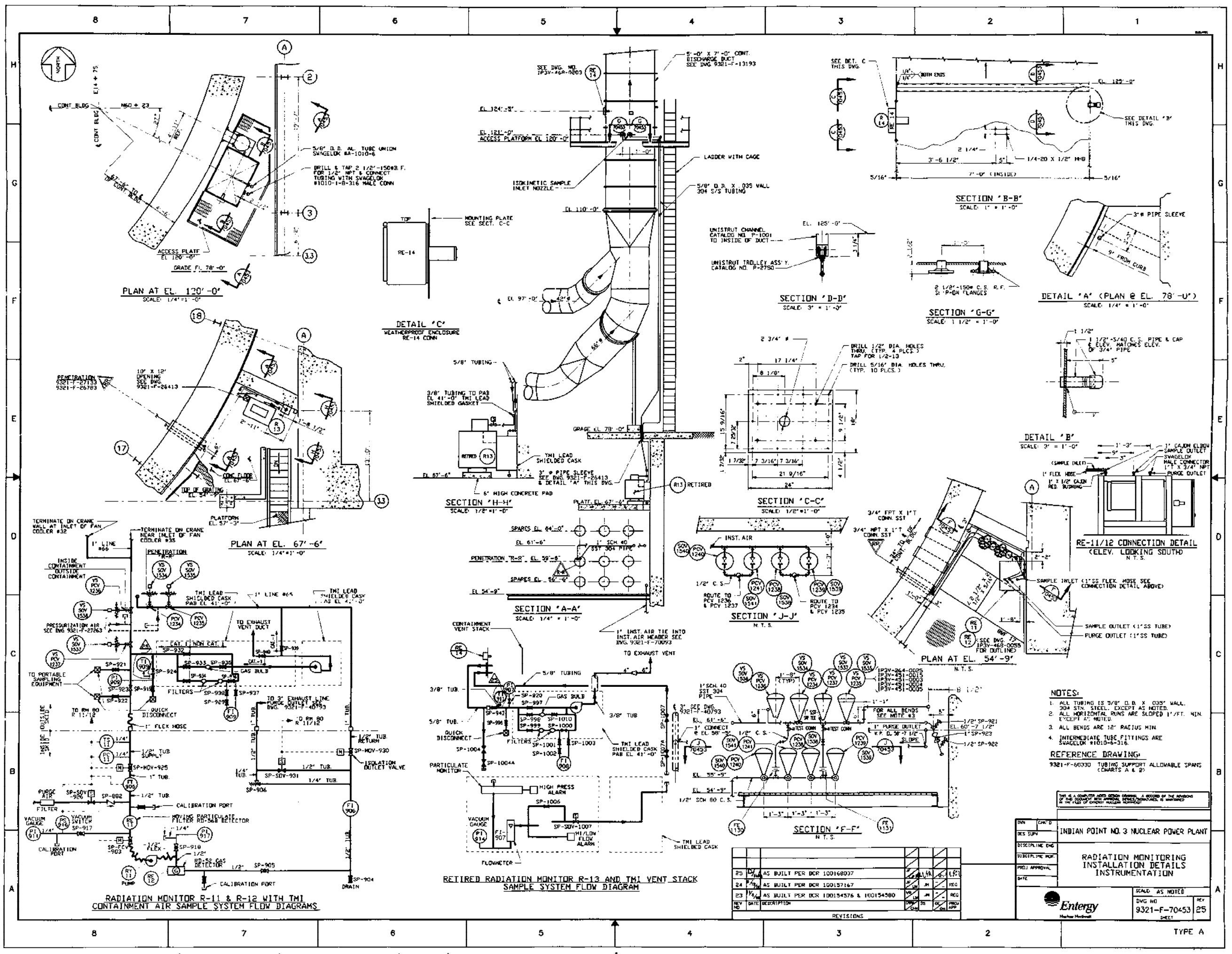
FLOW DIAGRAM
VENTILATION SYSTEM FOR CONTAINMENT, PRIMARY AUXILIARY AND FUEL STORAGE BUILDINGS

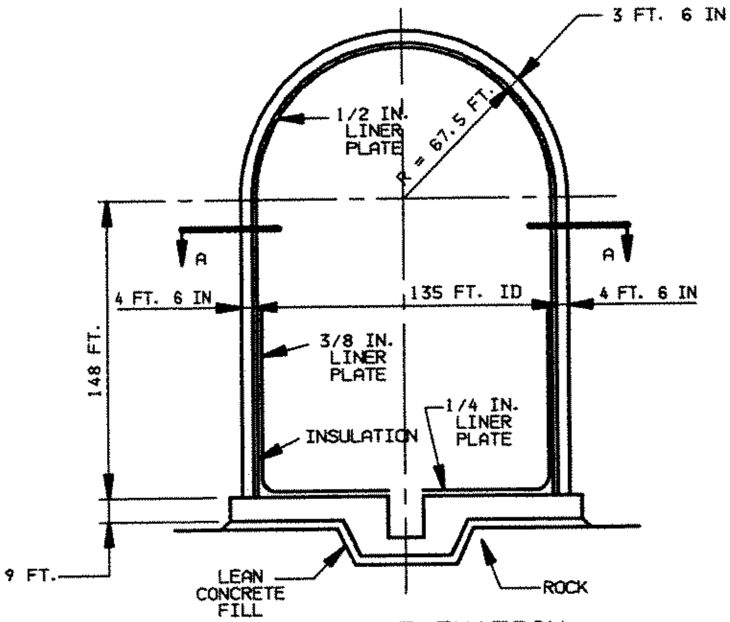
New York Power Authority

SCALE: NONE
DWG NO 9321-F-40223 REV 32

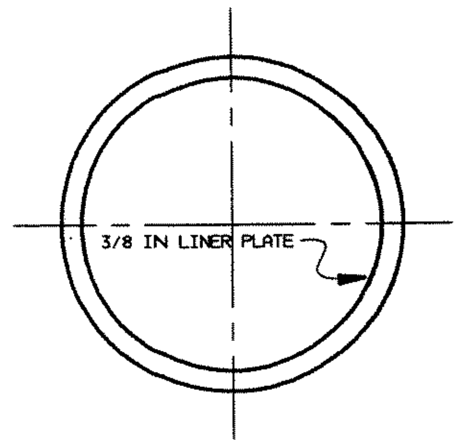
TYPE A/FSAR

THIS IS A COMPUTER AIDED DESIGN DRAWING. A RECORD OF THE REVISIONS OF THIS DOCUMENT WITH APPROVAL INITIALS/SIGNATURES, IS MAINTAINED IN THE FILES OF THE NEW YORK POWER AUTHORITY.

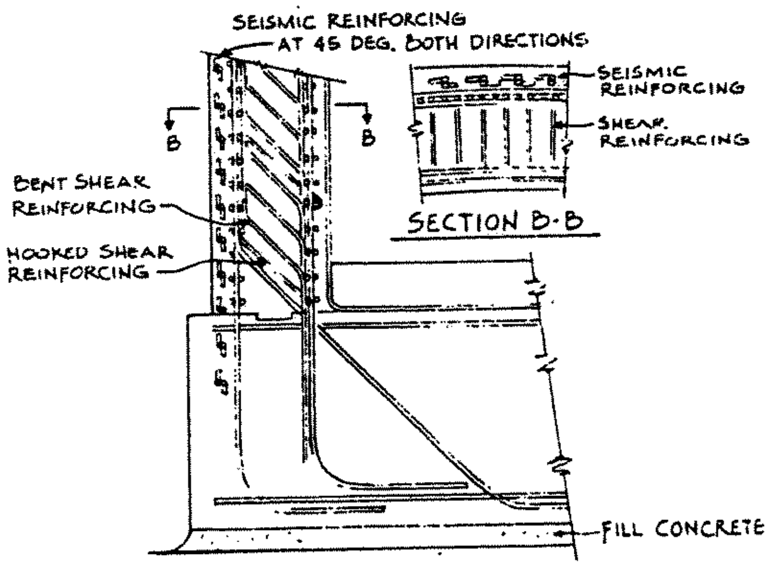

PLAN AT EL. 120'-0"
PLAN AT EL. 67'-6"
DETAIL "C"
WEATHERPROOF ENCLOSURE RE-14 CONN
SECTION "H-H"
SECTION "A-A"
SECTION "B-B"
SECTION "D-D"
SECTION "G-G"
DETAIL "A" (PLAN @ EL. 78'-0")
DETAIL "B"
SECTION "C-C"
SECTION "J-J"
RE-11/12 CONNECTION DETAIL (ELEV. LOOKING SOUTH)
PLAN AT EL. 54'-9"
SECTION "F-F"
RADIATION MONITOR R-11 & R-12 WITH TMI CONTAINMENT AIR SAMPLE SYSTEM FLOW DIAGRAMS
RETIRED RADIATION MONITOR R-13 AND TMI VENT STACK SAMPLE SYSTEM FLOW DIAGRAM
NOTES:
1. ALL TUBING IS 5/8" O.D. X .035" WALL, 304 STN. STEEL, EXCEPT AS NOTED.
2. ALL HORIZONTAL RUNS ARE SLOPED 1"/FT. MIN. EXCEPT AS NOTED.
3. ALL BENDS ARE 12" RADIUS MIN.
4. INTERMEDIATE TUBE FITTINGS ARE SWAGELOK #1010-6-316.
REFERENCE DRAWING:
9321-F-60030 TUBING SUPPORT ALLOWABLE SPANS (CHARTS A & 2)
25 AS BUILT PER DCR 100168037
24 AS BUILT PER DCR 100157167
23 AS BUILT PER DCR 100154376 & 100154580
REV NO DATE DESCRIPTION
REVISIONS
INDIAN POINT NO. 3 NUCLEAR POWER PLANT
RADIATION MONITORING INSTALLATION DETAILS INSTRUMENTATION
Entergy
SCALE AS NOTED
DWG NO 9321-F-70453
REV 25
TYPE A



SECTION ELEVATION

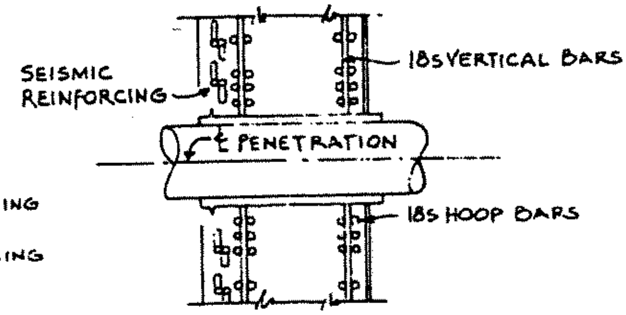


SECTION A-A

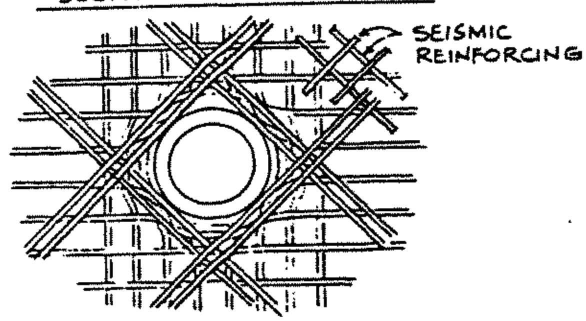


SECTION B-B

TYPICAL REINFORCING AT JUNCTURE OF CYLINDRICAL SHELL AND MAT

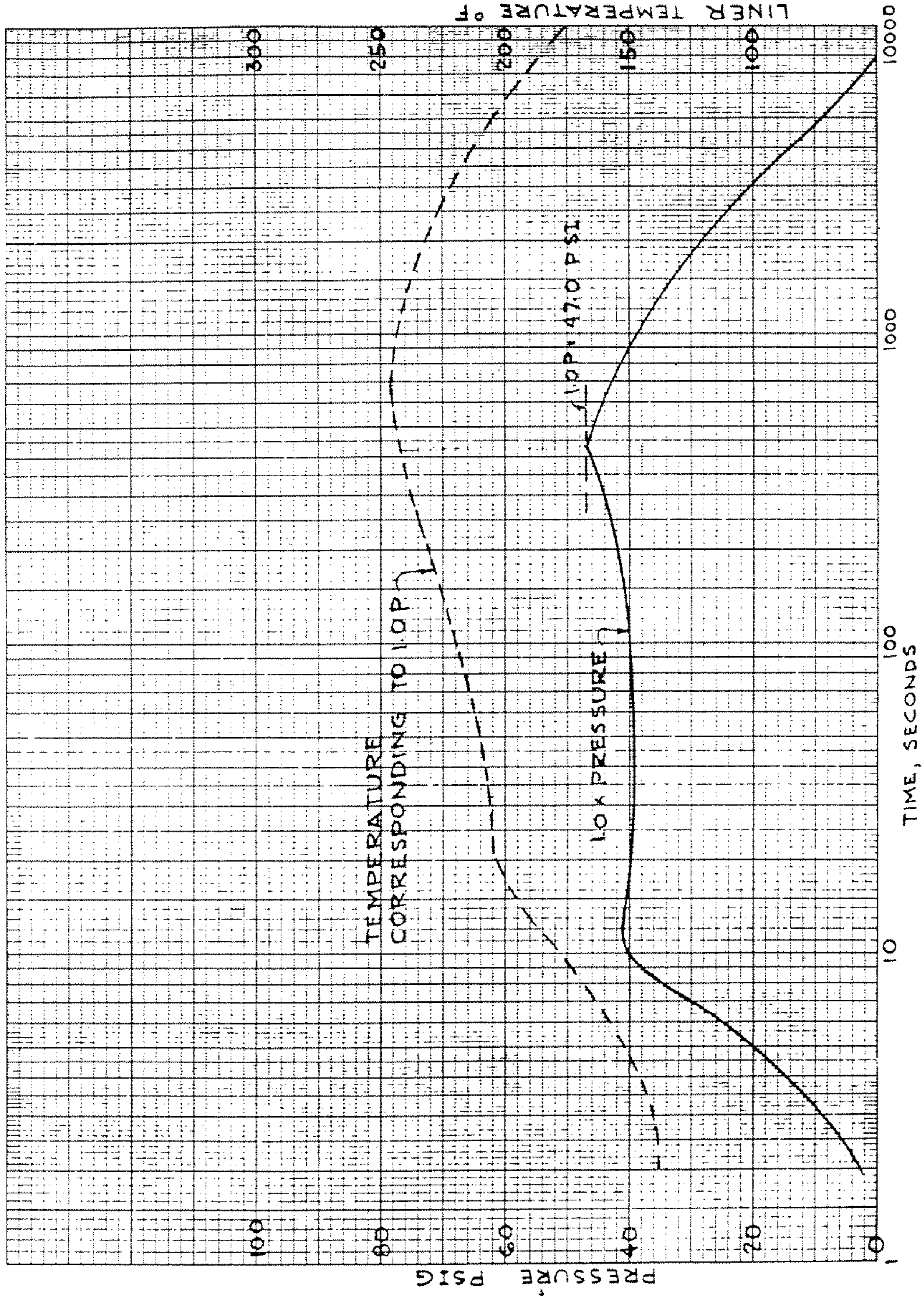


SECTION AT PENETRATION



ELEVATION AT PENETRATION

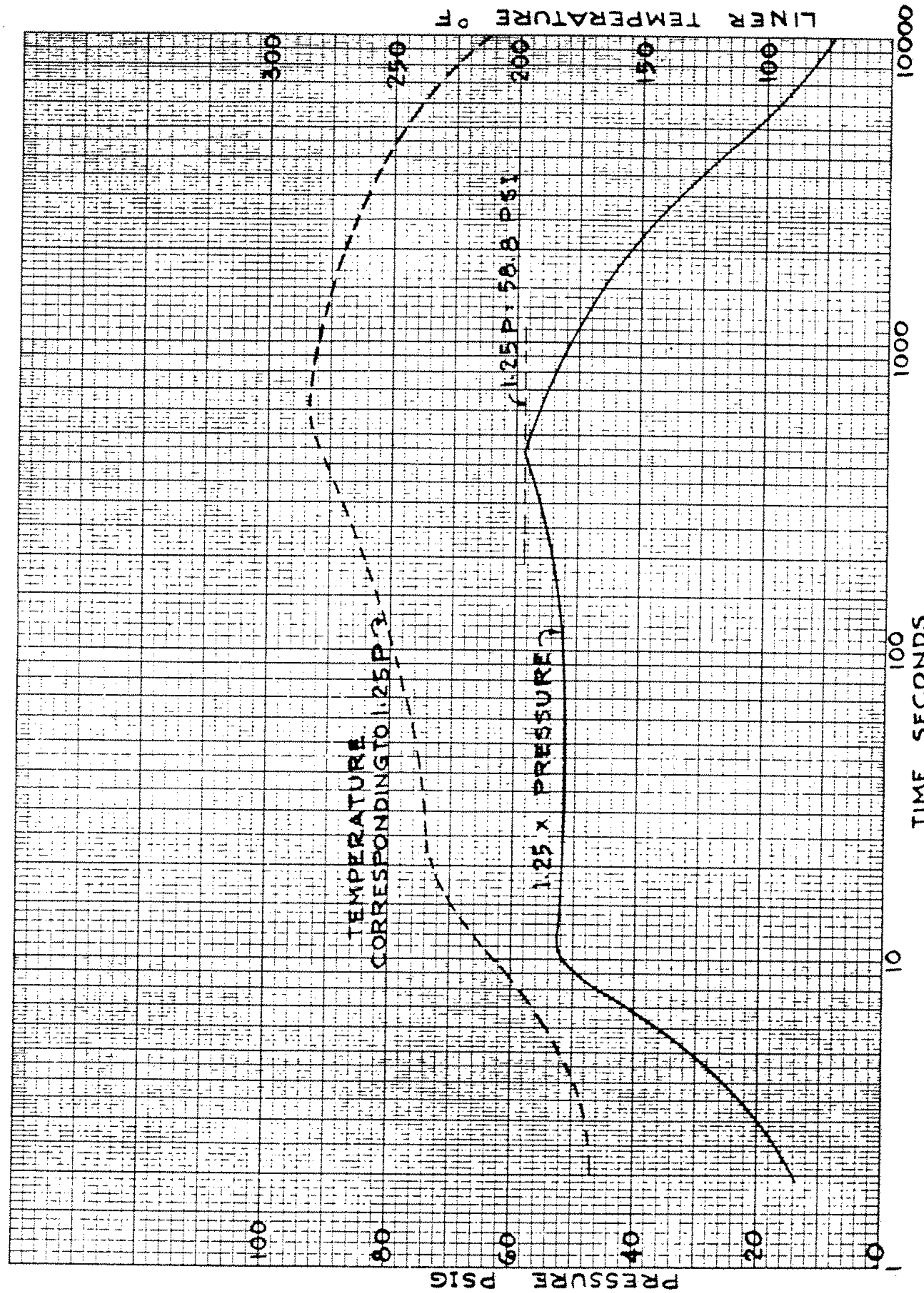
INDIAN POINT 3 FSAR UPDATE	
CONTAINMENT STRUCTURE	
REV. 1 NOV 2001	FIG. NO. 5.1-1



INDIAN POINT 3	FSAR UPDATE

DESIGN PRESSURE -
TEMPERATURE TRANSIENT

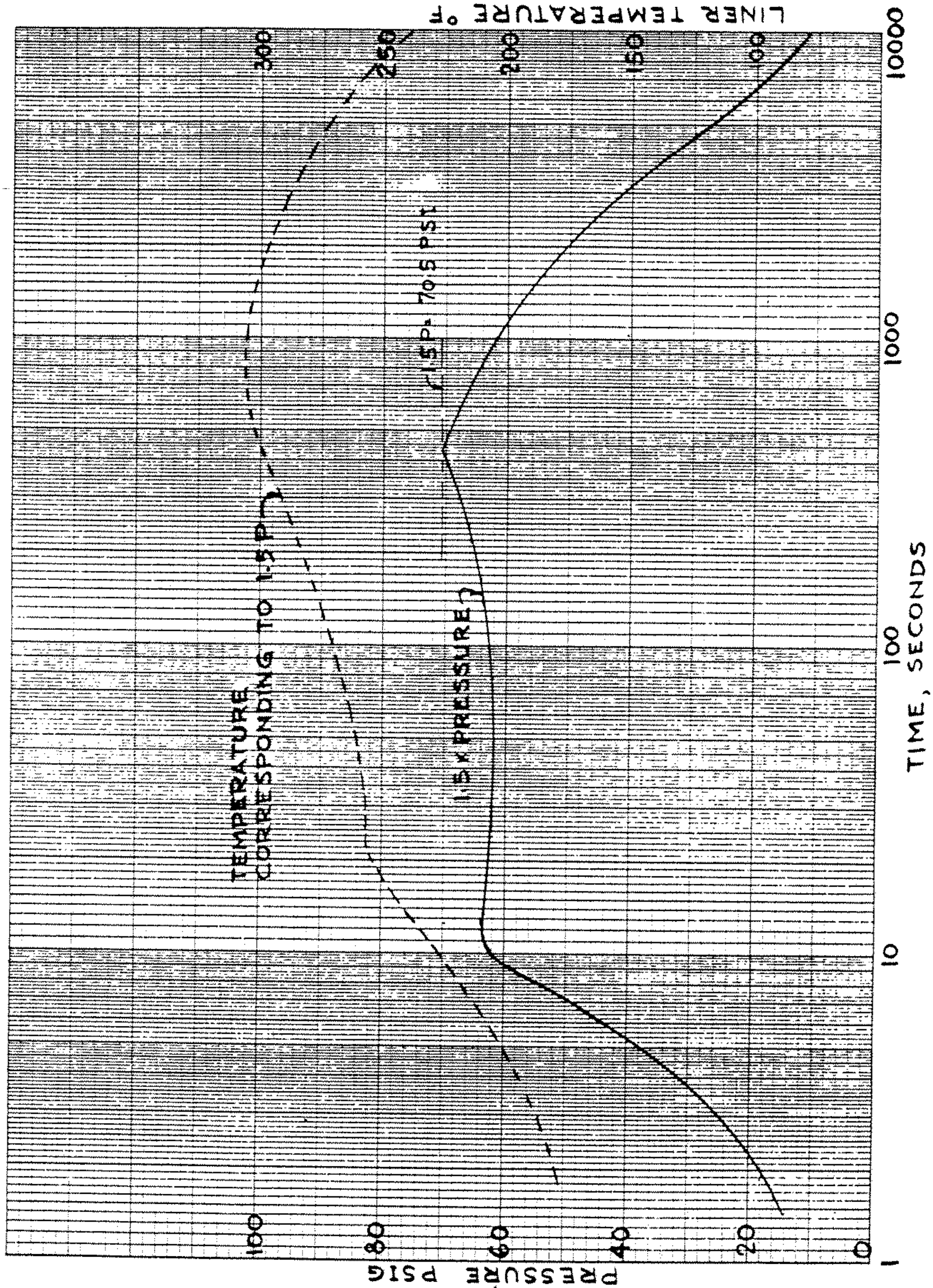
REV. 0	JULY, 1982	FIGURE NO.	5.1-8



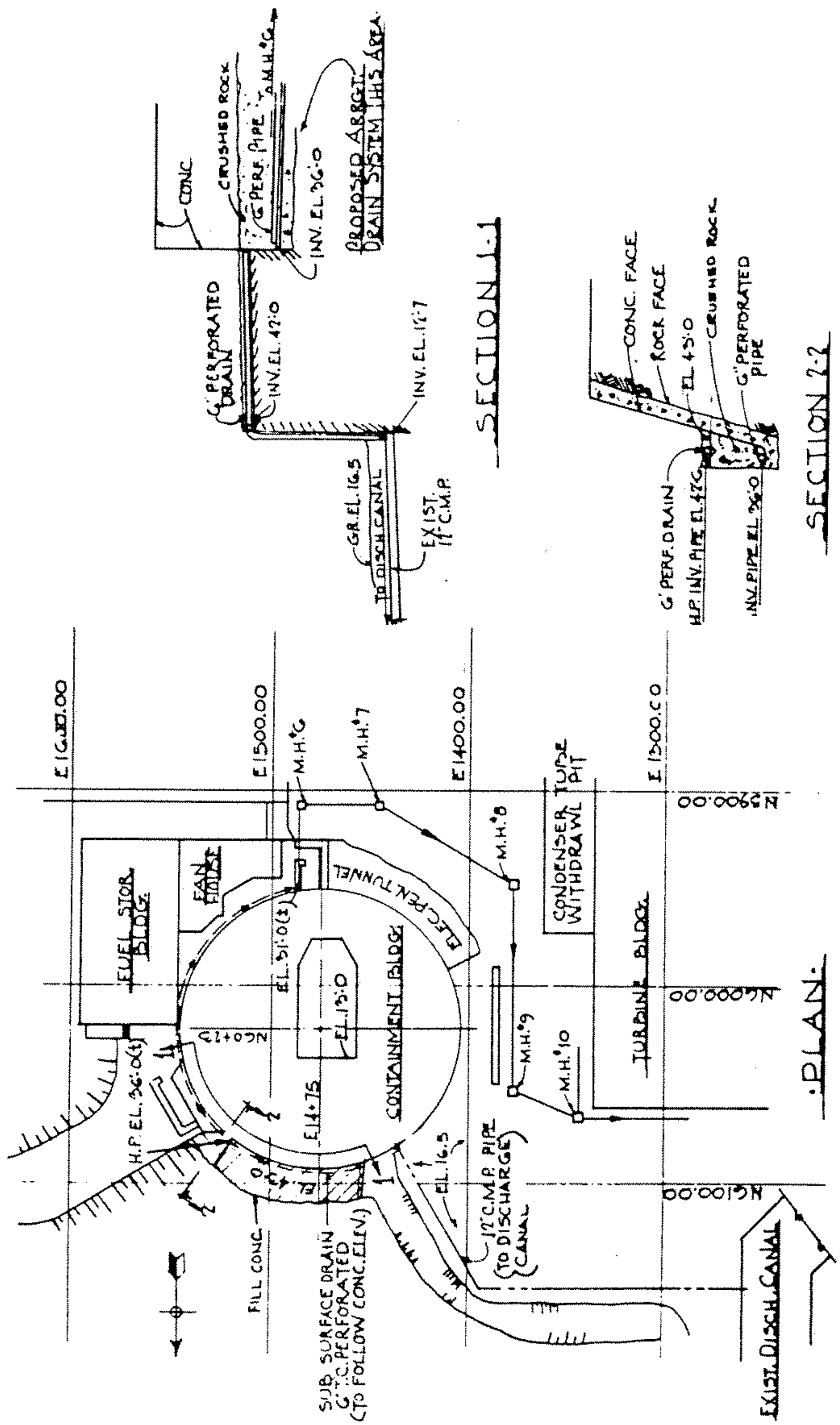
INDIAN POINT 3	FSAR UPDATE

1.25 X DESIGN PRESSURE - TEMPERATURE TRANSIENT

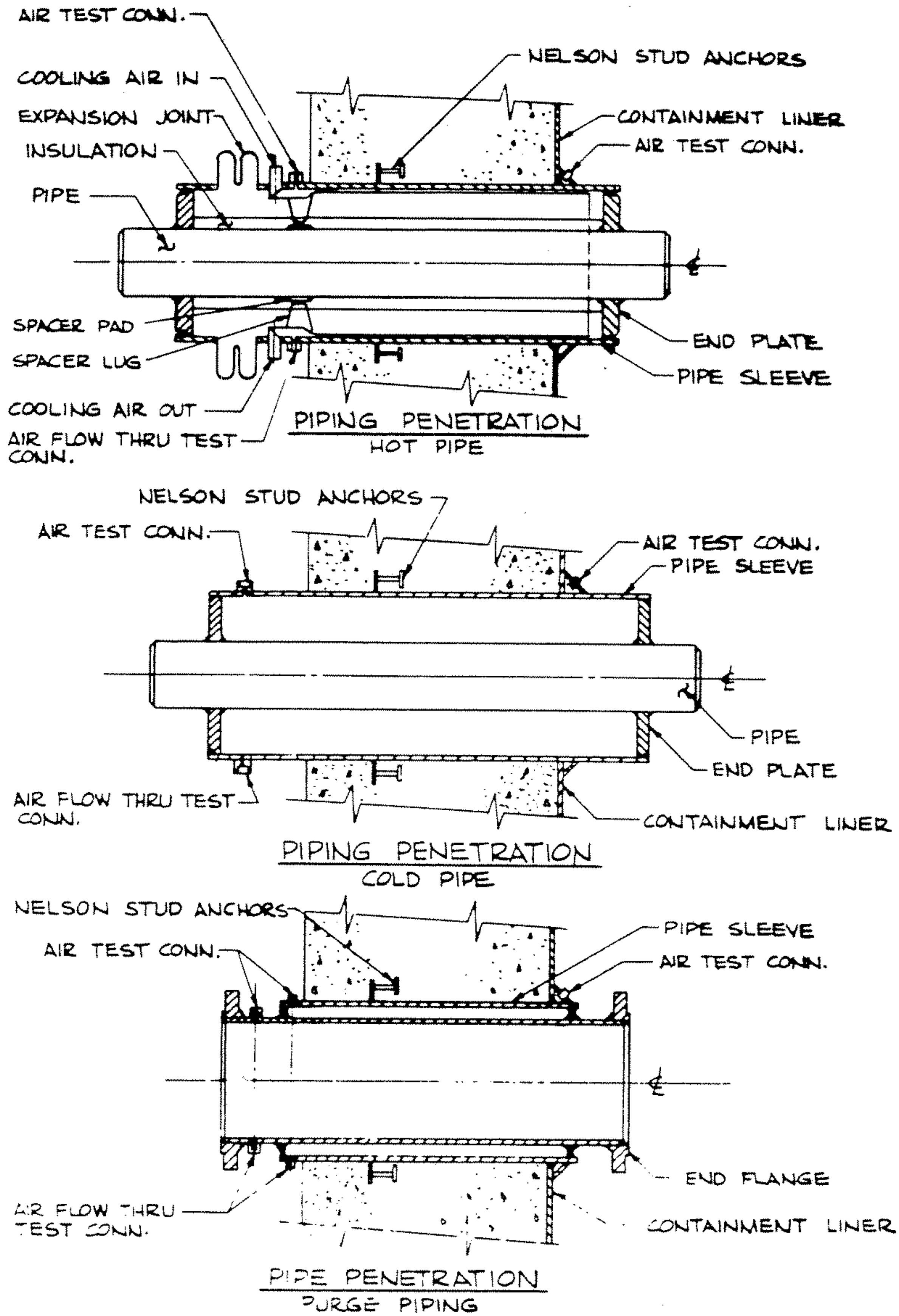
REV 0 JULY, 1982 | FIGURE NO. 5.1-9



INDIAN POINT 3		FSAR UPDATE
1.50 × DESIGN PRESSURE - TEMPERATURE TRANSIENT		
REV. 0	JULY, 1982	FIGURE NO. 5.1-10

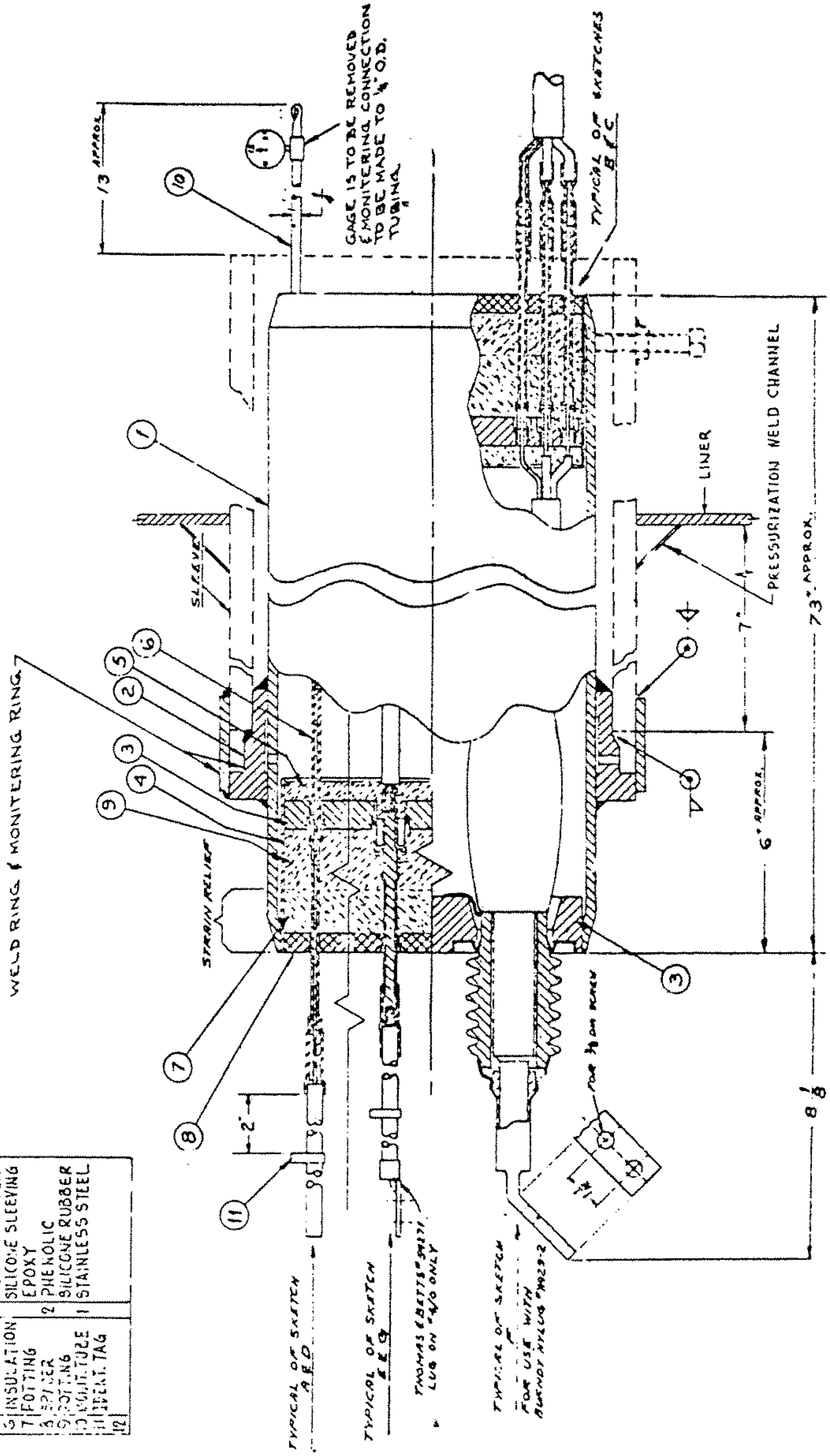


INDIAN POINT 3		FSAR UPDATE
SUB-SURFACE DRAINAGE SYSTEM		
REV. 0	JULY, 1982	FIGURE NO. 5.1-11



INDIAN POINT 3	FSAR UPDATE

TYPICAL PIPING PENETRATION

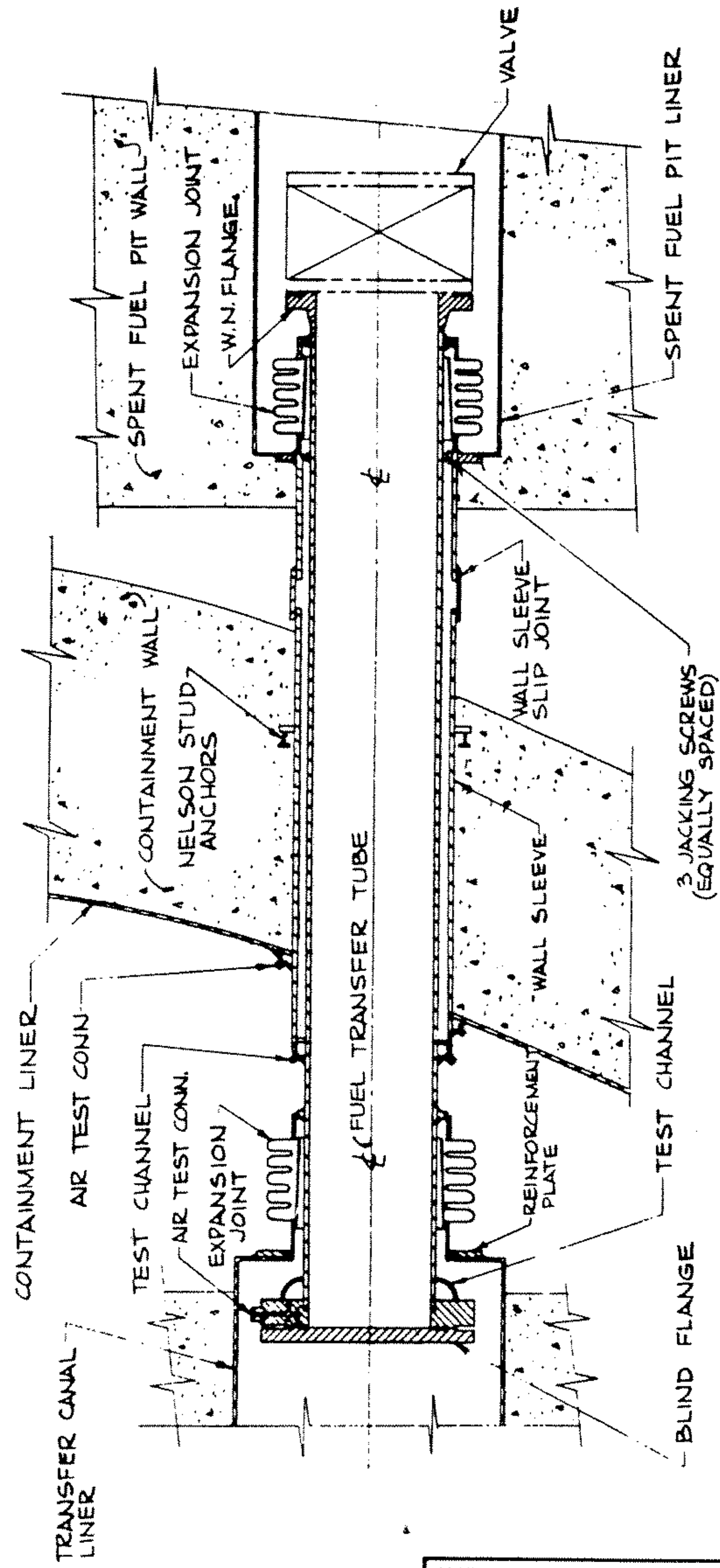


NO	PART NAME	QTY	DESCRIPTION
1	CONTAINER	1	STAINLESS STEEL
2	FLANGE	1	CARBON STEEL
3	HEADER	2	STAINLESS STEEL
4	TUBE	2	STAINLESS STEEL
5	POTTING		SILICONE RUBBER
6	INSULATION		SILICONE SLEEVING
7	POTTING		EPOXY
8	SPACER	2	PHENOLIC
9	POTTING		SILICONE RUBBER
10	MONIT. TUBE	1	STAINLESS STEEL
11	IDENT. TAG		
12			

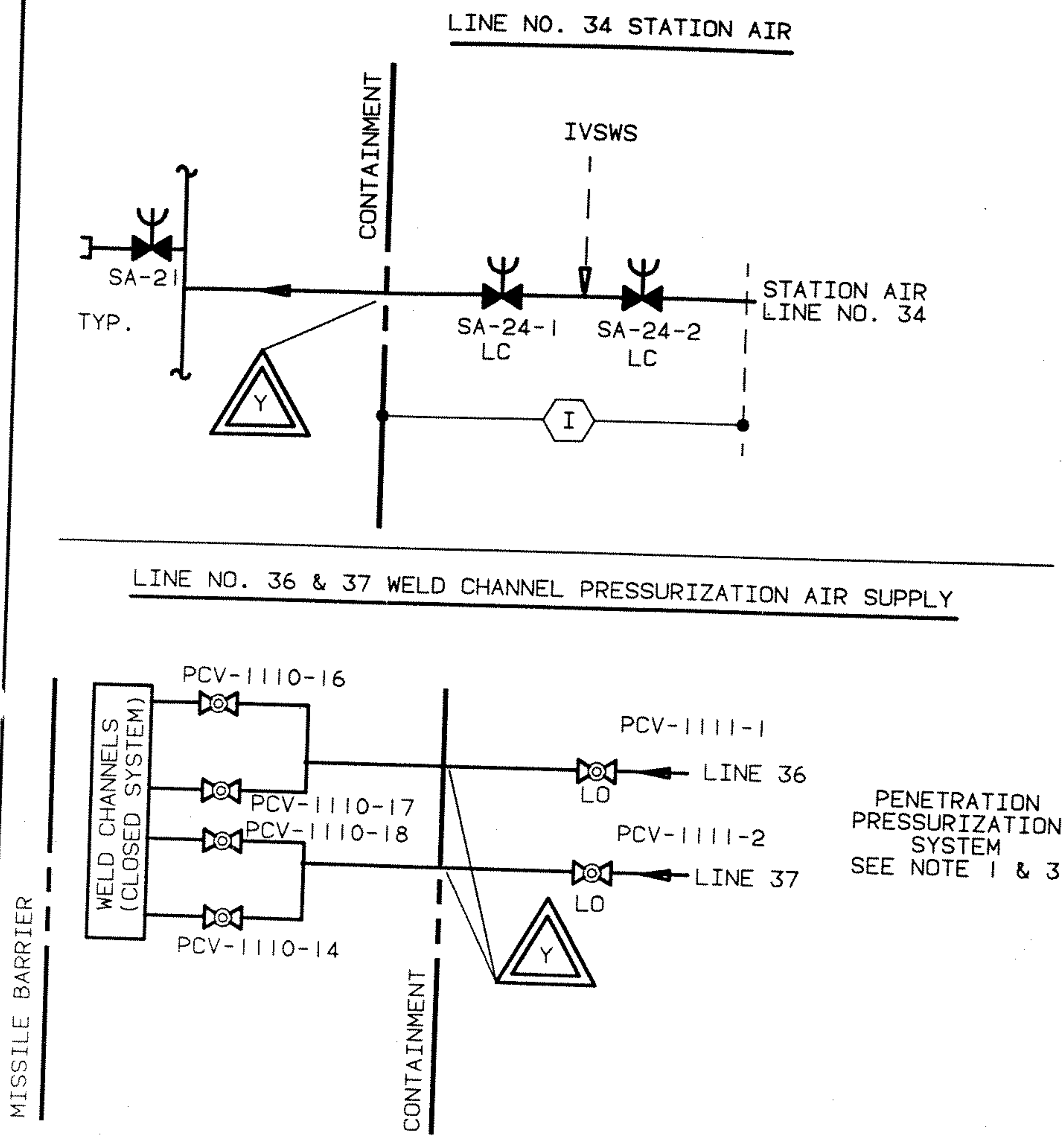
INDIAN POINT 3 **FSAR UPDATE**

TYPICAL ELECTRICAL PENETRATION

REV. 0 JULY 1982 FIGURE NO. 5.1-13



INDIAN POINT 3		FSAR UPDATE
FUEL TRANSFER TUBE PENETRATION		
REV. 0	JULY, 1982	FIGURE NO. 5.1-14



NOTE

1. ENTIRE SYSTEM IN PRESSURIZATION SYSTEM IS SEISMIC CLASS I DESIGN
2. FOR LEGEND SEE FIGURE 5.2-29.
3. VALVE NUMBERS ARE PRECEDED BY PS UNLESS OTHERWISE NOTED.

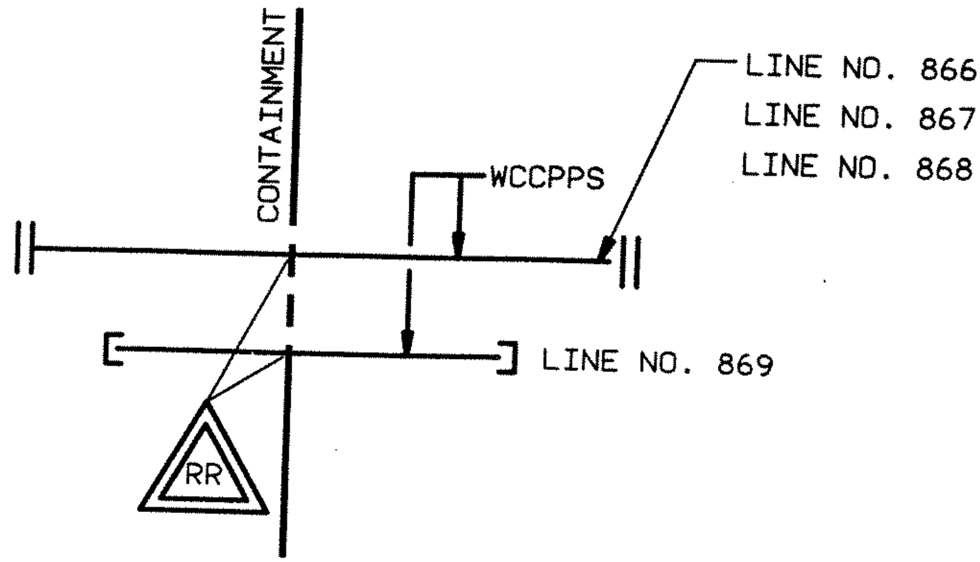
INDIAN POINT 3 FSAR UPDATE

CONTAINMENT ISOLATION SYSTEM SCHEMATICS LINES 34, 36 & 37

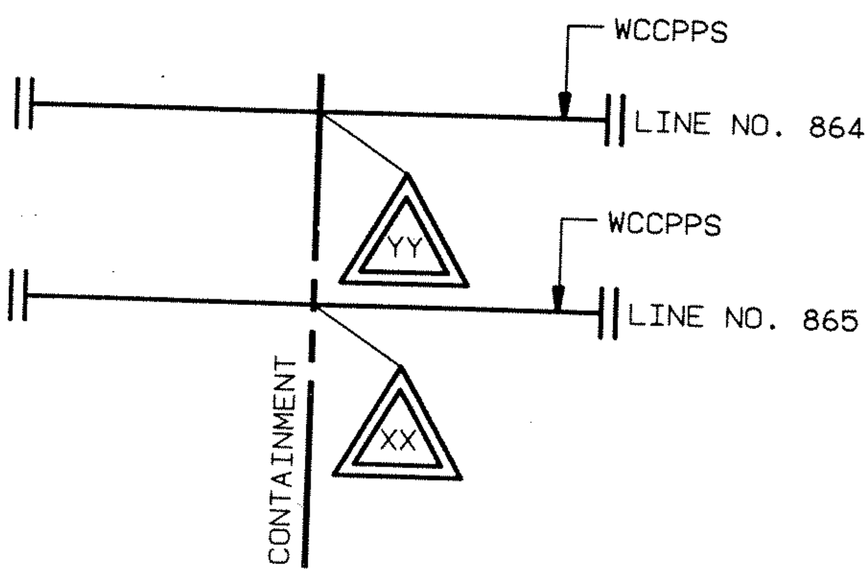
REV. 5 JUN, 2000 | FIGURE NO. 5.2-17

F-20353 REV. 27 / F-27263 REV. 40

LINE NO. 866 THRU 869 CONTAINMENT LEAK TEST INSTRUMENT SENSOR LINE



LINE NO. 864 & 865 CONTAINMENT TEST AIR LINE



NOTES:

1. FOR LEGEND, SEE FIGURE 5.2-29
2. ENTIRE SYSTEM SHOWN IS SEISMIC CLASS I DESIGN.
3. TFP (TEMPORARY FIBER OPTIC PENETRATION FLANGE) MAY BE INSTALLED IN COLD SHUT DOWN/ REFUELING TO SATISFY CONTAINMENT ISOLATION FUNCTION FOR REFUELING OPERATIONS AT EITHER OR BOTH XX OR YY

INDIAN POINT 3	**FSAR UPDATE**
CONTAINMENT ISOLATION SYSTEM SCHEMATICS LINES 864 THRU 869	
REV. 2 JUN, 2000	FIGURE NO. 5.2-26

LEGEND

VALVES

- PLUG
- GLOBE
- DIAPHRAGM
- GATE
- BALL
- DOUBLE DISC GATE
- CHECK
- BUTTERFLY
- RELIEF
- NEEDLE
- NON RETURN (PISTON TYPE)
- SELF CONTAINED PRESSURE REGULATOR

OPERATORS

- AIR DIAPHRAGM
- M AIR CYLINDER (PISTON)
- M MOTOR
- S SOLENOID

VALVE POSITION (NORMAL)

- OPEN
- CLOSED

MISCELLANEOUS

- CONTAINMENT PENETRATION
- S STRAINER
- TEST CONNECTION
- VENT CONNECTION
- DRAIN CONNECTION
- STEMLEAKOFF
- RM RAD. MONITOR

SYSTEM PREFIX IDENTIFICATION

AA	AUTOMATIC PRESSURIZATION WITH AIR
N_2	MANUAL PRESSURIZATION WITH NITROGEN
S	OPENED ON SAFETY INJECTION SIGNAL
T	TRIPPED CLOSED BY CONTAINMENT ISOLATION SIGNAL, PHASE A
P	TRIPPED CLOSED BY CONTAINMENT ISOLATION SIGNAL, PHASE B
P*	TRIPPED OPEN BY CONTAINMENT ISOLATION SIGNAL, PHASE B
LO	LOCKED OPEN
LC	LOCKED CLOSED
I	SEISMIC CLASS I
II	SEISMIC CLASS II
III	SEISMIC CLASS III
NC	NORMALLY CLOSED
SS	SAMPLING SYSTEM
FC	FAIL CLOSED
LT	LOCKED THROTTLED
IVSWS	ISOLATION VALVE SEAL WATER SYS.
DWS	DEMINERALIZED WATER SYSTEM
FPS	FIRE PROTECTION SYSTEM
RCS	REACTOR COOLANT SYSTEM
RHR	RESIDUAL HEAT REMOVAL
PRT	PRESSURIZER RELIEF TANK
DT	REACTOR COOLANT DRAIN TANK
RWST	REFUELING WATER STORAGE TANK
BIT	BORON INJECTION TANK

INDIAN POINT 3 FSAR UPDATE

CONTAINMENT ISOLATION SYSTEM SCHEMATICS-LEGEND FOR SYMBOLS AND NOTATION

REV. 4 JUN. 1999 FIGURE NO. 5.2-29

CHAPTER 6

ENGINEERED SAFETY FEATURES

6.0 GENERAL DESIGN CRITERIA

Criteria applying in common to all engineered safety features are given in Section 6.1.1. Criteria which are related to engineered safety features, but which are applicable to specific features or systems, are listed and cross referenced in Section 6.1.2.

The engineered safety features are discussed in detail in this Chapter. In each section a separate safety feature is described and evaluated. In the evaluation section for each engineered safety feature, a single failure evaluation is provided which delineates the components of that safety feature system and the interconnected auxiliary systems that must function for the proper operation of that engineered safety feature. An examination of these tables shows that some components of the Residual Heat Removal System, Component Cooling Water System, and the Service Water Systems are necessary for proper operation of the Engineered Safety Features. These systems and their components are discussed in Sections 9.3 and 9.6; the instrumentation associated with these systems is also discussed in the referenced sections. Since the auxiliary system components, both inside and outside the containment, and their instrumentation and power systems are not required for actuation of the engineered safety features, neither IEEE-279 nor the General Design Criteria apply.

The General Design Criteria presented and discussed in this section are those which were in effect at the time when Indian Point 3 was designed and constructed. These general design criteria, which formed the bases for the Indian Point 3 design, were published by the Atomic Energy Commission in the Federal Register of July 11, 1967, and subsequently made a part of 10 CFR 50.

The Authority has completed a study of compliance with 10 CFR Parts 20 and 50 in accordance with some of the provisions of the Commission's Confirmatory Order of February 11, 1980. The detailed results of the evaluation of compliance of Indian Point 3 with the General Design Criteria established by the Nuclear Regulatory Commission (NRC) in 10 CFR 50 Appendix A, and in effect at the time of study, were submitted to NRC on August 11, 1980, and approved by the Commission on January 19, 1982. These results are presented in Section 1.3.

6.1. Engineered Safety Features Criteria

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. (CDC 37 of 7/11/67)

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 were designed to cope with any size reactor coolant pipe break, up to and including the circumferential rupture of any pipe, assuming unobstructed discharge form both ends, and to cope with any steam or feedwater line break, up to and including the main steam or feedwater headers.

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 passive accumulator tanks which are self actuated 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 steel-lined, reinforced concrete Reactor Containment with testable, doubly sealed penetrations and most liner weld channels, the spaces of which are continuously pressurized above accident pressure, and which form a virtually leak-tight barrier to the escape of fission products should a loss-of-coolant accident occur.

   Isolation of process lines by the Containment Isolation System which imposes double barriers in each line which penetrates the containment except for lines utilized during the accident. An Isolation Valve Seal Water System provides a water or nitrogen seal at the isolation valves thus sealing some of the pipes penetrating the containment.

2. Reducing the fission product concentration in the containment atmosphere. This is accomplished by:
   a) Containment Air recirculation filters which provide for rapid removal of particles and iodine vapor from the containment atmosphere.
   b) Chemically treated spray which removes elemental iodine vapor 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:
   a) Containment Spray System
   b) Containment Air Recirculation and Cooling System

<u>Reliability and Testability of Engineered Safety Features</u>

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 of 7/11/67)

A comprehensive program of plant testing was formulated for all equipment, systems and system controls vital to the functioning of engineered safety features. The program consists of performance tests of individual pieces of equipment in the manufacturer's shop, 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.

The initial tests of individual components and the integrated test of the system as a whole complemented each other to assure performance of the system as designed and to demonstrate the proper operation of the actuation circuitry.

Routine periodic testing of the engineered safety features components is performed as specified in the Technical Specifications.

Missile Protection

Criterion: Protection for engineered safety features shall be provided against dynamic effects and missiles that might result from plant equipment failures. (GDC 40 of 7/11/67)

A Loss-of-Coolant Accident or other plant equipment failures might result in dynamic effects or missiles. For engineered safety features which are required to ensure safety in the event of such an accident or equipment failure, protection is provided primarily by the provisions which 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. See Chapter 5 for a discussion of missile protection. The dynamic effects associated with postulated pipe breaks in the Primary Coolant System (hot legs, cold legs, crossover legs) need not be a design basis (NRC SER dated March 10, 1986).

Injection paths leading to unbroken reactor coolant loops are protected against damage as a result of 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. Separation of the individual injection lines is provided to the maximum extent practicable. Movement of the injection lines, 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.

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 from these missiles.

All hangers, stops and anchors were designed in accordance with ANSI B31.1 Code for Pressure Piping and ACI 318 Building Code Requirements for Reinforced Concrete which provide minimum requirements on material, design and fabrication with ample safety margins for both dead and dynamic loads over the life of the equipment. Additional information on the design and re-analyses of hangers, stops and anchors is presented in Section 16.3.

Where necessary to prevent pipe whip, restraints were installed with the proper arrangement and spacing to prevent a plastic hinge mechanism from forming as a result of the forces associated with a pipe rupture. Restraint spacing was determined by calculation of the unsupported pipe length resulting in a plastic hinge formation for two basic support arrangement and break location cases. Both slot and guillotine breaks were considered. Slot breaks are defined as instantaneous openings in the pipe parallel to the axis of the pipe with an opening length twice the length of the nominal pipe diameter and with an opening area equal to the area of the pipe interior cross-section.

Slot breaks were assumed to occur anywhere in the piping system, including fittings. Guillotine breaks are defined as instantaneous severance of the pipe cross-section and are assumed to occur any point of discontinuity in the piping system (such as valves, fittings and elbows). The pipe break loads were determined from

$$P' = P_O A$$

where P_O = system pressure

A = inside cross sectional area,

except for the main steam lines downstream of the flow limiting device, where force resultants are limited by the restriction of the flow limiting device. Such loads for the steam lines were taken as P' = 340 kips; and for the feedwater lines, P' = 200 kips. For both slot and guillotine breaks, restraints were spaced such that plastic hinge mechanisms cannot form in the piping system which would permit unrestrained rotation of the piping.

The restraints were designed such that the maximum applied load or stress be less than the lesser of the yield strength of the material or 0.67 times the rated ultimate load capacity of the support. High strength cable restraints were designed such that the maximum applied load be less than 0.4 times the rated ultimate load capacity of the cable. In those instances where the integrity of the restraint is also dependent on reinforced concrete anchorage, the concrete behavior limits are in accordance with ACI-318-63, Part IV-B, requirements and bearing stress is limited to 0.8 f'_c.

Vital equipment is protected from pipe whip by locating restraints on nearby high pressure lines such that the two free ends of a broken pipe cannot reach the equipment.

The plant arrangement provides the basic protection against pipe whip. The four loops of the primary coolant system are spaced to the maximum extent possible; the crane wall protects the reactor compartment from pipe whip in the annulus; pipe lines are run radially outward from the reactor compartment. Wherever possible, redundant engineered safeguards piping is physically separated so that a failure of one pipe and subsequent whipping cannot cause the failure of the second pipe. Where physical separation is impossible, for instance the Accumulator Tanks' discharge piping, both pipes are restrained in such a way that a plastic hinge cannot form in case of a double ended rupture.

Containment fan cooler units are separated from high pressure pipe lines by the floor at Elev. 68' –0”.

Small lines are treated no differently from large lines in so far as containment isolation, separation, pipe whip protection, etc. Separation is provided where whipping of larger lines would otherwise result in damage to many small lines.

Small lines having significant internal pressure are supported and restrained in a manner that would preclude any failure of the containment vessel from the failure of the small line. In addition, see Section 5.2 for the containment isolation provisions for these lines.

<u>Engineered Safety Features Performance Capability</u>

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 of 7/11/67)

Each engineered safety feature provides sufficient performance capability to accommodate any single failure of an active component and still function in 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 100, i.e., 25 rem to the whole body (TEDE) in two hours at the exclusion radius and 25 rem to the whole body (TEDE) over the duration of the accident at the low population zone distance. The accident condition considered is the hypothetical case of a release of fission products per Alternate Source Term (NUREG-1465 / Regulatory Guide 1.183). Also, the total loss of all outside power is assumed concurrently with this accident. With all engineered safety features systems functioning at full capacity, the offsite exposure would be within 10 CFR 20 limits.

Under the above accident conditions, the Containment Air Recirculation Cooling and Filtration System and the Containment Spray System are designed and sized so that either system operating with partial effectiveness is able to supply the necessary post-accident cooling capacity to assure 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 operation of a system with at least one active component failure. Both systems together, each operating with partial effectiveness, are capable of providing the necessary post-accident iodine removal such that the resulting off-site exposures are within the guidelines of 10 CFR 100.

<u>Engineered Safety Features Components Capability</u>

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 of 7/11/67)

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

The Safety Injection System pipes serving each loop are anchored at the crane wall, which constitutes the missile barrier in each loop area, to restrict potential accident damage to the portion of piping beyond this point. The anchorage was designed to withstand, without failure, the thrust force of any branch line, severed from the reactor coolant pipe and discharging fluid to the atmosphere; and to withstand a bending moment equivalent to that which produces failure of the piping under the action of free discharge to atmosphere or 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.

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 of 7/11/67)

The reactor is to be maintained subcritical following a pipe rupture accident. Introduction of borated cooling water into the core results in a net negative reactivity addition. The control rods are inserted and remain inserted.

The supply of water by the Safety Injection System to cool the core cladding reduces the potential for significant metal-water reaction (less than 1.0%).

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.

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 of 7/11/67)

The residual heat removal pumps and heat exchangers serve dual functions. Although the normal duty of the residual heat exchangers and residual heat removal pumps is performed during periods of reactor shutdown, during all plant operating periods these residual heat removal pumps are aligned to perform the low head safety injection function. In addition, during the recirculation phase of a Loss-of-Coolant Accident, the residual heat exchangers of this system perform the core cooling function and the containment cooling function as part of the Containment Spray System, and the residual heat removal pumps, which are part of the external recirculation loop, provide back-up capability to the recirculation pumps which comprise part of the internal recirculation loop as described in Section 6.2.3.

Demonstration checking of the system, performed as dictated by the Technical Specifications, provides assurance of correct system alignment for the safety injection function of the components.

During the injection phase, the safety injection pumps do not depend on any portion of other systems. During the recirculation phase, if Reactor Coolant System pressure stays high due to a small break accident, suction to the safety injection pumps is provided by the internal recirculation pumps, and can also be provided by the Residual Heat Removal pumps.

The Containment Air Recirculation and Filtration System also serves the dual function of containment cooling during normal operation and containment cooling after an accident. Since the method of operation for both cooling functions is the same, the dual aspect of the system does not affect its function as an engineered safety feature.

The steam supply and city water systems at the Indian Point site were shared by all three reactor facilities. However, independent steam supply and city water systems have been installed at Indian Point 3 (See Chapter 9); the city water system for Indian Point 2 is presently used by Indian Point 3 as a backup supply. The steam supply and city water systems are used for the following purposes:

a) Steam for unit heaters for standby heating.
b) Steam to valved hose connections for maintenance purposes.
c) Water to emergency showers.
d) Water to hose connections for maintenance purposes.
e) (Deleted)
f) Water supply to fire protection tanks.
g) Water supply for make-up demineralizers in Condensate Polishing Facility (CPF).
h) Redundant source of makeup water to the spent fuel pit.
i) Backup water supply to Charging Pumps' Fluid Drive Coolers.

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 sections, as listed:

Title of Criterion (7/11/67 issue)	Reference
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 and GDC 24)	Chapter 8

6.2 SAFETY INJECTION SYSTEM

6.2.1 Design Basis

The General Design Criteria presented and discussed in this section are those which were in effect at the time when Indian Point 3 was designed and constructed. On November 22, 1965, the Atomic Energy Commission (AEC) published and requested comments on Proposed General Design Criteria which were developed to assist in the evaluation of applications for nuclear power plant construction permits. On July 11, 1967, a revised set of General Design Criteria were published for comment. The revision reflected extensive public comments, suggestions from meetings with the Atomic Industrial Forum (AIF) and review within the AEC. In the July to October 1967 time frame, AIF Incorporated assembled nuclear industry comments and transmitted to the AEC revised wording of the 1967 Draft General Design Criteria along with a description of the changes. It was the AIF version of the 1967 General Design Criteria which formed the bases of the Indian Point 3 design and are discussed in this section. The AEC subsequently revised the 1967 version of the General Design Criteria and incorporated them into 10 CFR 50, Appendix A in 1971.

The Authority has completed a study of compliance with 10 CFR Parts 20 and 50 in accordance with some of the provisions of the Commission’s Confirmatory Order of February 11, 1980. The detailed results of the evaluation of compliance of Indian Point 3 with the General Design Criteria presently established by the Nuclear Regulatory Commission (NRC) in 10 CFR 50 Appendix A, were submitted to NRC on August 11, 1980, and approved by the Commission on January 19, 1982. These results are presented in Section 1.3.

Emergency Core Cooling System Capability

Criterion 44: 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.

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 system assures that the core will remain intact and in place with its essential heat transfer geometry preserved following a rupture in the Reactor Coolant System. It also assures that the extent of metal-water reaction is limited such that the amount of hydrogen generated from this source in combination with that from other sources, is tolerable in the Containment.

This capability is provided during the simultaneous occurrence of a Design Basis Earthquake. 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) or loss of coolant associated with the rod ejection accident,
3) or steam generator tube rupture.

The primary function of the emergency Core Cooling System (ECCS) for the ruptures described is to remove the stored and fission product decay heat from the core such that fuel damage to the extent that would impair effective cooling of the core is prevented. This implies that the core remain intact and in place with its essential heat transfer geometry preserved. To assure effective cooling of the core, limits on peak clad temperature and local metal-water reaction will not be exceeded. It has been demonstrated in the Westinghouse Rod Burst Program that for conditions within the area of safe operation, fuel rod integrity is maintained.

To limit the production of hydrogen in the Containment, the overall metal water reaction is limited to 1%.

In evaluating ECCS performance, consideration was given to core geometry distortion caused by swelling or fuel rod bursting.

For any rupture of a steam pipe and the associated uncontrolled heat removal from the core, the Safety Injection System (SIS) 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 intact.

Redundancy and segregation of instrumentation and components are incorporated to assure 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.

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.

Inspection of Emergency Core Cooling System

Criterion 45: Design provisions shall, where practical, be made to facilitate inspection of all physical parts of the Emergency Core Cooling System, including reactor vessel internals and water injection nozzles.

Design provisions are made to the extent practical in order 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 non-destructive test inspection where such techniques are desirable and appropriate as detailed in Section 6.2.5.

Testing of Emergency Core Cooling System Components

Criterion 46: Design provisions shall be made so that components of the Emergency Core Cooling System can be tested periodically for operability and functional performance.

The design provides for periodic testing or active components of the Safety Injection System for operability and functional performance as detailed in Section 6.2.5.

Power sources are arranged to permit individual actuation of each active component of the Safety Injection System.

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 and can be tested periodically. All remote operated valves can be exercised and actuation circuits can be tested during routine plant maintenance.

Testing of Emergency Core Cooling System

Criterion 47: 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.

An integrated system test is 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 initiation of safety injection.

Level and pressure instrumentation are 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.

The accumulators and the safety injection piping up to the final isolation valve are maintained sufficiently full of borated water at boron concentrations consistent with the accident analysis while the plant is in operation to ensure the systems remain operable and perform properly. The accumulators and injection lines are refilled with borated water as required by using the safety injection pumps to recirculate refueling water through the injection headers. A small bypass line and a return line are provided for this purpose.

Flow in each of the high head injection branch lines and in the main flow line for the residual heat removal pumps is monitored by a flow indicator.

Pressure instrumentation is also provided for the main flow paths of the high head and residual heat removal pumps.

Testing of Operational Sequence of Emergency Core Cooling System

Criterion 48: 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.

The design provides for capability to test, 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, Tests and Inspections.

Engineered Safety Features

The Engineered Safety Features are discussed in detail herein. In each section of this Chapter 6 a separate safety feature is described and evaluated. In the evaluation section for each Engineered Safety Feature, a single failure table is provided which lists the components of that safety feature system and the interconnected auxiliary systems that must function for the proper operation of that Engineered Safety Feature. An examination of these tables shows that some components of the Residual Heat Removal System, Component Cooling System, and Service Water System are necessary for proper operation of the Engineered Safety Features. These systems and their components are discussed in Section 9.3 and 9.6. The instrumentation associated with these systems is also discussed in those sections. As the auxiliary system components outside the Containment, as well as those inside the Containment, their instrumentation and power systems are not required for actuation of the Engineered Safety features; neither IEEE-279 nor the General Design Criteria apply.

Codes and Classifications

Table 6.2.1 tabulates the codes and standards to which the Safety Injection System components were designed.

Service Life

All portions of the system located within the Containment were designed to operate without benefit of maintenance and without loss of functional performance for the duration of time the component is required.

6.2.2 System Design and Operation

System Description

Adequate emergency core cooling following a Loss-of-Coolant Accident is provided by the Safety Injection System as shown in Plant Drawings 9321-F-27353 and -27503 [Formerly Figures 6.2-1A & 6.2-1B]. The system components operate in the following possible modes:

1) Injection of borated water by the passive accumulators.

2) Injection of borated water from the Refueling Water Storage Tank with the safety injection pumps. (NOTE: Technical Specification Amendment 139 eliminates the requirement to maintain a boron injection 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 as described in Section 6.2.3.)

The initiation signal for core cooling by the safety injection pumps and the residual heat removal pumps is the safety injection signal which is actuated by any of the following:

Low pressurizer pressure (2/3)
High containment pressure (2/3, High Pressure)
High differential pressure between any other two steam generators (2/3)
After time delay (maximum of 6 seconds): high steam flow in any two of the four steam lines (1/2 per line) coincident with low T_{avg} (2/4) or low steam pressure (2/4)
Manual Actuation
High-High containment pressure (two sets of 2/3, High-High pressure) [energize to actuate]

In the Technical Specifications, limits are set on minimum number of operable channels and required plant status for all reactor protection and ESF instrumentation.

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 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 below the N_2 cover gas operating pressure (approximately 650 psig), thus rapidly assuring 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 starts the safety injection and residual heat removal pumps and opens the Safety Injection System isolation valves (certain valves have their motor leads disconnected and are locked open). The valves on Plant Drawings 9321-F-27353 and -27503 [Formerly Figures 6.2-1A & -B] marked with a “S” receive the safety injection signal.

Separate and independent key-lock switches one for each SI train are provided in series to each of the auto SI actuation relays to allow manual blocking of the automatic Engineered Safeguards System actuation when the unit is in cold shutdown.

The operation of the key-lock switches into the “defeat” position will activate the existing separate annunciation for each train (Safeguard Train “A” in test and Safeguard Train “B” in test) and separate status lights (one for each train) in the Control Room. While the operator can deactivate the alarm, the individual status lights and the alarm windows will stay lit as long as the key-lock switches are in the defeat position.

The considerations involved insure that:

1) The operation of the key-lock switch to defeat the auto SI is normally carried out during the plant conditions which do not require the actuation of auto SI.

   The key-lock switch will be used only during normal plant operation, with the plant in the cold shutdown condition. The Technical Specifications do not require the operability of the SI system or any of its components during the cold shutdown conditions.

2) The operation of the key-lock switch to defeat auto SI is also permitted following an SI activation if the normal method or resetting SI is unavailable. This action is required to restore control of plant equipment to the operators.

3) Annunciation devices are provided to augment the administrative procedures.

   The operation of the key-lock switch will activate the individual annunciations and individual status lights. During the time auto SI is in the “defeat” position, the “alarm windows” and the status lights will stay lit.

The safety injection pumps (high head) deliver borated water to two separate discharge headers. The flow from each header can be injected into each of the three available cold legs (one of four cold leg lines per header has been permanently isolated by locking closed valves SI-856A on 2” Line #56 and SI-856F on 1-½” Line #754, as evaluated in Reference 2) and one hot leg of the Reactor Coolant System. Isolation valves in each of the three available cold leg injection lines are open and valves in the 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.

One high head injection header contains the retired-in-place Boron Injection Tank (BIT), which formerly contained concentrated boric acid for rapid insertion of negative reactivity in the safety injection mode. A modification replaced the contents of the BIT with water from the Refueling Water Storage Tank (RWST), (Reference 3).

> NOTE: Technical Specification Amendment 139 eliminates the requirement to maintain a BIT.

However, the BIT is an in-line, passive component of the Safety Injection System, therefore it is only “functionally eliminated,” but not physically. Furthermore, in the event of a safety injection scenario, the BIT will continue to “function” in a passive mode, to convey refueling water to the reactor core.

No credit is taken for any boron concentration in the BIT, or in any Safety Injection piping downstream of the RWST.

The BIT inlet and outlet isolation valves, (two pairs of motor operated valves, each pair arranged in parallel), are maintained in the open position, as their function to isolate the BIT is not required since implementation of the BIT elimination modification (References 2 through 5). Maintaining the BIT isolation valves open provides the benefits of eliminating an active safety function and potentially minimizing the time delay in delivery of safety injection flow. The BIT isolation valves may be individually closed for testing (one of each pair at a time) during normal power operation. Only one inlet isolation valve (SI-1852A or B) and one outlet isolation valve (SI-1835A or B) must be open to achieve the emergency core cooling safety function.

However, closing a single BIT isolation MOV presents the potential for loss of the function of the BIT header should a coincident spurious or inadvertent closure of the parallel BIT isolation MOV occur. Spurious or inadvertent mis-positioning of MOVs are considered to be credible single failures. When configured with both parallel BIT inlet or outlet MOVs closed simultaneously, the motor actuators' calculated capabilities lack the opening margin required by the GL 89-10 program. Therefore, in accordance with the IP3 GL 89-10 program, if any of the BIT isolation valves are to be closed in support of maintenance or testing, then the potential for loss of BIT header function via closure of the parallel valve (as caused by any reason including single failure) must be eliminated by administrative means.

A Safety Injection Signal still generates a signal to open the BIT isolation valves. However, based on the limited margin available in the capabilities of the motor actuators of these valves, opening in response to an SI signal would require modification in order to meet the margins required under the GL 89-10 program. Such modifications are unnecessary provided the normal position of the BIT isolation valves is open.

While refueling water has relatively low boron concentration (nominally 2,500 ppm), analyses performed to support implementation of the modification, assumed zero boron concentration in the BIT and associated piping, for conservatism. The Westinghouse "Revised Feasibility Report for BIT Elimination for Indian Point Unit 3," (July 1988) determined that the concentration of boron in the BIT may be reduced to that in the RWST while continuing to meet applicable safety criteria.

The high-head safety injection system is configured as follows:

1. The three available cold leg (one of the four cold leg lines has been permanently isolated by locking closed valve SI-856F on 1-½" Line #754) injection lines on the discharge header containing the retired-in-place BIT are physically connected to the reactor coolant pressure boundary.

2. The three available cold leg (one of the four cold leg lines has been permanently isolated by locking closed valve SI-856A on 2" Line #56) injection lines on the Non-BIT discharge header are physically connected to the accumulator discharge lines upstream of the reactor coolant pressure boundary.

3. The two hot leg injection lines on both discharge headers are physically connected to the reactor coolant pressure boundary.

This configuration was implemented in a modification installed during the 3R13 Refueling Outage to accommodate Stretch Power Uprate to provide for Hot Leg Switchover (HLSO) prior to 6.5 hours following a LBLOCA, and to resolve sump particle and ECCS valve erosion concerns identified in NRC Information Notice 96-27 and 97-76.

Since a small break in the reactor coolant pressure can include a cold leg injection line, safety injection flow capability can be limited by the resulting flow from only five available intact cold leg (Note: one of the original four cold leg lines per header has been isolated by a locked closed valve) 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 six available cold leg (Note: one of the original four cold leg lines per header has been isolated by a locked closed valve) 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 seconds for the largest design break) and a high flow rate is required to quickly recover the exposed fuel rods and limit possible core damage. 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 pumps are available in order to provide for an active component failure. Delivery from these pumps supplements the accumulator discharge. Since the Reactor Coolant System back pressure is relatively low (rapid depressurization for large breaks), a broken injection line would not appreciably change the flows in the other injection lines delivering to the core.

The residual heat removal pumps take suction from the refueling water storage tank.

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 sensors are provided in the containment sump which also give backup indication when injection can be terminated and recirculation initiated.

<u>Recirculation Phases</u>

After the injection operation, coolant spilled from the break and water collected from the containment spray is cooled and returned to the Reactor Coolant System by the recirculation system.

Following a Loss-of-Coolant Accident (LOCA), sampling is accomplished as necessary from outside of the Containment via the sampling connection from the recirculating pump discharge.

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. Water is delivered from the Containment to the residual heat removal pumps from a separate sump inside the Containment.

Although the residual heat removal pump is an acceptable alternative for providing core cooling and containment spray flow in lieu of the recirculation pump, there is no single failure that would require its use. The residual heat removal pump(s) would be used only in scenarios beyond the design basis involving multiple active failures. Use of a residual heat removal pump during the long-term recovery phase could be required in the event of ECCS leakage outside Containment.

The motor operated valves in the recirculation suction lines from the containment sump are maintained in the normally closed position at all times, however, they could be opened to allow for residual heat removal pump recirculation operation if that mode was required.

The valves are exercised in accordance with Technical Specification requirements. The valves are operated one at a time and each valve is returned to its normal position before exercising the next one.

No automatic opening features are provided; hence, the probability of a spurious signal to open the valves is nil. The only time these valves are opened is for periodic testing and the procedure ensures that both valves are closed immediately after the test. In addition, the two valves are provided in series to protect against the inadvertent opening of one valve.

The procedure used for periodic testing of these valves ensures that the only water which would be drained from these lines is the small amount trapped between the two valves. This water will discharge to the containment sump. The sump contains two sump pumps which operate on level control and will periodically pump the sump contents to the waste holdup tank during normal plant operation.

For small breaks the depressurization of the Reactor Coolant System is augmented by steam dump and auxiliary feed water addition to the Steam System. For the small breaks in the Reactor Coolant System where recirculated water must be injected against higher pressures for long term core cooling, the system is arranged to deliver the water from the residual heat exchangers to the high-head safety injection pump suction and, by this external recirculation route, to the reactor coolant loops. 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.

The recirculation pumps, the residual 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 recirculation related 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.

Various flow barriers are installed in the Vapor Containment to channel the recirculation flow into the Reactor Cavity Sump area, up and out of the Incore Instrumentation Tunnel, through the Crane Wall and VC Sump labyrinth wall via specially designed openings, and into the annulus area outside the Crane Wall. The recirculation flow will migrate towards the Recirculation Sump or the Containment Sump depending on which pump(s) are operating. Flow channeling barriers are installed on the Reactor Cavity Sump platform el. 29’ -4” around the Incore Instrumentation Tunnel, on the Recirculation Sump trenches, and at the Containment Sump. Flow channeling barriers 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 directly above the Recirculation Sump cubicle on the RHR Heat Exchanger 66’ -0” platform to preclude debris from washing through the existing grating and directly into the Sump area. Forcing the recirculation flow path through the Reactor Cavity Sump area (which is a low velocity zone) induces larger debris to settle out and minimizes its transport to the Sumps.

The Recirculation and Containment Sumps 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 modules are of four different lengths to satisfy physical configuration constraints of each Sump. The perforated plate possesses 3/32” diameter holes sized to limit downstream effects. Each module has four (4) layers of perforated surface for straining debris from the sump fluid. These layers are essentially concentric perforated metal tubes of decreasing diameter arranged in two (2) pairs. The modules feature an internal vortex suppressor which prevents air ingestion into the piping system. Furthermore, stainless steel mesh has been installed between each pair of perforated plate tubes to minimize fibrous debris bypass through the strainer. The top-hat modules are attached to strainer water boxes.

The Recirculation Sump relies on two (2) 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 an effective surface area of approximately 3156 square feet and an effective interstitial volume of approximately 471 cubic feet. Water will enter the top-hat modules through the cylindrical 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 modules, water will flow into the strainer water boxes, then over the Sump weir wall, into the Sump pump bay towards the Recirculation Pumps. The water approach velocity to the Recirculation Sump is less than one foot per second.

The Containment Sump relies on a water box with 51 top-hat modules of varying size located in the Sump pit for the purpose of preventing particles greater than 3/32” in diameter from entering the line from the Sump to the RHR Pump suction. The Containment Sump strainer has an effective area of approximately 1058 square feet and an effective interstitial volume of approximately 134 cubic feet. Water will enter the top-hat modules through the cylindrical 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 modules, water will flow into the strainer water box, which is directly connected to the RHR Pump suction line. The water approach velocity to the Containment Sump is less than one foot per second.

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 is contained within a concentric guard pipe which is connected to the containment liner and terminates within a leak tight compartment. This sump line has two remote motor operated normally closed valves for containment isolation purposes, one of which is within this leak tight compartment.

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 the residual heat removal pumps if backup capability is 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 located outside of the Containment which are designed to circulate under post-accident conditions radioactively contaminated water collected in the Containment meet the following requirements:

- Shielding to maintain radiation levels within the guidelines set forth in 10 CFR 100
- Collection of discharges from pressure relieving devices into closed systems
- Means to detect and control radioactivity leakage into the environs to the limits consistent with guidelines set forth in 10 CFR 100.

This criterion is met by minimizing leakage from the system. External recirculation loop leakage is discussed in Section 6.2.3.

One recirculation pump and one residual heat exchanger of the recirculation system provides sufficient cooled recirculated water to keep the core flooded with water by injection through the cold leg connections while simultaneously providing, if required, sufficient containment spray flow to prevent the containment pressure from rising above design limits because of the boil-off from the core. These systems are kept sufficiently filled with water to ensure the systems remain operable and performs properly. Only one pump and one heat exchanger are required to operate for this capability at the earliest time recirculation is initiated. 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. This design ensures that heat removal from the core and Containment is effective in the event of a pipe or valve body rupture.

<u>Cooling Water</u>

The Service Water System (Section 9.6.1) provides cooling water to the component cooling loop, which in turn, cools the residual heat exchangers, all of which are part of the Auxiliary Cooling Systems (Section 9.3). Three conventional 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. With the component cooling water system in long term recirculation mode, the following components are required in order to meet core cooling requirements, one residual heat removal pump and heat exchanger, one component cooling water pump, one component cooling water heat exchanger, one service water pump on the nonessential header, and two essential service water pumps on the essential header. All of this equipment with the exception of the residual heat exchangers is located outside Containment.

<u>Containment Building Water Level Monitoring</u>

Continuous indication of containment water level during and after an accident is provided by three systems with redundant measuring loops distributed as follows:

- Containment Sump (El. 38' 3"), narrow range, 0' to 10' of water.
- Recirculation Sump (El, 34' 0"), narrow range, 0' to 14' of water.
- Containment Building (El, 46' 0"), wide range, 0' to 8' of water.

Each loop consists of a sensor and a transmitter located inside the containment building, a recorder and power supply at the control room. Refer to Plant Drawing 9321-F-27353 [Formerly Figure No. 6.2-1A].

Change-Over 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 for the changeover from injection to recirculation in the case of a large rupture beginning form the time of the safety injection signal:

In approximately ten minutes, sufficient water has been delivered to provide the required NPSH to start the recirculation pumps.

In approximately 15 to 20 minutes, (1) one of two low level alarms on the RWST sounds, and the redundant containment recirculation sump level indicators show the sump water level. The alarm serves to alert the operator to start the switchover to the recirculation mode. The redundant containment recirculation sump level indicators provide verification the RWST water has been delivered during the injection phase, in addition to providing consideration to the case of a spurious (i.e., early) RWST low level alarm. The operator would see on the control board that the redundant recirculation sump level indications are at the appropriate points; switch-over to the recirculation phase of safety injection is performed at this time.

With the initiation of the switch sequence (e.g., Switch No. 1), only one spray pump will continue in operation. This spray pump will continue to draw from the RWST for approximately 25 minutes. [Deleted]

Recirculation pump motors are 2'-2" above the highest water level after addition of the injected water to the spilled coolant.

The entire switchover from injection to recirculation phase is carried out by manually initiating equipment starts/stops and closing a series of switches (each of which carries out several operations) located in the Control Room. At only two points in the switchover routine is any reliance on instrumentation necessary:

1) When the low level alarm occurs from the refueling water storage tank low level (LT-920 and/or LIC-921 outside containment), the operator is alerted to begin closing the series of recirculation switches. The low level alarm for each instrument is set to actuate when there is between 10.5 feet and 12.5 feet of water in the tank.

2) After closing switch 4, the operator is required to make a decision whether to close switch 5 or switch 6. The basis for this decision is the flow reading on the flow meters FT-946A, 946B, 946C and 946D. If three or more of these flow meters each indicate greater than zero and the lowest of these readings is at least 360 gpm, $\pm$ 10 gpm, the operator will close switch 6; otherwise, the operator will close switch 5.

Analysis indicates that approximately 662 gpm to the core is required to match boil-off at 1398 seconds (the earliest time at which recirculation could be initiated). This includes a 20% penalty to allow for the effects of hot metal quenching. The flow rates that follow ensure an actual flow of ≥662 gpm. Accordingly, a requirement of 360 gpm, ± 10 gpm minimum flow rate on the lowest indicating loop has been specified to account for uncertainties in flow measurement and to provide margin.

The decision making process with regard to the flow to the Reactor Coolant System via the low head injection lines is based on readings of the four injection line flowmeters. The rationale for this basis is the following:

For four flow meters each reading greater than zero (i.e., none indicating zero flow):

1) Assume one flow meter fails to an inaccurate high reading, (as a result of a single failure); if the flow rate reads greater than 360 gpm, ± 10 gpm then this meter is not used as a basis for the 360 gpm, ± 10 gpm setpoint, but rather, the lowest indicating meter (greater than 0 gpm) is used.

2) Of the four injecting lines, the highest flow line is assumed to be connected to the spilling line; therefore, flow from this line is ineffective.

3) For the three remaining intact lines, their total flow of 751 gpm is delivered to core; if at least 360 gpm, ±10 gpm is indicated by the lowest line, total flow requirement of 662 gpm is satisfied using low head recirculation.

For one (or two) flow meters each reading greater than zero (i.e., two indicating zero flow):

4) Assume failure of one or two flow meters due to loss of common power supply (i.e., single failure). If the flow rate reads greater than 360 gpm, ± 10 gpm then this meter is not used as a basis for the 360 gpm, ± 10 gpm setpoint, but rather, the lowest indicating meter (greater than 0 gpm) is used.

5) Of the four injecting lines, the highest flow line is assumed to be connected to the spilling line; therefore, flow from this line is ineffective.

6) For the three remaining intact lines, their total flow of 751 gpm is delivered to the core; if at least 360 gpm, ± 10 gpm is indicated by the lowest line; total flow requirement of 662 gpm is satisfied using low head recirculation.

Thus, the entire recirculation switchover sequence requires only the aforementioned instrumentation. Manual switchover, that is, without use of the “eight-switch sequence,” does not require any additional instrumentation; it requires only that the operator close a switch for each and every operation in the switchover sequence.

The control circuitry for the associated valves and pumps is located external to the Containment. The motor operators, associated power cables, and instrumentation inside the Containment have been designed to withstand the LOCA environment as stated in Appendix 6F.

Several motor operated valves operated during the transfer to cold leg or hot leg recirculation are maintained de-energized in their safeguards position during normal power operation in accordance with the Technical Specifications. The subject valves are 856B, 856G, 1810, 882, 744, 842, 843, 883, 1870, 743, 894A, 894B, 894C, and 894D. Each valve (except for the submerged 894C) may be energized during or following the transfer to recirculation at a point in time when spurious or inadvertent mispositioning would not defeat a safety function relied upon to mitigate the consequences of the event.

The manual switchover by the operator which accomplishes the changeover from injection to recirculation is listed below. This switchover takes place when the level indicator or level alarms on the refueling water storage tank indicates that the fluid has been injected. The level indicators in the containment sump will verify that the level is sufficient within the Containment. The sequence is followed regardless of which power supply is available. The time required to complete the switchover to recirculation is the time for the switch gear to function. All the recirculation switches are grouped together on the safeguard control panel. The service water pumps and component cooling water pumps are located on the auxiliary coolant 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 back-up component.

The following sequence maintains the loads on the 480V buses within analyzed limits, maintains sufficient core cooling flow during and following the transfer to recirculation and ensures that components are operated within their analyzed limits. While this sequence does not attempt to mirror each procedural step, the major steps listed below must be performed in the sequence described:

1) Terminate safety injection actuation signal and containment spray actuation signal in order that the control logic permits manipulation of the system (at any time following completion of the auto start sequence)

2) Close switch one (remove and isolate unnecessary loads from the diesels).

   Trips high head safety injection pump No. 32 if all three are operating (no action if two are operating). Isolates pump No. 32 from the Refueling Water Storage Tank.

   Trips spray pump No. 32 if both are operating (no action if only one is operating). Closes isolation valve at the inoperative spray pump discharge.

3) Close switch three (remove and isolate unnecessary loads from the diesels). Trips both residual heat removal pumps.

   Closes isolation valves at pump suction and discharge headers (the Technical Specifications require the motor operators for these valves to be de-energized).

4) Secure electric auxiliary feedwater pumps prior to closing switch two.
   a) If continued Auxiliary Feedwater flow is required, the turbine driven pump is used.

   If continued, Auxiliary Feedwater flow is required and the turbine driven pump is unavailable, only one motor driven Auxiliary Feedwater pump may be run.

5) Close switch two (establish cooling flow for Residual Heat Exchangers)
   a) Starts on one non-essential service water pump (the second or third pump is given a start signal if the first or second pump fails to start).

b) Starts one component cooling water pump (the second or third pump is given a start signal if the first or second pump fails to start).

6) Manually initiate internal recirculation flow.

a) Manually start recirculation Pump A (if Pump A fails to start, use manual start for Pump B; Pump B control switch is adjacent to switch four).

b) Close switch four to open valves on discharge of recirculation pumps. Starting a Recirculation Pump prior to closing switch four minimizes the potential pressure differential across these motor operated valves.

c) Valves SI-HCV-638 and / or SI-HCV-640 are throttled to maintain recirculation flow. For one pump operation, throttling is required to maintain recirculation pump flow within maximum pump flow limits.

7) Check Flow to Reactor Coolant System via the low head injection lines.

a) For the preferred operating mode of omitting switch five and closing switch six (i.e., provides recirculation at low system pressure), the following flow conditions must be verified:
   1) With flow in three or more lines greater than zero, the lowest of these flows is at least 360 gpm, $\pm$ 10 gpm.

b) If the above flow conditions are met, the following actions are taken:
   1) Direct operators in the field to throttle service water valves SWN-35-1 and 35-2 to maintain CCW temperature within prescribed limits.
   2) Close switch six, which trips operating safety injection pumps.

c) If the above flow conditions are not verified, close switch five and omit switch six (provides recirculation at elevated system pressure).

   1) Aligns flow from residual heat exchanger to high head safety injection pumps. (The motor-operated valves on the outlet of the residual heat exchangers to the suction of the high-head safety injection pumps are opened. The motor-operated valves on the outlets of the residual heat exchangers to the low-head injection lines are closed together with the safety injection pump mini-flow and residual heat removal pump mini-flow).

   2) Direct operators in the field to throttle service water valves SWN-35-1 and 35-2 to maintain CCW temperature within prescribed limits.

8) Close switch seven
- Starts a second non-essential service water pump.
- Starts a second component cooling water pump only if the second service water pump successfully started.
- Starts a second recirculation pump only if the second component cooling water pump successfully started. Note: Running two 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.

9) Close switch eight (complete the isolation of the safety injection system and containment spray system test lines to the refueling water storage tank).

   a) Close the valve on the spray test line.

   b) Close the valve in the safety injection pumps suction line from the Refueling Water Storage Tank (control power for this valve is de-energized as required by the Technical Specifications).

If an RHR pump is used for low head recirculation, then valves SI-HVC-638 and/or SI-HCV-640 are throttled to maintain RHR pump flow to the cold legs (and recirculation spray) less than the maximum pump flow limit. Section 9.6.1 describes additional requirements to be met relating to alignment and operation of the service water system at the beginning of the post- LOCA recirculation phase.

Although the listed recirculation 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 and when he should proceed with the next switching operation. In addition, lamps indicating completion of the individual functions for a given switch are provided. These lamps are adjacent to the switches. The time required to complete the switchover is just the time for the switch gear to operate. 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 (Table 6.2-11) which are under manual control (that is, 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. Reference is made to Table 6.2-11 which is a listing of the instrumentation readouts on the control board which the operator can monitor during recirculation. In addition, an audible annunciation alerts the operator to the condition.

Hot leg recirculation is initiated after 4 hours but prior to 6.5 hours.

<u>Location of the Major Components Required for Recirculation</u>

The residual heat removal pumps are located in the residual heat removal pump of the Primary Auxiliary Building (El. 15'). The residual heat exchangers are located on a platform above the basement floor of the Containment Building (El. 66').

The recirculation pumps are located directly above the recirculation sump in the Containment Building (El. 46').

The component cooling pumps and heat exchangers are located in the Primary Auxiliary Building (El. 41' and 73', respectively).

The service water pumps are located in the intake structure and the redundant piping to the component cooling heat exchangers is run underground.

Steam Break Protection
A large break of a steam system pipe rapidly cools the reactor coolant causing insertion of reactivity into the core and depressurization of the system. Compensation is provided by injection of borated water from the refueling water storage tank (RWST). Redundant isolation valves open upon a safety injection signal, providing a supply of borated water with a boron concentration of 2500 ppm nominally. Even assuming all of the safety injection lines downstream of the RWST, including the BIT, contain unborated water, this is sufficient to terminate the reactor power transient before any clad damage results. The analysis of the steam line rupture accident is presented in Section 14.2.5.

Components

All associated components, piping, structures, and power supplies, of the Safety Injection System were designed in accordance with the seismic criteria provided in Section 16.1.1 and were predominately designated as seismic Class 1. Refer to Plant Drawing 9321-F-27353 and -27503 [Formerly Figures 6.2-1A and 1B] for indication of the seismic class piping boundaries.

All components inside the Containment are capable of withstanding or are protected from differential pressure which may occur during the rapid pressure rise to 47 psig in 10 seconds.

Emergency core cooling components are austenitic stainless steel, and hence, are quite compatible with the spray solution over the full range of exposure in the post-accident regime. [Deleted] Corrosion tests performed with simulated spray (original NaOH buffer additive) showed negligible attack, both generally and locally, in stressed and unstressed stainless steel at containment and ECCS conditions. These tests are discussed in WCAP-7153[(1).] Corrosion tests performed with several buffering agents showed corrosion to submerged aluminum is higher in the presence of sodium tetraborate compared to NaOH but was not excessive, and corrosion to submerged steel in sodium tetraborate was comparable to NaOH. These tests are discussed in WCAP-16596-NP.

The quality standards of all Safety Injection System components are tabulated in summary form in Table 6.2-12.

Accumulators

The accumulators are pressure vessels filled with borated water and pressurized with nitrogen gas. During normal plant operation each accumulator is isolated from the Reactor Coolant System by two check valves in series. Should the Reactor Coolant System pressure fall below the accumulator operating pressure, the check valves open and borated water is forced into the Reactor Coolant System. Mechanical operation of the swing-disc check valves is the only action required to open the injection paths from the accumulators to the core via each cold leg.

Indian Point 3 does not utilize hot leg injection. No timer is involved or provided. The two hot leg connections are provided to allow hot leg recirculation. However, these connections are closed at all times during plant operation.

The level of borated water in each accumulator tank is adjusted remotely as required during normal plant operation. During normal plant operation, the fluid level can be reduced by draining through the Sampling System to the Sample Sink in the PAB. The water level can also be reduced by draining to the reactor coolant drain tank or to the VC sump; however, these drain paths degrade the accumulator function by exposing the affected accumulator(s) to non-seismic piping from which it can not be isolated in accordance with design criteria. Draining accumulators to the Reactor Coolant Drain tank or the VC sump may only be performed under the conditions delineated by the plant Technical Specifications. To increase and/or maintain the accumulator water level, refueling water is added using a safety injection pump. Samples of the solution in the tanks are taken at the sampling station for periodic checks of boron concentration.

The accumulators are passive Engineered Safety Features because the gas forces injection; no external source of power or signal transmission is needed to obtain fast-acting, high flow capability when the need arises. On accumulator is attached to each of the 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 water after the end of blowdown, to reflood the core. (Section 14.3)

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 indictors are provided with readouts on the control board. Each indicator is equipped with high and low level alarms.

For the Accumulator Discharge Valves (894 A, B, C, D), the following indications are provided to supervise the administrative procedures and to highlight the existence of an incorrect configuration:

1) Red (open) and Green (closed) position indicating lights at the control switch for each valve. These lights are powered by valve control power and actuated by valve motor operator limit switches.

2) An additional indicating system of lights is used, whereby each valve has a two light sugar cube. The right side of the cube is a WHITE light (which glows PINK if the adjacent RED light is lit) that indicates power applied to the indicating system; the left side of the cube is a RED light to indicate when the respective valve is in its proper position enabling safeguards operation. This grouping highlights a valve not properly lined up. These lights are energized from a separate monitor light supply and the RED light is actuated by a valve motor operator limit switch.

3) In the event a valve is closed for accumulator or valve testing at the time injection is required, a safety injection signal is applied to open the valve (if power is available), overriding the test closure.

Prior to commercial operation, the AEC required that the electric power to these valves be locked out to prevent a spurious closure. The lockout will be implemented whenever RCS temperature is above 350°F. These valves are closed during plant shutdown conditions to isolate the pressurized accumulators from the depressurized reactor coolant system.

The accumulator design parameters are given in Table 6.2-2.

Boron Injection Tank

The Boron Injection Tank (BIT) is “functionally” retired-in-place. That is, it is no longer relied upon to provide concentrated boric acid for injection into the reactor core during emergency core cooling. It does, however, remain a passive component of the Safety Injection System, and as such, it is relied upon for its properties as a pressure vessel. Because the BIT no longer contains concentrated boric acid, the specialized handling requirements associated with that substance, such as heating and recirculation, no longer need to be met. The heaters which reside at the bottom of the BIT have been permanently de-energized. Furthermore, the recirculation flowpath between the BIT and the Boric Acid Storage Tanks has been valved off.

The BIT inlet and outlet isolation valves (two pairs of motor operated valves, each pair arranged in parallel) are maintained in the open position, as their function to isolate the BIT is not required since implementation of the Reference 3 modification. A Safety Injection Signal still generates a signal to open the BIT isolation valves. However, based on the limited margin available in the capabilities of the motor actuators of these valves, opening in response to a SI signal would require modification in order to meet the margins required under the GL 89-10 program. Such modifications are unnecessary provided the normal position of the BIT isolation valves is open.

The BIT remains in the safety injection flowpath, and continues to be relied upon to convey the water contained in it, as well as water from the Refueling Water Storage Tank, in the same manner as a section of piping would. Although the BIT contents are identical to the contents of the RWST, that is, borated water with a nominal boron concentration of 2,500 ppm, no credit is taken by the core response analyses for any boron in the BIT, or the safety injection piping downstream of the RWST, for conservatism.

The design parameters of the BIT are presented in Table 6.2-3.

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 system remains operable and performs properly. During plant operation, it is aligned to these pumps.

The capacity of the refueling water storage tank is based on the requirement for filling the refueling canal. When filled to Technical Specification requirements, approximately 342,200 gallons is available for delivery. One low level alarm is set to actuate at between 10.5 feet and 12.5 feet of water in the tank. This tank capacity and these alarm settings provide an amount of borated water to assure:

1) A sufficient volume of water on the floor to permit the initiation of recirculation (195,800 gal).

2) A volume sufficient to allow switchover to recirculation pumps, containment pressure relief, and sump pH control via containment spray system following a reactor coolant pressure boundary break (66,700 gal).

3) Adequate volume to allow for instrument uncertainties (total 52,100 gallons between the Technical Specifications minimum RWST level of 35.4’ and the nominal

containment spray shutoff point of 1.5'. Of this volume, 26,100 gallons are eventually added to the Containment, but the remaining 26,000 gallons are considered by the analysis to be unusable.

4) The total RWST volume, when added with accumulator discharge to the reactor coolant system, will assure 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 a concentration which assures reactor shutdown by at least 5% ∈k/k when all RCC assemblies are inserted and when the reactor is cooled down for refueling. The maximum boric acid concentration is approximately 1.5 weight percent boric acid. At 32°F the solubility limit of boric acid is 2.2%. Therefore, the concentration of boric acid in the refueling water storage tank is well below the solubility limit at 32°F.

The contents of the Refueling Water Storage Tank are kept above 32°F by a steam heated, austenitic stainless steel pipe coil in the bottom of the tank. Steam is supplied to this coil through a single header from the auxiliary boilers which are used to supply all required auxiliary steam to Indian Point 3.

The passive heating coil and passive single supply header are supplied with steam from any one of five sources. In the remote case of loss of steam to this tank, there would be a time period of at least 24 hours available for repair or connection to another steam source before freezing problems would arise, even under the most severe weather conditions. If the electrical heat tracing on the tank discharge line remains operable it is very probable that a freezing problem would not arise.

The steam to the heating coil is automatically flow controlled to maintain a minimum tank water temperature of 35°F. In response to low RWST temperature, steam is admitted by temperature control valve TCV-1116, and the pressure is controlled automatically by pressure control valve PCV-1250 to maintain a nominal 7 psig steam pressure in the coil (see Plant Drawing 9321-F-27273 [Formerly Figure 9.6-16]). During normal and shutdown operations when the tank is filled with borated water, the water pressure outside the heating coil will be approximately 15 psig, thereby preventing leakage of steam out of the coil and subsequent dilution of the borated water.

All outdoor piping connected to the Refueling Water Storage Tank is electrically heat traced. The failure of any section of heat tracing is annunciated in the Control Room. The power source for the heat tracing can be manually switched between two MCC's, each powered automatically by different emergency diesel generator buses.

The design parameters are presented in Table 6.2-4

Pumps

Class I (seismic) pumps in the Emergency Safeguards Systems, their required Net Positive Suction Head (NPSH) at extreme operating conditions, the fluid operating temperature, the NPSH available, the atmospheric pressure assumption, and the elevation of each pump are given in Table 6.2-13.

The Internal Recirculation Pumps were replaced under modification ER-04-3-066 during Refueling Outage 14. The new pumps are Flowserve model 267APKD- three (3) stage, double suction pumps. These pumps require less NPSHR over the entire flow range than the previous 24-APK3 single suction pumps. The NPSH data for the Internal Recirculation Pump is provided in Table 6.2-13. The data given for single pump operation is based on conservatively high NPSHR values that were used in the calculation, and which are greater than those provided by the latest curve (Figure 6.2-4). For an actual flow rate of 3530 GPM at the pump discharge nozzle, the NPSHR was conservatively taken to be 7.95 feet and the NPSHA is 10.54 feet.

An analysis predicts that, for the large break LOCA, there will be 10.54 ft. of NPSH available, which credits the remainder of the RWST water delivered to the containment prior to the start of recirculation containment spray (References 8, 9, 10). In the case of a small-break LOCA, when elevated RCS pressure would preclude direct low head recirculation, high head recirculation would then be established using the Recirculation Pump(s) to deliver a suction supply to the SI Pumps. During high head recirculation, the Recirculation Pumps operate at lower flow rates and the NPSH requirements are correspondingly lower.

NPSH calculations assume saturated water in the sumps so that no credit is taken for containment pressure exceeding the vapor pressure of the sump water. While this conservative assumption is appropriate at accident initiation, it does not allow any credit for the increase of NPSHA which would result from the gradual cooling of the sump fluid to below saturated conditions.

A review performed pursuant to NRC Generic Letter 85-22 had also established that the actual containment water level would be above the minimum switchover level indicated in Table 6.2-13. This actual water level would provide sufficient additional NPSH available to overcome the head loss effects of debris which has been postulated to result from the destruction of steam generator thermal insulation by LOCA jet forces.

The three (high head) safety injection pumps for supplying borated water to the Reactor Coolant System are horizontal centrifugal pumps driven by electric motors. Parts of the pump in contact with borated water are stainless steel or equivalent corrosion resistant material. 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. The bypass line joins a common miniflow line shared by the other pumps. Each safety injection pump is sized at 50% of the capacity required to meet the design criteria outlined in Section 6.2.1. The design parameters are presented in Table 6.2-5, and Figure 6.2-2 gives the performance characteristics of these pumps.

The two residual heat removal (low head) pumps of the Auxiliary Coolant System are used to inject borated water at low pressure to the Reactor Coolant System. The two recirculation pumps are used to recirculate fluid from the recirculation sump and send it back to the reactor, the spray headers or to suction of the safety injection pumps. All four of these pumps are of the vertical centrifugal type, driven by electric motors. Parts of the pumps which contact the borated water and sodium tetraborate solution during recirculation are stainless steel or equivalent corrosion resistant material. A minimum flow bypass line is provided on the discharge of the residual 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. Additionally, each residual heat removal pump is provided with a dedicated recirculation line. These recirculation lines prevent either pump from operating at shutoff conditions and, also, during dual-pump operation preclude the stronger residual heat removal pump from dead heading the weaker pump as described in IE Bulletin 88-04. A minimum flow bypass discharging back into the recirculation sump, is provided to protect the recirculation pumps should their normal flow paths be blocked. Figure 6.2-3 and 6.2-4 give the performance characteristics of these pumps. The design parameters are presented in Table 6.2-5.

The safety injection pump bearings are cooled by booster pumps using component cooling water. The booster pumps are directly connected to the injection pump motor shaft. The pump seals were designed to operate at accident conditions without cooling water. Pump data is provided in chapter 9.3.

The recirculation pump motors are enclosed fan cooled. The air is cooled by coils utilizing component cooling water and four auxiliary component cooling pumps located outside the Containment. During recirculation the sump water cools the pump bearings. The four (i.e., two pairs) auxiliary component cooling pumps are started during the injection phase; either pump of a pair is capable of protecting its recirculation pump motor from the containment atmosphere. The fans are directly connected to the recirculation pump motor shafts. The auxiliary component cooling pumps are a part of the Component Cooling Water System and pump data is provided in Chapter 9. The component cooling water volume constitutes a large heat sink so that the main component cooling pumps are not needed during the injection phase (with loss of offsite power).

Details of the component cooling pumps and service water pumps, which serve the Safety Injection System, are presented in Chapter 9.

The pressure containing parts of the high head safety injection pumps are castings, conforming to ASTM A-296, Grade CA-15. The pressure containing parts of the Residual Heat Removal Pumps and the Recirculation Pumps are castings conforming to ASTM A-296, Grade CF-8a (chromium content 21.0 to 22.5) and ASTM A-351, Grade CF3M, respectively. Stainless steel forgings were procured per ASTM A-182, Grade F304 or F316, or ASTM A-336, Class F8 or F8M, and stainless plate was constructed to ASTM A-240, type 304 or 316. All bolting material conforms to ASTM A-193. Materials such as weld-deposited Stellite or Colmonoy were used at points of close running clearances in the pumps to prevent galling and to assure continued performance ability in high velocity areas subject to erosion.

All pressure containing 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 containing 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 ANSI 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 evaluation of the shaft seal and bearing design to determine that 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 minutes.

Each pump was given a complete shop performance test in accordance with Hydraulic Institute Standards. The pumps were run at design flow and head, shut-off head and three additional points to verify performance characteristics. Where NPSH is critical, this value was established at design flow by means of adjusting suction pressure.

Pump Cooling Water Supply	
Pump	Source of Cooling Water
1. Internal Recirculation Pumps	Auxiliary Component Cooling Water pumps are used to deliver – 40 gpm to each motor cooler.
2. High Head Safety Injection Pumps	Booster pumps connected to the shafts of the SI pumps are designed to circulate – 40 gpm of CCW per pump.
3. Residual Heat Removal Pumps	Cooling water for seals is not required when the temperature of the pumped fluid is less than 150° F. This is the case during the injection phase after a LOCA. During the recirculation phase, when the pumped fluid temperature may be more than 150° F, a component cooling pump will be running to supply cooling water to the RHR pump.
4. Containment Spray Pumps	This pump pumps fluid with a temperature never in excess of 110° F. Therefore no cooling water is required.

The only period of concern when these pumps experience a lack of cooling water is during the injection phase following a LOCA, since during the recirculation phase and at all other times, the component cooling pumps will be available.

During the injection phase, the only heat removal requirement is for the high head safety injection pumps and for the internal recirculation pumps.

Safety Injection Pumps:

Component cooling water is required for cooling the bearings of these pumps. The heat load is estimated to be 75,000 Btu/hr per pump or a total of 225,000 Btu/hr for 3 pumps.

Internal Recirculation Pumps:

Cooling water is required to protect the pump motors from the containment environment during a LOCA. The heat load is estimated to be approximately 150,000 Btu/hr per pump or a total of 300,000 Btu/hr for two pumps.

Since the component cooling pumps do not run during the injection phase, (with loss of offsite power), the water volume of the component cooling system is used as a heat sink. This heat load causes a temperature rise of approximately 7°F/hour in the component cooling water (no credit is taken for the water volume in the surge tank). With 110° F cooling water at the start of the accident, 6 hours are available before the cooling water temperature reaches 150° F; 10 hours are available before reaching 180° F.

Heat Exchangers

The two residual heat removal heat exchangers of the Auxiliary Coolant System cool the recirculated sump water. These heat exchangers were sized for the cool-down of the Reactor System. Table 6.2-6 gives the design parameters of the heat exchangers.

The ASME Boiler and Pressure Vessel Code has strict rules regarding the wall thickness 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 TEMA (Tubular Exchanger Manufacturers Association) 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. The design and inspection requirements included: confined-type gaskets, main flange studs with two nuts on each end to ensure permanent leak tightness, general construction and mounting brackets suitable for the plant seismic design requirements, 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, a hydrostatic test duration of not less than thirty minutes, 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 a SA-285 Grade C carbon steel shell, a SA-234 carbon steel shell end cap, SA-213 TP-304 stainless steel tubes, a SA-240 type-304 stainless steel channel, a SA-240 type 304 stainless steel channel cover, and a SA-240 type 304 stainless tube sheet.

Valves

All parts of valves used in the Safety Injection System in contact with borated water are austenitic stainless steel or equivalent corrosion resistant material. The motor operators on the injection line isolation valves are capable of rapid operation. All valves required for initiation of safety injection or isolation of the system have remote position indication in the Control Room.

Valving is specified for exceptional tightness, and where possible, instrument valves and packless diaphragm valves are used. All valves, except those which perform a control function, are provided with backseats which are capable of limiting leakage to less than 1.0 cc per hour per inch of stem diameter, assuming no credit for valve packing. Backseats can also be employed to facilitate repacking the valve stem. As a general rule, the plant relies on packing to minimize valve stem leakage. Normally closed globe valves are installed with recirculation flow under the seat to prevent leakage of recirculated water through the valve stem packing. Relief valves are totally enclosed. Control and motor-operated valves which are 2-1/2 in and larger and which are exposed to recirculation flow are provided with double-packed stuffing boxes and stem leakoff connections which are piped to the Waste Disposal System.

The check valves which 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 in order to prevent overpressure in the lines which have a lower design pressure than the Reactor Coolant System. The relief valve is set at the design pressure of the safety injection piping.

The gas relief valves on the accumulators protect them from pressures in excess of the design value.

Motor Operated Gate Valves

The pressure containing parts (body, bonnet and discs) of the valves employed in the Safety Injection System were designed per criteria established by the ANSI B16.5 (1955) or MSS SP66 specifications. The materials of construction for these parts were procured per ASTM A182, F316 or A351, GR-CF8M or CF8. All material in contact with the primary fluid, except the packing, is austenitic stainless steel or equivalent corrosion resisting material. The pressure containing cast components were radiographically inspected as outlined in ASTM E-446 Class 1 or Class 2. The body, bonnet and discs were liquid penetrant inspected in accordance with ASME Boiler and Pressure Vessel Code Section VIII, Appendix VIII. The liquid penetrant acceptable standard was as outlined in ANSI B31.1, Case N-10.

When a gasket is employed, the body-to-bonnet joint was designed per ASME Boiler and Pressure Vessel Code Section VIII or ANSI B16.5 with a fully trapped, controlled compression, spiral wound asbestos or suitable material, 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 were procured per ASTM A193 and A194, respectively.

The entire assembled unit was hydrotested as outlined in MSS SP-61 with the exception that the test pressure was maintained for a minimum period of 30 minutes. The seating design of the Darling parallel disc 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 has to work only against the frictional component of the hydraulic unbalance on the disc and against the packing box friction. The discs are guided throughout the full disc travel to prevent shattering and 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, procured and heat treated to Westinghouse specifications. These materials were selected because of their corrosion resistance, high tensile properties, and their resistance to surface scoring by the packing. The valve stuffing box was 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. The experience with this stuffing box design and the selection of packing and stem materials has been very favorable in both conventional and nuclear power plants.

Valves 744, 882, 1810, are required to be open during the injection phase of the LOCA and then must be closed for long-term recirculation. There is no time that these valves would be closed during plant power operation. The motors for these valves are normally de-energized with their breakers locked open. In addition, these valves are provided with red/green position indicating lights and monitor lights to highlight valves configuration as described in Section 6.2.2 “Accumulators,” items 1 and 2.

Valves 856B and 856G are required to be closed during the injection and cold leg recirculation phases. The motors for these valves are normally de-energized with their breakers locked open and normal indicating lights de-energized. In addition, these valves are interlocked with corresponding cold leg injection line valves on each header to prevent simultaneous opening of all high-head safety lines on each header. The valves are equipped with a position monitor light, via limit switch and separate DC circuit, and an alarm via limit switch.

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 move the valve off its main seat or backseat while allowing the motor to attain its operational speed.

The valve was 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, were submitted to Westinghouse for approval.

For those valves which function on the safety injection signal, 10 seconds operation or other justified times are typically required. The BIT isolation valves are maintained open, but still receive a safety injection signal to open based on their former application as normally closed valves. An opening stroke time of 11 seconds had been justified for these valves when they were maintained normally closed. However, changing their position from normally closed to normally open has superseded the requirement for a 10 or 11 second opening stroke time. For all other valves in the system, the valve operator completes its cycle from one position to the other typically in 120 seconds.

Valves which must function against system pressure were typically designed such that they function with a pressure drop equal to full system pressure across the valve disc.

Manual Valves

The stainless steel manual globe, gate and check valves were designed and built in accordance with the requirements outlined in the motor operated valve description above.

The carbon steel valves were built to conform with ANSI B16.5. The materials of construction of the body, bonnet and disc conform to the requirements of ASTM A105 Grade II, A181 Grad II, or A216 Grade WCB or WCC. The carbon steel valves pass only non-radioactive fluids and were subjected to hydrostatic test as outlined in MSS SP-61 except that the test pressure was maintained for at least 30 minutes. Since the fluid controlled by the carbon steel valves is not radioactive, the double packing and seal weld provisions are not provided.

Accumulator Check Valves

The pressure containing parts of this valve assembly were 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 WNES specifications. The cast pressure-containing parts were radiographed in accordance with ASTM E-94 and with the acceptance standard as outlined in ASTM E-446 Class 1 or Class 2. The cast pressure-containing parts, machined surfaces, finished hard facings, and gasket bearing surfaces were liquid penetrant inspected per ASME B&PV Code, Section VIII and the acceptance standard as outlined in ANSI B31.1 Code Case N-10. The final valve was hydrotested per MSS SP-66 except that the test pressure was maintained for at least 30 minutes. The seat leakage was conducted in accordance with the manner prescribed in MSS SP-61.

The valve was 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 was manufactured from 17-4 pH stainless steel, heat treated to Westinghouse Specifications. The clapper arm shaft bushings were manufactured from Stellite No. 6 material. The various working parts were selected for their corrosion resistant, tensile, and bearing properties.

The disc and seat rings were manufactured from a forging. The mating surfaces are hard faced with Stellite No. 6 to improve the valve seating life. The disc is permitted to rotate, providing a new seating surface after each valve opening.

The valves are operated in the closed position with a normal differential pressure across the disc of approximately 1700 psi. The valves remain in this position except for testing and safety injection. Since the valve will not be required to normally operate in the open condition, which would subject the valve to impact loads caused by sudden flow reversal, this equipment does 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 Emergency Injection System of the Carolina-Virginia Tube Reactor in a similar system 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 six inch check valve. Check valve leakage was not a problem. This was further substantiated by the satisfactory experience obtained from operation.

Relief Valves

The accumulator relief valves were sized to pass nitrogen gas at a rate in excess of the accumulator gas fill line delivery rate. The relief valves can also pass water in excess of the expected leak rate, but this is not necessary because the time required to fill the gas space gives the operator ample opportunity to correct the situation. For an inleakage rate 15 times the manufacturing test rate, there will be about 1000 days before water will reach the relief valves. Prior to this, level and pressure alarms would have been actuated.

The safety injection test line relief valves are provided to relieve any pressure above design that might build up in the high head safety injection piping. The valve can pass a nominal 15 gpm (2.25 x 10^5 cc/hr), which is far in excess of the manufacturing design leak rate of 24 cc/hr.

Leakage Limitations of Valves

Valving was specified for exceptional tightness and, where possible, instrument valves, packless diaphragm valves were used.

Normally open valves have backseats which are capable of limiting leakage to less than one cubic centimeter per hour per inch of stem diameter assuming no credit for packing in the valve. Backseats can also be employed to facilitate repacking the valve stem. As a general rule, the plant relies on packing to minimize valve stem leakage. Normally closed globe valves were 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 were provided with double-packed stuffing boxes and stem leakoff connections which are piped to the Waste Disposal System.

The specified leakage across the valve disc required to meet the equipment specification and hydrotest requirements is as follows:

Conventional globe – 3 cc/hr/in of nominal pipe size

Gate valves – 3 cc/hr/in of nominal pipe size; 10/cc/hr/in for 300 and 150 pound USA Standard

Motor operated gate valves – 3 cc/hr/in of nominal pipe size: 10/cc/hr/in for 300 and 150 pound USA Standard

Check valves – 3 cc/hr/in of nominal pipe size: 10/cc/hr/in for 300 and 150 pound USA Standard

Accumulator check valves – 10 cc/hr/in of nominal pipe size; relief valves are totally enclosed.

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 and recirculation pumps.

The piping beyond the accumulator stop valves was designed for Reactor Coolant System conditions (2485 psig, 650°F). All other piping connected to the accumulator tanks was designed for 700 psig and 400°F.

The safety injection pump and residual heat removal pumps suction piping (210 psig at 300°F) from the refueling water storage meets NPSH requirements of the pumps.

The safety injection high pressure branch lines (1500 psig at 300°F) were 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 incorporated 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. Two SI pump discharge headers are provided in a configuration which allows 2 of 3 SI Pumps to deliver into either header. The suction flow paths are configured to allow isolation of the 32 SI Pump suction piping from the common suction flow path, with an alternate suction piping alignment dedicated to the 32 SI Pump. The common and alternate suction flow paths are cross-tied via a 0.75" pressure equalization pipe, with two normally closed valves and a normally closed vent valve.

The piping was designed to meet the minimum requirements set forth in (1) the ANSI B31.1 Code (1955) for the Pressure Piping, (2) Nuclear Code Case N-7, (3) ANSI Standards B36.10 and B36.19 and (4) ASTM Standards with supplementary standards plus additional quality control measures.

Minimum wall thickness were determined by the ANSI Code (1955) formula found in the power piping Section 1 of the ANSI Code (1955) for Pressure Piping. This minimum thickness was increased to account for the manufacturer's permissible tolerance of minus 12-1/2 percent on the nominal wall. Purchased pipe and fittings had 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 stress analyses were performed in accordance with ANSI B31.1 code (1967). Special attention was 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.

Pipe and fittings materials were procured in conformance with all requirements of the ASTM and ANSI specifications. All materials were verified for conformance to specification and documented by certification of compliance to ASTM material requirements. Specifications imposed additional quality control upon the suppliers of pipes and fittings as listed below:

1) Purchased pipe and fittings required the submittal of actual heat chemical and physical test results. Each item or part of a fabrication required identification to an individual test report. Welding materials required the submittal of heat or manufacturers' lot reports showing heat chemical and physical test results.

2) Pipe branch lines 2-1/2 inch and larger between the reactor coolant pipes and the isolation stop valves conform to ASTM A376 and meet the supplementary requirement S6 ultrasonic testing. Fittings conform to the requirements of ASTM A403. Fittings 2-1/2 inch and larger had requirements for UT inspection similar to S6 of A376.

Shop fabrication of piping subassemblies was performed by reputable suppliers in accordance with specifications which defined and governed material procurement, detailed design, shop fabrication, cleaning, inspection, identification, packaging and shipment.

Welds for pipes sized 2-1/2 inch and larger were butt welded. Reducing tees were used where the branch size exceeds ½ of the header size. Branch connections of sizes that are equal to or less than ½ of the header size were of a design that conforms to the ANSI rules for reinforcement set forth in the ANSI B31.1 Code for Pressure Piping. Bosses for branch connections are attached to the header by means of full penetration welds.

All welding was performed by welders and welding procedures qualified in accordance with the ASME Boiler and Pressure Vessel Code Section IX, Welding Qualifications. The Shop Fabricator was 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 required prior approval.

All high pressure piping butt welds containing radioactive fluid at greater than 600°F temperature and 600 psig pressure of equivalent were radiographed. The remaining piping butt welds were randomly radiographed. The technique and acceptance standards were those outlined in UW-51 of the ASME B&PV Code Section VIII. In addition, butt welds were liquid penetrant examined in accordance with the procedure of ASME B&PV Code, Section VIII, Appendix VIII and the acceptance standard as defined in the ANSI Nuclear Code Case N-10. Finished branch welds were liquid penetrant examined on the outside and where size permitted, on the inside root surfaces.

A post-bending solution anneal heat treatment was performed on hot-formed stainless steel pipe bends. Completed bends were then completely cleaned of oxidation from all affected surfaces. The Shop Fabricator was 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) was governed by basic ground rules set forth in the specifications. For example, these specifications prohibited the use of hydrochloric acid and limited the chloride content of service water and demineralized water.

Packaging of the piping subassemblies for shipment was done so as to preclude damage during transit and storage. Openings were closed and sealed with tight-fitting covers to prevent entry of moisture and foreign material. Flange facings and weld end preparations were protected from damage by means of wooden cover plates and securely fastened in position. The packing arrangement proposed by the Shop Fabricator was subject to approval.

Field Run Piping

Field running of small diameter piping for essential system including all Engineered Safety Features was not permitted. All seismic Class I and II piping ¾ inch diameter and larger was designed by the architect-engineer. All supports and restraints for that piping were located by the architect-engineer and designed by either the architect-engineer or the subcontractor supplying the pipe support hardware. All seismic Class I stainless steel piping sub-assemblies were prefabricated offsite at one or more subcontractors' pipe fabrication shops. All seismic Class I carbon steel piping subassemblies 2-1/2 inches in diameter and smaller were fabricated in the field.

Certain seismic Class I and II systems comprised of small diameter tubing were field run, for example, the N.S.S.S. Sampling System, which is Class II and utilizes 3/8 diameter tubing.

In instrumentation design, virtually all tubing was field run including tubing for engineered safety related devices. However, the following detailed information was supplied by the architect-engineer where critical design requirements were to be met:

1) Physical location of tubing where separation is required for redundant measurements
2) Detailed design of tubing where thermal expansion of vessels to which tubing is attached requires special expansion loops, etc.
3) Detailed design of typical instrument tubing supports and anchors
4) Detailed design of missile protection of small diameter tubing
5) Detailed design of tubing where proper operation of the instrument is dependent upon adequate slope of lines, etc.

It was found practical to eliminate field running of all seismic Class I and II piping ¾ inch in diameter and larger. However, it was not found practical to limit the use of field running to a greater extent, namely all small diameter seismic Class I and II tubing. Most tubing was erected near the end of the construction phase. At that time, the tubing erection forces had access to more potential support points than were known to the Architect-Engineer. Also, at that time, the construction forces necessarily erected the tubing around objects which would otherwise have been unknown interference during the design phase.

Fabrication of all Class I and II piping, 3/4" diameter and larger, was done by piping subcontractors following orthographic piping drawings prepared by the Architect-Engineer. These same piping drawings were also used by the A-E to prepare isometric piping drawings which were then used for both the stress analysis by the designer and for installation by the field groups. These isometrics also showed locations of pipe supports and restraints. Any deviation from the drawing required approval by the Architect-Engineer. A copy of the final as-built information was directed to the Architect-Engineer for final design review.

Site Quality Control Procedures were followed embracing: purchasing and receiving inspection to assure that all weld filler materials, field run pipe and fittings met Class I and II quality requirement; joint by joint inspection to assure cleanliness; proper weld fit-up; proper welding and welder certification; and performance of required NDT. All these procedures were in accordance with A-E specifications for installation, ASA B31.1, and Section IX of the ASME Code. Detailed Quality Control records were maintained on a piece by piece basis and were also recorded on approved spool and line isometric drawings to assure that complete Quality Control coverage was obtained.

Hydrostatic tests plus hot and/or cold functional tests were performed on completed systems as required and at the appropriate time. No other special quality assurance measures were necessary.

<u>Pump and Valve Motors</u>

<u>Motors Outside the Containment</u>

Motor electrical insulation systems were supplied in accordance with ANSI, IEEE and NEMA standards and tested as required by such standards.

Temperature rise design selection was such that normal long life is achieved even under accident loading conditions.

Criteria for motors of the Safety Injection System required that under any anticipated mode of operation, the motor name plate 1.15 service factor rating is not to be exceeded. Design and test criteria ensure that motor loading does not exceed the application criteria.

Motors Inside the Containment

The SI Recirculation pumps are three stage, vertical pumps driven by 3 phase, 60 cycle, 350 HP motors, and are powered from 480V bus 5A (31) and 480V bus 6A (32). The recirculation pump motors were designed to operate in an ambient condition of saturated steam at 271°F and 47 psig pressure for one day, followed by operation for at least one year at 155°F and 5 psig in a steam atmosphere. The motors are mounted directly to their respective pumps, approximately 2 ft above the highest anticipated water level.

The SI Recirculation Pump motors are provided with thermalastic epoxy insulation and with a heat exchanger. The motors have Class F insulation, temperature rating of 155°C. However, the motor insulation was derated to Class B (130°C) level to provide a safety margin. The operating temperature of the motor insulation is dependent on cooling water temperature rather than the ambient temperature.

The recirculation pump motors are cooled by radiator type coolers using CCW as the cooling medium. Fans are directly connected to the recirculation pump motor shafts. Rotation of the motor rotor and its end fans forces air through the heat exchanger, and air is contained and returned to the ends of the rotor via ducts. A pressure equalizing device permits incident pressure to enter the motor air system so that the bearings are not subject to differential pressures.

The motors are equipped with high temperature grease lubricated ball bearings which would not break down if the bearings were subjected to incident ambient temperatures.

The motors for the valves inside Containment were 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 which are provided only to drive Engineered Safety Features equipment are normally run only for test, the design loading and temperature rise limits are based on accident conditions. Normal design margins were specified for these motors to make sure the expected lifetime included allowance for the occurrence of accident conditions.

Valve Motor Operators

A production line valve motor has been irradiated to a level of 2×10^8 rads using a cobalt-60 irradiation source. The irradiated motor and an identical unirradiated motor have undergone series of reversing tests at room temperature, followed by a series of reversing tests at 275F. The room temperature test was repeated while vibrating the motors at a frequency of 30 cycles per second. 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.

Two independent valve operator manufacturers conducted loss of coolant environmental tests on units similar to those used in this plant. Reports of results indicated that all units operated satisfactorily at test conditions more severe than those expected in the loss-of-coolant or steam-break environment for this plant.

In addition, Westinghouse performed environmental tests on a unit similar to that being used in this plant. The results of the Westinghouse tests indicated that the equipment would perform its required function in the post-LOCA environment.

Electrical Supply

Details of the normal and emergency power sources for the Safety Injection System are presented in Chapter 8.

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, 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, hence, protection is provided in the coolant loop area.

In the event of a Loss-of-Coolant Accident, all piping systems required to function are designed to remain within acceptable stress limits. The stresses due to dead weight, pressure, operational or design basis earthquake, and maximum motions of the Reactor Coolant Loop imposed on the attached Safety Injection Piping were evaluated in accordance with the stress limits in Section 16.1. The inclusion of the stresses in the injection lines required to function due to movements of the Reactor Coolant Loop assures that these lines maintain their integrity during a Loss-of-Coolant Accident.

All piping supports were designed for the loads imposed by the supported system. The rated loads of allowable stress limits for standard manufactured support components are in accordance with requirements of MSS-SP-58-1967. For non-standard supports designed by analysis, the requirements of AISC-1969 were followed. Where support integrity is dependent on reinforced concrete anchorage, the design was in accordance with the Requirements of ACI-318-63.

These standards provide minimum requirements on materials, design and fabrication with ample safety margins for both dead and dynamic loads over the life of the equipment. Specifically, these standards required that:

1) All materials used be in accordance with ASTM specifications which establish quality levels for the manufacturing process, minimum strength properties, and for test requirements which ensure compliance with the specifications

2) There be proper qualification of welding processes and welders for each class of material welded and for types and positions of welds

3) Maximum allowable stress values be established which provide an ample safety margin on both yield strength and ultimate strength.

6.2.3 Design Evaluation

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 where the core has been uncovered for postulated large area ruptures. The result of this performance is to sufficiently limit any increase in clad temperature below a value where emergency core cooling objectives are met. (See Section 6.2.1)

With minimum onsite emergency power available (two-of-three diesel generators), the emergency core cooling equipment consists of two out of three safety injection pumps, one or two out of two residual heat pumps, and three out of four accumulators for a cold leg break and four accumulators for a hot leg break. With these systems, the calculated maximum fuel cladding temperature is limited to a temperature less than that which meets the emergency core cooling design objectives for all break sizes up to and including the double-ended severance of the reactor coolant pipe. (See Section 14.3)

For large area ruptures analyzed (see Section 14.3) the clad temperatures are turned around by the accumulator injection. The active pumping components serve only to complete the refill started by the accumulators. Either two safety injection pumps or one residual heat removal pump provides sufficient addition of water to continue the reduction of clad temperature initially caused by the accumulator.

System Response

To provide protection for large area ruptures in the Reactor Coolant System, the Safety Injection System must respond to rapidly 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.

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 (less than 1%).

The function of the safety injection or residual heat removal pumps is to complete the refill of the vessel and ultimately return the core to a sub-cooled state. The flow from either two safety injection pumps or one residual heat removal pump is sufficient to complete the refill with no loss of level in the core.

The design features applied to the Residual Heat Removal System (RHRS) Valves 730 and 731, that isolate it from the Reactor Coolant System provide a diverse combination of control interlock and mechanical limitations preventing improper opening of these valves and also pressure relief capacity capable of limiting pressure if the valves are not closed upon startup of the plant. These features are:

1) That the valves that are separately interlocked with independent pressure control signals to prevent their being opened whenever the Reactor Coolant System pressure is greater than a designated setpoint (which is below the RHRS design pressure).

The pressure interlock was not specifically designed to meet the requirements of IEEE Standard 279-1971. However, each valve, its associated pressure channel and related circuitry are powered from separate instrument buses, and wiring separation is provided to preclude any single failure from rendering both of the valves' control circuits inoperable. Each of the pressure channels is provided with separate Control Room indication to show channel operability.

A separate pressure interlock is provided for each of the two Valves Nos. 730 and 731. Each pressure interlock prevents its valve from being opened when the Reactor Coolant System pressure is greater than a designated open permissive setpoint and also automatically closes the valve whenever the Reactor Coolant System pressure is above a designated auto-close setpoint. These setpoints are below the design pressure of the RHRS.

While the automatic closure interlock for MOV-730 and –731 will prevent over-pressurizing the RHR system piping during an RCS pressure increase transient, this interlock will isolate the suction source of the operating RHR pump(s), potentially causing pump failure. In order to prevent inadvertent isolation of the RHR pump suction, this auto-closure interlock may be defeated by de-energizing the motor operators to MOV-730 and –731. Prior to de-energizing these MOV's Reactor Coolant System T_{ave} must be below 200°F, depressurized and vented through a minimum equivalent opening of two (2) square inches.

2) That the Reactor Coolant System pressure interlocks meet single failure criteria.

3) That the motors are qualified in accordance with IEEE 323-1974, IEEE 344-1975, IEEE 382-1972 for increased reliability and operability in the normal and accident containment environment.

The Residual Heat Removal System was designed for a pressure of 600 psig and 400°F and was hydrostatically tested at a pressure of 900 psig prior to initial operation. Insofar as the piping itself is concerned, the piping code (USAS B31.1) allows a rating of 700 psig at 400°F for schedule 40 stainless steel pipe. Thus the piping system, as presently designed, incorporates a considerable margin in that it is rated at a pressure-temperature condition which is less than that allowed by Code. It is also noted that the Code allows an overpressure allowance above the design pressure under transient conditions.

4) That the RHRS is equipped with a pressure relief valve RV-1836 sized with a relief capacity of 400 gpm. This is a diverse backup to administrative closure of the isolation valves prior to startup to prevent overpressurization when returning the plant to operation. In addition, Technical Specification Section 3.4.12 restricts operation of the SI pumps when the RCS average cold leg temperature is below the OPS enable temperature. These restrictions help to preclude RHR overpressurization.

5) To preclude spurious closure of the valves, the control circuitry is of the energize-to-actuate principle.

6) That each of the pressure channels has a separate Control Room indication to show channel operability.

7) Open/close position indication lights are provided for these valves as well as a visual and audible alarm to indicate when either valve leaves its full open position.

Initial response of the injection systems is automatic with appropriate allowances for delays in 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 analysis of system performance, delays in reaching the programmed trip points and in 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 designed so that delivery of the full rated flow is reached within 27 seconds after the process parameters reach the set points for the injection signal.

EVENT	SECONDS
Time to initiate the safety injection signal	2
Time for diesel generators to come up to speed	10
Time for safety injection pumps to come up to speed	10
Time for Residual Heat Removal Pumps to come up to speed	5
Total	**27**

Motor control centers are energized and injection valves are opened during this time to allow pumped ECCS delivery.

This delay is consistent with the 25 second delay which is assumed in the analysis of the Loss-of-Coolant Accident as described in Chapter 14. The modeling of a 25 second SI delay time is conservative for this action sequence since no credit is taken for Charging or SI flow prior to 25 seconds (although these pumps are actually up to speed), and credit is not taken for partial RHR flow up to 25 seconds. On this basis, the integral injection flow for the assumed 25 second delay time remains less than the actual injection flow that would be delivered if partial credit for pumps on prior to 25 seconds was assumed.

To reduce inadvertent Safety Injection System Actuations due to instrumentation lags in the engineered safeguards system high steamline flow, low average temperature Tavg/Low steamline pressure coincidence circuitry, a time delay will be installed in each train (a maximum time delay of 6 seconds will meet the acceptance criteria for a steamline rupture).

Single Failure Analysis

A single active failure analysis is presented in Table 6.2-7. All credible active system failures were considered. The analysis of the Loss-of-Coolant Accident presented in Chapter 14 is consistent with the single failure analysis.

It 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.

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 evaluated in Table 6.2-8.

The procedure followed to establish the alternate flow path also isolates the spilling line. A valve is provided in the containment recirculation line to the residual heat removal pumps in order 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 Systems under Loss-of-Coolant Accident conditions are described in Sections 9.3 and 9.6, respectively.

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. 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 the reactor shutdown operations. Whenever the reactor is at power, the pumps are aligned for emergency duty.

Shared Function Evaluation

Table 6.2-9 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.

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 opened 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 disc 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 heat-up operation, the check valves are tested for leakage as soon as there is about 100 psi differential across the valve. This test confirms the seating of the disc and whether or not there has been an increase in 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 6.2-10 indicates what in-leakage 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 test (100 cc/hr, i.e., 10 cc/hr/in).

In-leakage at a rate of 10 cc/hr/inch would require that the accumulator water volume be adjusted approximately once every 8 months. This would indicate that level adjustments can be scheduled and that this work can be done at the operator’s convenience. At a leak rate of 80 cc/hr/inch (8 times the acceptance leak rate), the water level will have to be readjusted approximately once a month. This readjustment will take about 2 hours 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 accumulators would cause an increase in the containment pressure of approximately 0.1 psi. This release of gas has been included in the containment pressure analysis for the Loss-of-Coolant Accident, Chapter 14.

During normal operation, the flow rate through the reactor coolant piping is approximately five 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

<u>Emergency Flow to the Core</u>

Special attention is given to factors that could adversely affect the accumulator and safety injection flow to the core. These factors are:

- Steam binding in the core, including flow blockage due to loop sealing
- Loss of accumulator water during blowdown

- Short circuiting of the accumulator from the core to another part of the Reactor Coolant System
- Loss of accumulator water through the breaks.

All of the above are considered in the analysis and are discussed quantitatively in Chapter 14.

External Recirculation Loop Leakage

The Authority has established a program to identify and reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident.

The following systems are included in the subject program for leak identification and reduction:

Chemical and Volume Control System
a) Volume Control Tank up to the outlet isolation valve including gas space
b) RCP seal return line from the containment penetration to the VCT

Residual Heat Removal System
Suction to pumps form Loop 32
Discharge of pumps to containment

Safety Injection System
a) Recirculation path from containment sump thru RHR pumps to RHR heat exchanger and to safety injection pump suction
b) Recirculation path from recirculation pump discharge to safety injection pump suction
c) Safety injection pump discharge path to containment
d) Boron injection tank
e) Alternate recirculation path from the containment sump thru the RHR pump to the suction of 32 SI pump (bypassing the RHR heat exchanger).

Primary Sampling System
a) Reactor Coolant Hot Leg Sample
b) Recirculation Pumps Sample
c) RHR Loop Sample
d) Volume Control Tank Sample up to the outlet isolation valve

Post-Accident Containment Air Sampling System
Sample from Containment and return to Containment.

6) Containment Hydrogen Monitoring System
From Containment and to Containment

Leakage detection exterior to the Containment is achieved through use of sump tank and waste holdup tank level detection. The Primary Auxiliary Building sump pumps start automatically in the event that liquid accumulates in the sump. Valving is provided to permit the operator to individually isolate the residual heat removal pumps.

Pump NPSH Requirements

Residual Heat Removal Pumps

The NPSH of the residual heat removal pumps is evaluated for normal plant shutdown operation, and both the injection and recirculation phase operation 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. The flow produced by one (1) RHR Pump, operating in conjunction with one (1) RHR heat Exchanger, is presently throttled to 5000 gpm to avoid flow-induced vibrations of the heat exchanger tube bundle. When one (1) RHR Pump delivers flow through both RHR Heat Exchangers, the combined flow will be throttled to less than 4500 gpm. At this flow rate, the NPSH margin is 15%.

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, and 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, 40 percent NPSH margin is available assuming all three safety injection pumps running at run-out condition together with two RHR pumps at design flow.

Recirculation Pumps

The NPSH for the Recirculation Pumps is evaluated for the recirculation phase of operation for the design basis accident. The NPSH available is determined from the elevation head of the water above the vendor-provided reference point for the entrance of the first stage impeller of the pumps.

The new Internal Recirculation Pump is a conventional vertical type pump with a double suction inlet design, which requires considerably less NPSH than the previous single suction pump design it replaces.

In the NPSH evaluation, water is assumed to be at saturated conditions and no credit is taken for the Containment pressure after the accident. At the initiation of the recirculation phase, adequate NPSH is available for one (1) or two (2) pump operation.

6.2.4 Minimum Operating Conditions

The Technical Specifications establish limiting conditions regarding the operability of the system when in MODES 1, 2, 3, and 4.

6.2.5 Inspections and Tests

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.

In addition, to the extent practical, the critical parts of the reactor vessel internals, pipes, valves and pumps are inspected visually or by boroscopic examination for erosion, corrosion, and vibration wear evidence and for non-destructive test inspection where such techniques are desirable and appropriate.

Pre-Operational Testing

Component Testing

Pre-operational 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. NPSH was established at design flow by means of adjusting suction pressure for a representative pump.

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.

System Testing

Initial functional tests of the core cooling portion of the Safety Injection System were conducted before initial plant startup. These tests were performed following the flushing and hydrostatic testing of the system and with the Reactor Coolant System cold. The Safety Injection System valving was set initially to simulate the system alignment for Plant Power Operation.

The functional tests were divided into two parts:

1) Demonstrating the proper function of instrumentation and actuation circuits, confirm valve operating times, confirm pump motor starting times, and demonstrate the proper automatic sequencing of load addition to the emergency diesels.

   These tests were repeated for the various modes of operation needed to demonstrate performance at partial effectiveness, i.e., 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 tests were performed without delivery of water to the Reactor Coolant System, but included the starting of all pumping equipment involved in each test.

2) Demonstrating the proper delivery rates of injection water to the Reactor Coolant System.

To initiate the first part of 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 the low water level and low-pressure signals from the pressurizer instrumentation. Simultaneously, the breakers supplying outside power to the 480 volt buses were tripped manually and operation of the emergency diesel system automatically commences. 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 second portion of the test was initiated by manually starting individual pumps on mini-flow and manually opening the appropriate isolation valve to deliver water to the Reactor Coolant System. Data was taken to verify proper pump performance and flow delivery rates.

The systems were accepted only after demonstration of proper actuation and after demonstration of flow delivery and shutoff head within design requirements.

Post-Operational Testing

Component Testing

Routine periodic testing of the Safety Injection System components and all necessary support systems at power is performed. The safety injection and residual heat removal pumps are to be tested in accordance with the Indian Point 3 Inservice Testing Program, to check the operation of the starting circuits, verify the pumps are in satisfactory running order, and verification is made that required discharge head is attained. No inflow to the Reactor Coolant System occurs whenever the reactor coolant pressure is above 1500 psi. If 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 the period within which the component should be restored to service.

The operation of the remote stop valves in the accumulator tank discharge line may be tested by opening the remote test valves just downstream of the stop valve. Flow through the test line can be observed on instruments and the opening and closing of the discharge line stop valves can be sensed on this instrumentation. Test circuits are provided to periodically examine the leakage back through the check valves and to ascertain that these valves seat whenever the Reactor Coolant System pressure is raised.

This test is routinely performed when the reactor is being returned to power after an outage and the reactor pressure is raised above the accumulator pressure. If leakage through a check valve should become excessive, the isolation valve would be closed (the safety injection actuation signal will cause this valve to open should it be in the closed position at the time of a Loss-of-Coolant Accident). The performance of the check valves has been carefully studied and it is concluded that it is highly unlikely that the accumulator lines would have to be closed because of leakage.

The recirculation pumps are normally in a dry sump. Minimum flow testing of these pumps can be performed during refueling operations by filling the recirculation sump and opening the mini flow valve on the discharge of the pump and directing the flow back to the sump. Those service water and component cooling pumps which are not running during normal operation may be tested by alternating with the operating pumps.

The content of the accumulators, the Boron Injection Tank and the Refueling Water Storage Tank are sampled periodically to determine that the required boron concentration is present.

System Testing

System testing can be 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 components receive the safety injection signal in the proper sequence. The Safety Injection and Residual Heat Removal pumps are blocked from starting. Isolation valves in the injection lines are blocked closed so that flow is not introduced into the reactor coolant system. The system test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry. The test is considered satisfactory if control board indication and visual observations indicate all components have operated and sequenced properly. A complete system test cannot be performed when the reactor is operating because a safety injection signal would cause a reactor trip. The method of assuring complete operability of the Safety Injection System is to combine the system test performed during plant shutdown with more frequent component tests, which can be performed during reactor operation.

The accumulators and the safety injection piping up to the final isolation valve are maintained sufficiently full of borated water while the plant is in operation to ensure the system remains operable and performs properly. The accumulators and the high head injection lines are refilled with borated water as required by using the safety injection pumps to recirculate refueling water through the injection lines. A small test line is provided for the purpose in each injection header.

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 eight-switch sequence for recirculation operation may be tested following the above injection phase test 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 utilized during external recirculation (safety injection and residual heat removal pumps) in turn at near shutoff head conditions 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, leak-off connections or other potential points of leakage are 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 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.

In the event of any modification to the high head safety injection system piping and/or valve arrangement, a system flow test is required. System flow testing establishes and verifies that the actual performance capability of the system is within minimum calculated safety analysis flow ranges.

References

1) Bell, M. J. et al., "Investigations of Chemical Additives for Reactor Containment Sprays," WCAP-7153 (Westinghouse Confidential), March 1968.

2) Revised Feasibility Report for BIT Elimination for Indian Point Unit 3, dated July 1988 (Westinghouse).

3) Modification MOD 86-03-150 SIS, "Elimination of Boron Injection Tank, Phase I."

4) Nuclear Safety Evaluation No. NSE 86-03-150 SIS, "Elimination of Boron Injection Tank, Phase I."

5) Classification CLAS 86-03-150 SIS, "Elimination of Boron Injection Tank, Phase I."

6) FSAR, Section 14.2.5

7) Modification ER-04-3-066, "Containment SI Recirculation Pumps Replacement."

8) NYPA Calculation IP3-CALC-VC-02513, "IP3 Sump Water Level Following LBLOCA-DEPSG."

9) NYPA Calculation IP3-CALC-SI-02430, "NPSHA / NPSHR for SI Internal Recirculation Pumps 31 and 32."

10) Westinghouse Calculation SEE-04-24, "Hydraulic Performance of the Containment Recirculation Pumps (Ingersol-Rand 24APK-3) for IP3."

TABLE 6.2-1

SAFETY INJECTION SYSTEM – CODE REQUIREMENTS

Components	Code
Refueling Water Storage Tank	AWWA D100-67
Residual Heat Removal Heat Exchanger	
Tube Side	ASME Section III Class C
Shell Side	ASME Section VIII
Accumulators	ASME Section III Class C
Boron Injection Tank	ASME Section III Class C
Valves	ANSI B16.5 (1955)
Piping	ANSI B31.1 (1967)

TABLE 6.2-2

ACCUMULATOR DESIGN PARAMETERS

Quantity	4
Type	Stainless Steel lined/carbon steel
Design Pressure, psig	700
Design Temperature, °F	300
Operating Temperature, °F	#130
Normal Operating Pressure, psig	650
Minimum Operating Pressure, psig	615
Total Volume, ft^3	1100
Minimum/Maximum Water Volume at Operating Conditions, ft^3	775/815
Minimum/Maximum Boron Concentration – Normal Operating Conditions (as boric acid), ppm	2000/2600
Relief Valve Set Point, psig*	700

*The relief valves have soft seats and are designed and tested to ensure zero leakage at the normal operating pressure.

TABLE 6.2-3

BORON INJECTION TANK DESIGN PARAMETERS

QUANTITY	1
Total Volume, gal	900
Boron Concentration (as boric acid) nominal, ppm	21,000**
Design Pressure, psig	1750
Design Temperature, °F	300
Operating Pressure, psig	0 – 1500*
Operating Temperature, °F	150 – 180***
Material	Stainless Steel
Number of Strip Heaters	12 (permanently de-energized)
Heater Capacity, each, kW	1 (permanently de-energized)

* 1500 psig is normal maximum, but could reach 1670 psig for short periods.
** Actual boron concentration is maintained at approximately 2500 ppm.
*** Actual operating temperature is ambient.

TABLE 6.2-4

REFUELING WATER STORAGE TANK DESIGN PARAMETERS

Quantity	1
Material	Stainless Steel
Total Tank Capacity at Overflow, gal	355,200
Nominal Water Volume, gal	342,200
Normal Pressure, psig	Atmospheric
Operating Temperature, °F	Above freezing
Design Pressure, psig	Atmospheric
Design Temperature, °F	120
Minimum/Maximum Boron Concentration – Normal Operating Conditions (as boric acid), ppm	2400/2600
Type of heating	Steam

TABLE 6.2-5

PUMP DESIGN PARAMETERS

Safety Injection Pump	
Quantity	3
Design Pressure, discharge, psig	1700
Design Pressure, suction, psig	250
Design Temperature, °F	300
Design Flow Rate, gpm	400
Maximum Flow Rate, gpm	675
NPSH Required at Maximum Flow Rate, ft	35
Design Head, ft	2500
Design Shutoff Head, ft	3500
Material	Austenitic Stainless Steel
Type	Horizontal, centrifugal
Motor Horsepower	400

Recirculation Pump	
Quantity	2
Type	Vertical, centrifugal
Design Pressure, discharge, psig	250
Design Temperature, °F	300
Design Flow, gpm	3000
Design Head, ft	370*
Material	Austenitic Stainless Steel
Maximum Flow Rate, gpm	4580*
Shutoff Head, ft	460*
Motor Horsepower	350

*Highest of the two pumps.

PUMP DESIGN PARAMETERS

Residual Heat Removal Pump	
Quantity	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	4500
Design Shutoff Head*, ft	390
Motor Horsepower	400

*For analytical purposes these design values should be reduced by 18 ft. to allow for pump wear (See Figure 6.2-3).

TABLE 6.2-6

RESIDUAL HEAT REMOVAL SYSTEM HEAT EXCHANGERS DESIGN PARAMETERS

Number	2	
Type	Vertical shell and U-tube	
Heat exchanged, Btu/hr	30.8×10^6	
Fouled transfer rate, Btu/hr-°F-ft^2	309	
Clean transfer rate, Btu/hr-°F-ft^2	410	
Surface area, ft^2	3579	
Overall heat transfer coefficient*, Btu/hr-°F	1.1×10^6	
Design cycles (85°F - 350°F)	200	
Design Conditions:		
Parameter	Tube Side	Shell Side
Pressure, psig	600	150
Temperature, °F	400	200
Flow, lb/hr	1.44×10^6	2.46×10^6
Inlet temperature, °F	135	95
Outlet temperature, °F	113.5	100.8
Material	Stainless Steel	Carbon Steel

*Fouled transfer rate multiplied by the design surface area.

TABLE 6.2-7
SINGLE ACTIVE FAILURE ANALYSIS – SAFETY INJECTION SYSTEM

Component	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 and recirculation phase as marked)		
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 plus at least two safety injection pumps
3) Component cooling*	Fails to start	A total of 1 of 3 required during recirculation
4) Conventional and nuclear service water*	Fails to start	A total of 3 of 6 required during recirculation
5) Recirculation*	Fails to start	Two provided. One required to operate during recirculation
6) Auxiliary component cooling	Fails to start	Two pairs provided. One required per pair to operate during injection
C. Automatically Operated Valves: (Repositioned on Safety Injection Signal) – (Injection phase)		
Boron Injection Tank Isolation		
Inlet (Valves 1852A & 1852B)	Fails closed	Valves are normally open in two parallel lines, one valve in either line remains open
Outlet (Valves 1835A & 1835B)	Fails closed	Valves are normally open in two parallel lines, one valve in either line remains open
2) Accumulator discharge valves (894A-D	Fails closed	All four valves are normally open during power operation with AC power removed
3) Boron injection tank recirculation isolation valves (1851A/B)	Fails closed	Flowpath is isolated by locked closed valves 1844, 1848, 1198A and 1198B
4) Residual heat removal line isolation valve at residual heat exchanger discharge (valves 638, 640, 746, 747, 899A, 899B)	Fails to start	Cross-over line provided to assure sufficient flow for closure of any valve with two RHR pumps running

TABLE 6.2-7
(Cont.)

SINGLE ACTIVE FAILURE ANALYSIS – SAFETY INJECTION SYSTEM

Component	Malfunction	Comments
5) Isolation valve on component cooling water line from residual heat exchangers (valves 822A & 822B)	Fails to open	Two parallel, one valve in either line is required to open
D. Power Operated Valves : (Spurious Repositioning) (Injection Phase)		
1) Safety injection pump recirculation isolation valves (842, 843)	Fails closed	Valves are normally open during power operation with AC power removed
2) Residual heat removal recirculation isolation valves (743, 1870)	Fails closed	Valves are normally open during power operation with AC power removed
3) Refeueling water storage tank suction isolation removal line isolation valves (1810, 882)	Fails closed	Valves are normally open during power operation with AC power removed
4) Residual heat removal pump discharge header isolation valve (744)	Fails closed	Valve is normally open during power operation with AC power removed
5) High-head safety injection header hot leg isolation valves (856B, 856G)	Fails open	Valves are normally closed during power operation with AC power removed
6) High-head safety injection header cold leg isolation valves (856C, 856E, 856H, 856J)	Fails closed	Valves are normally open during power operation with AC power supplied. The reduced flow capability with single valve closure is analyzed by the flow delivery of all three safety injection pumps
E. Emergency Power: (injection or recirculation phase)		
1) Emergency Diesel 31	Fails to run	Two of three safety injection pumps, one of two residual heat removal pumps and two of two recirculation pumps available to operate
2) Emergency Diesel 32	Fails to run	Two of three safety injection pumps, one of two residual heat removal pumps and one of two recirculation pumps available to operate
3) Emergency Diesel 33	Fails to run	Two of three safety injection pumps, two of two residual heat removal pumps and one of two recirculation pumps available to operate

F. Valves Operated from Control Room for Recirculation (recirculation phase)		
1) Recirculation internal recirculation isolation (valves 1802A & 1802B)	Fails to open	Two valves in parallel, one valve in either line is required to open
2) Safety injection pump suction valve at residual heat exchanger discharge (valves 888A & 888B)	Fails to open	Two valves in parallel, one valve is required to open
3) Isolation valve on the miniflow line returning to the refueling water storage tank (valves 842 & 843)	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 (valves 1810 & 847)	Fails to close	Two valves in series, one required to close (one valve is a check valve)
5) Residual heat removal pump recirculation line (valves 743 & 1870)	Fails to close	Two valves in series, one required to close
6) Residual heat removal pump discharge line (valve (744 and check valve 741)	Fails to close	Two valves in series, one required to close (one valve is a check valve)
7) Residual heat removal line isolation valves at residual heat exchanger discharge (valves 746 & 899A and 747 & 899B)	Fails to close	Two valves in series in each of two parallel lines; one valve of each pair is required to close
8) Isolation valves for high head hot leg recirculation (Valves 856B and 856G)	Fails to open	Two valves, one in each of two parallel lines, one valve is required to open
9) Safety injection pump 32 suction isolation valves (887A/B)	Fails to close	Two valves provided in series; only one valve required to isolate high-head recirculation suction flow paths

TABLE 6.2-7
(Cont.)

SINGLE ACTIVE FAILURE ANALYSIS – SAFETY INJECTION SYSTEM

10) High-head safety injection header hot leg isolation valves (856B, 856G)	Fails open	Valves have AC power until realignment to high-head hot leg recirculation to prevent spurious mispositioning
11) High-head safety injection header cold leg isolation valves (856C, 856E, 856H, 856J)	Fails closed	Valves are normally open during cold leg recirculation. The spurious closure of one valve results in adequate flow performance with only two safety injection pumps operating.
12) High-head safety injection header cold leg isolation valves (856C, 856E, 856H, 856J)	Fails open	Valves have AC power removed following realignment to high-head hot leg recirculation to prevent spurious mispositioning
13) High-head safety injection header hot leg isolation valves (856B, 856G)	Fails closed	Valves have AC power removed following realignment to high-head hot leg recirculation to prevent spurious mispositioning
14) Limit switches on high-head safety injection header hot leg isolation valves (856B, 856G)	Fails to stop valve in throttled position while opening	Fully open valve causes pump failure in affected header. Remaining header provides adequate flow for core cooling. (Note that 856B is normally full open by design for hot leg injection.)
15) Boron Injection Tank Outlet Isolation Valves, (1835A/B)	Fails to open	If closed during low head recirculation, one valve in either parallel line is required to open during transfer to leg recirculation.

* Recirculation phase

Note: The status of all active components of the Safety Injection System is indicated on the main control board. Reference is made to Table 6.2-12.

TABLE 6.2-8
SINGLE PASSIVE FAILURE ANALYSIS
(LOSS OF RECIRCULATION FLOW PATH)

Flow Path	Indication of Loss of Flow Path	Alternate Flow Path
Low Head Recirculation From recirculation sump to low head injection header via the recirculation pumps and the residual heat exchanger.	1. Insufficient flow in low head injection lines (one flow monitor in each of the four low head injection lines*)	From recirculation pump to high head injection header via the recirculation pumps, one of the two residual heat exchangers and the safety injection pump.**
	2. As 1 above.	a. From containment sump to discharge header of the residual heat exchanger via the residual heat removal pumps. b. If flow is 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.
High Head Recirculation From recirculation sump to high head injection header via the recirculation pumps, one of the two residual heat exchangers and the high head injection pumps.	1. No flow in high head injection header (three flow monitors, one in each injection line, and one pressure monitor). (Note: One of the four cold leg lines per header has been isolated by a locked closed valve.)	a. From containment sump to high head injection header via the residual heat removal pumps, one of the two residual heat exchangers and the high head injection pumps.

Note: As shown on Plant Drawings 9321-F-27353 and 27503 [Formerly Figures 6.2-1A and –B], there are valves at all locations where alternative flow paths are provided.

* With the flow meters on three or more lines indicating greater than zero and with the lowest of these flows at least 360 gpm, $\pm$ 10 gpm, or with zero flow indicated on two lines and the lowest flow meter for each of the remaining lines reading at least 360 gpm, $\pm$ 10 gpm, the supply of recirculated water using low head recirculation will maintain the core flooded even in the event of a low head line spilling and one failed flow meter or other single failure.

TABLE 6.2-8
(Cont.)
SINGLE PASSIVE FAILURE ANALYSIS
(LOSS OF RECIRCULATION FLOW PATH)

Flow Path	Indication of Loss of Flow Path	Alternate Flow Path
		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 safety injection pump 32 (by-passing the residual heat exchangers*).
	2. Flow in only one of the two high head injection branch headers (three flow monitors per branch header). (Note: One of the four cold leg lines per header has been isolated by a locked closed valve.)	a. as 1 (b) except that flow from safety injection pump 32 is only supplied to the unbroken branch header.

NOTE: As shown on Plant Drawings 9321-F-27353 and 27503 [Formerly Figures 6.2-1A and –B],, there are valves at all locations where alternative flow paths are provided.

* 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 boil-off of the water to the Containment; heat is then removed from the Containment by either the containment fan coolers or/and the Containment Spray System (using cooled water from the recirculation sump via the recirculation pumps and one residual heat exchanger).

TABLE 6.2-9

SHARED FUNCTIONS EVALUATION

Component	Normal Operating Function	Normal Operating Arrangement	Accident Function	Accident Arrangement
Boron Injection Tank (BIT)	None	Lined up to discharge of safety injection pumps	None*	Lined up to discharge of safety injection pumps
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)	Accumulator fill. Core cooling inventory makeup during RCS reduced inventory	Lined up to hot and cold legs of reactor coolant piping	Supply borated water to core	Lined up to cold legs of reactor coolant piping
Residual Heat Removal Pumps 92)	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 or reactor coolant piping spray headers, and suction of safety injection pumps

*A modification replaced the concentrated boric accident in the BIT with refueling water, however, no credit is taken for any boron in the BIT by current accident analyses (Refer to FSAR Section 14.2.5).

TABLE 6.2-9
(Cont.)

SHARED FUNCTIONS EVALUATION

Component	Normal Operating Function	Normal Operating Arrangement	Accident Function	Accident Arrangement
Conventional Service Water Pumps (3)	Supply river cooling water to component cooling heat exchangers (2)	Two pumps in service	Supply river cooling water to component cooling heat exchanger (1) and safeguards components	*One or two pumps in service
Component Cooling Pumps (3)	Supply cooling water to station nuclear components	Two pumps in service	Supply cooling water to residual heat exchangers S.I. pumps bearings and recirculation pump motor coolers	*One or two pumps in service
Residual Heat Exchangers (2)	Remove residual heat from core during shutdown	Lined up for residual heat removal pump operation	Cool water in containment sump for core cooling and containment spray	Lined up for the discharge of recirculation pumps
Component Cooling Heat Exchangers (2)	Remove heat from component cooling water	Two heat exchangers in service	Cool water for residual heat exchangers and core cooling pumps	Both heat exchangers in service
Auxiliary Component Cooling Pumps (4)	None	Lined up for pump operation	Provide component cooling water to recirculation pump motor coolers	Lined up for pump operation
Nuclear Service Water Pumps (3)	Supply river cooling water to station safeguards and non-nuclear components	One or two pumps in service	Supply river cooling water to component cooling heat exchanger (1) and safeguards components	*Two or three pumps depending on active failure (all are started automatically)

* Recirculation phase

TABLE 6.2-10

ACCUMULATOR INLEAKAGE*

Frequency of Level Adjustment Months	Leak Rate (cc/hr)	Leak Rate/Maximum Allowable Leak Rate
1	787	7.87
3	262	2.62
6	131	1.31
7.9	100	1.00
9	87.4	0.874
12	65.6	0.656

TABLE 6.2-11

INSTRUMENTATION READOUTS ON THE CONTROL BOARD
FOR OPERATOR MONITORING DURING RECIRCULATION

System	Valves	Valve Number
SIS		MOV 1802 A, B
SIS		MOV 1810
SIS		MOV 885 A,B
SIS		MOV 899 A, B
SIS		MOV 888 A, B
SIS		MOV 866 A, B
SIS		MOV 889 A, B
SIS		MOV 851 A, B
SIS		MOV 856, C, E, G
SIS		MOV 856 B, H, J
SIS		MOV 882
SIS		MOV 842
SIS		MOV 843
SIS		MOV 1852 A, B
SIS		MOV 1835 A, B
SIS		AOV 1851 A, B
SIS		MOV 894 A, B, C, D
SIS		MOV 850 A, C
ACS		MOV 744
ACS		MOV 745 A, B,
SIS		MOV 746
SIS		MOV 747
SIS		MOV 883
SIS		MOV 887 A, B
SIS		AOV 1813
SIS		MOV 1869 A, B
SIS		HCV 638
SIS		HCV 640
ACS		MOV 743
ACS		MOV 1870

TABLE 6.2-11
(Cont.)

INSTRUMENTATION READOUTS ON THE CONTROL BOARD
FOR OPERATOR MONITORING DURING RECIRCULATION

System	Instruments	Channel Number
SIS		FI 945 A, B
SIS		FI 946 A, B, C, D
SIS		FI 924 A
SIS		FI 925
SIS		FI 926 (This line has been permanently locked closed: isolated by valve SI-856A).
SIS		FI 926 A (This line has been permanently locked closed: isolated by valve SI-856F).
SIS		FI 927
SIS		FI 980
SIS		FI 981
SIS		FI 982
SIS		LI 1251
SIS		LI 1252
SIS		LI 1253
SIS		LI 1254
SIS		LI 1255
SIS		LI 1256
SIS		PI 922
SIS		PI 923
SIS		PI 947
ACS		PI 635
ACS		FI 640 & FI 638
ACS		LIT 628 & LIT 629
ACS		TR 639 & TR 641
RCS		LRCA 459
RCS		LICA 460
RCS		LICA 461
RCS		LI 462
SIS		Safety Injection
SW		Service Water
ACS		Component Cooling
SIS		Containment Spray
SIS		Recirculation
ACS		Residual Heat Removal

TABLE 6.2-12

QUALITY STANDARDS OF SAFETY INJECTION SYSTEM COMPONENTS RESIDUAL HEAT EXCHANGER

Tests and Inspections

1. Hydrostatic Test
2. Radiograph of longitudinal and girth welds (tube side only)
3. UT 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

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

Test and Inspections

Hydrostatic Test
Dye penetrant test of welds

Special Manufacturing Process Control

Welding and NDT and procedure review
Surveillance of supplier quality control and product

QUALITY STANDARDS OF SAFETY INJECTION SYSTEM COMPONENTS
SAFETY INJECTION, RECIRCULATION AND RESIDUAL HEAT REMOVAL
PUMPS

Tests and Inspections
Performance Test
Dye penetrant of pressure retaining parts*
Hydrostatic Test

Special Manufacturing Process Control
Weld, NDT and inspection procedures for review
Surveillance of suppliers quality control system and product

ACCUMULATORS

Test and Inspections
Hydrostatic test
Radiography of longitudinal and girth welds
Dye penetrant/magnetic particle of weld

Special Manufacturing Process Control
Weld, fabrication, NDT and inspection procedure review
Surveillance of suppliers quality control and product

VALVES

A. Tests and Inspections
a) 200 psi and 200 F or below (cast or bar stock)
Dye Penetrant Test
Hydrostatic Test
Seat Leakage Test

* Except Internal Recirculation Pump

QUALITY STANDARDS OF SAFETY INJECTION SYSTEM COMPONENTS
SAFETY INJECTION, RECIRCULATION AND RESIDUAL HEAT REMOVAL
VALVES

b) Above 200 psi and 200 F

Forged Valves (2½" and larger)
1. UT of billet prior to forging
2. Dye penetrant 100% of accessible areas after forging
3. Hydrostatic test
4. Seal leakage test

(ii) Cast Valves
1. Radiograph 100%*
2. Dye penetrant all accessible areas*
3. Hydrostatic test
4. Seat Leakage

c) Functional Test Required for:

1. Motor Operated Valves
2. 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)

- For valves with radioactive service only

QUALITY STANDARDS OF SAFETY INJECTION SYSTEM COMPONENTS
REFUELING WATER STORAGE TANK

A. Tests and Inspections

Vacuum box test of tank bottom seams
1. Hydrostatic test of tank
2. Hydrostatic test of tank heater coil
3. Spot radiography of longitudinal and girth welds

B. Special Manufacturing Process and Material Control
1. Weld, fabrication, NDT and inspection procedure review
2. Surveillance of suppliers' quality control system and product
3. Material chemical and physical properties certification

PIPING

A. Test and Inspections

Class 1501 and below

Seamless or welded. If welded 100% radiography is required, shop fabricated and field fabricated pipe weld joints are inspected as follows:
2501R – 601R – 100% radiographic inspection and penetrant examination
301R – 302 – 20% random radiographic inspection
151R – 152R – 100% liquid penetrant examination

B. Special Manufacturing Process Control

Surveillance of suppliers' quality control and product

TABLE 6.2-13

NPSH REQUIREMENTS FOR CLASS 1 (SEISMIC) PUMPS

Pump	Operating Mode	NPSH Req'd	NPSH Avail.	Fluid Oper. Temp.	P Atmospheric	Elevation of Pump*
Containment Spray (2)	Two pumps taking suction from RWST @ 3,000 gpm each & pumping to spray header – until Transfer to Recirculation.**	17.0 Ft	26.5 Ft	120° F	14.7 psia	Approx. 44'
	Two pumps taking suction from RWST @ 3,000 gpm each and pumping to spray header – until Termination of CS Pump operation.***	17.0 Ft	17.5 Ft	120° F	14.7 psia	Approx. 44'
Safety Injection (3)	Injection to RCS-3 SI pumps @ 650 gpm each	30.0 Ft	42.1 Ft	100° F	14.7 psia	37' -3"
	Recirculation to RCS-2 SI pumps @ 675 gpm each (Fluid supplied by one internal recirculation pump)	35.0 Ft	79.9 Ft	256° F (Maximum)	138.7 psia (Developed from discharge pressure of internal recirculation pump(s))	37' -3" This value used in NPSHA calculation. Actual pump suction centerline elevation is 36' 8.5"
Residual Heat (2)	Injection to RCS-2 RHR pumps @ 3,000 gpm each	11 Ft	59.4 Ft	100° F	14.7 psia	17' - 0"
Internal Recir. (1)	Recirculation to RCS – one pump @ approx. 3,000 gpm (indicated)	7.95 Ft	10.54 Ft****	256° F (Maximum)	Pressure corresponding to saturation temperature of fluid in sump (33 psia max.)	37.85' (NPSH Reference point)

* Centerline of suction, except as noted.
** Sufficient NPSH is available per IP3-CAL-CS-02590 Rev. 0
*** Bounding Case since Design Basis has only one CS Pump operating during this scenario.
**** Credits remainder of RWST water delivered into Containment prior to start of recirculation containment spray. (Ref. 8 and 9)

TABLE 6.2-13
(Cont.)

NPSH REQUIREMENTS FOR CLASS 1 (SEISMIC) PUMPS

Pump	Operating Mode	NPSH Req'd	NPSH Avail.	Fluid Oper. Temp.	P Atmospheric	Elevation of Pump*
Comp. Cooling Water (1)	One pump, post Accident recirculation; @ 5500 gpm each	29.0 ft	31.9 ft at 197° F	179° F (Maximum)	14.7 psia	43' –3"
Motor Driven Auxiliary Feedwater (2)		20 ft	52 ft	Ambient (100° F Maximum)	14.7 psia	20' –0.5"
Turbine Driven Auxiliary Feedwater (1)		16 ft	66 ft	Ambient (100° F Maximum)	14.7 psia	20' –3.5"
I.A. Compr. Closed Cooling Water (2)		less than 4 ft	flooded (head tank)	130° F	14.7 psia	16' -6"
Service Water (6) (vertical)		22 ft (6000 gpm) 29 ft (7500 gpm)	37.9 ft	28-95° F	14.7 psia	8' -6" (Centerline of discharge)
Diesel Fuel Oil Transfer (3) (vertical)		1 ft Subm. (Tk. Mounted)	6.39 ft sub-mergence	35-110° F	14.7 psia	40' (Approximate centerline of discharge)
Primary Water Makeup (2)		5 ft	33 ft	40-100° F	14.7 psia	41' (Floor)

ATTACHMENT TO TABLE 6.2-13

SAMPLE CALCULATION FOR HEAD LOSS DUE TO FRICTION TO VERIFY LOSS

The Sample NPSH Calculation for Head Loss Due to Friction To Verify NPSH for the CS Pumps has been Superseded by IP3-CAL-CS-02590, "Containment Spray Pump NPSH Review."

Ref: Perry's Chemical Engineering Handbook
4th Edition, page 603, equation 6-11

$NPSH_A = h_{ss} - h_{fs} - p$

h_{ss} = static suction head = vertical distance between free level of source of supply and pump suction center line plus absolute pressure at the free level.

h_{fs} = friction loss in suction line between source of supply and pump suction.

p = vapor pressure of liquid at pumping temperature

Example: Containment Spray Pumps – two pumps operating @3000 gpm each

6.3 CONTAINMENT SPRAY SYSTEM

6.3.1 Design Bases

The General Design Criteria presented and discussed in this section are those which were in effect at the time when Indian Point 3 was designed and constructed. These general design criteria, which formed the bases for the Indian Point 3 design criteria, were published by the Atomic Energy Commission in the Federal Register of July 11, 1967, and subsequently made a part of 10 CFR 50.

The Authority has completed a study of compliance with 10 CFR Parts, 20 and 50 in accordance with some of the provisions of the Commission's Confirmatory order of February 11, 1980. The detailed results of the evaluation of compliance of Indian Point 3 with the General Design Criteria presently established by the Nuclear Regulatory Commission (NRC) in 10 CFR 50 Appendix A, were submitted to NRC on August 11, 1980, and approved by the Commission on January 19, 1982. These results are presented in Section 1.3.

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 of 7/11/67)

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 discussed in 6.3.2, and the Containment Air Recirculation Cooling and Filtration 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 of 47 psig at 271 F (100% R. H.). This protection is afforded for all pipe break sizes up to and including the hypothetical instantaneous circumferential rupture of a reactor coolant loop. Pressure and temperature transients for a Loss-of-Coolant Accident are presented in Chapter 14. Although the water in the core after a Loss-of-Coolant Accident is quickly sub-cooled by the Safety Injection System, the Containment Spray System design was 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 post-accident containment pressure below the design value, assuming that the core residual heat is released to the Containment as steam:

- Both containment spray pumps (and one of the two spray valves in the recirculation path)
- All five containment cooling fans (discussed in Section 6.4)
- One containment spray pump and any three out of the five containment cooling fans.

Inspection of Containment Pressure Reducing System

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 of 7/11/67)

Where practicable, all active components and passive components of the Containment Spray System are inspected periodically to demonstrate system readiness. The pressure retaining 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 are inspected for leaks. Design provisions for inspection of the Safety Injection System, which also functions as part of the Containment Spray System, are described in Section 6.2.5.

Testing of 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 of 7/11/67)

All active components in the Containment Spray System were adequately tested both in pre-operational performance tests in the manufacturer's shop and by in-place testing after installation. Thereafter, periodic tests are also performed after any component maintenance. Testing of the components of the Safety Injection System which are used for containment spray purposes is described in Section 6.2.5.

The component cooling water pumps and the conventional service water pumps which apply the 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 are tested periodically by changing the operating pump(s).

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 or 7/11/67)

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

The air test lines for checking that spray nozzles are not obstructed, connect downstream of the isolation valves. Air flow through the nozzles is monitored by means of the helium filled balloon method, or other suitable methods that can demonstrate that nozzles are not clogged.

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 system into action, including the transfer to alternate power sources. (GDC 61 of 7/11/67)

Capability was provided to test initially, to the extend practical, the operational start-up sequence of the Containment Spray System including the transfer to alternate power sources.

Performance Objectives

The Containment Spray System was designed to spray at least 5200 gpm of borated water, [deleted] into the Containment whenever the coincidence of two sets of two-out-of-three (high-high) containment pressure (approximately 50% of design value) signals occurs or a manual signal is given. Either of two subsystems containing a pump and associated valving and spray header are independently capable of delivering more than one-half of the design delivery flow, or at least 2600 gpm, based on a pump design flow of at least 2600 gpm at a containment back pressure of 47.0 psig. Actual flow is reduced by up to 150 gpm due to recirculation flow through the eductor resulting in a delivered flow of 2450 gpm per pump at a containment back pressure of 47.0psig. In the recirculation line only the eductor motive flow path is active as the spray additive flow path was isolated due to retirement of the NaOH spray additive system by License Amendment 236.

The design basis was to 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 airborne iodine from the containment atmosphere should it be released in the event of a Loss-of-Coolant Accident. The analysis showing the system's ability to limit offsite dose to within 10 CFR 100 limits after a hypothetical Loss-of-Coolant Accident is presented in Chapter 14. If all engineered safety features operate at design capacity, offsite doses will be limited to within the limits of 10 CFR 20.

The Containment Spray System was 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 post-accident pressure and consequent containment leakage.

Portions of other systems, which share functions and become part of the Containment Spray System when required, were designed to meet the criteria of this section. 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).

System piping located within the Containment is redundant and separable in arrangement unless fully protected from damage which may follow any Reactor Coolant System loop failure.

System isolation valves relied upon to operate for containment cooling are redundant, with automatic actuation.

Service Life

All portions of the system located within the Containment were designed to withstand, without loss of functional performance, the post-accident containment environment and operate without benefit of maintenance for the period of time needed to restore and maintain containment conditions at near atmospheric pressure.

Codes and Classifications

Table 6.3-1 tabulates the codes and standards to which the Containment Spray System components were designed.

6.3.2 System Design and Operations

System Description

Adequate containment cooling and iodine removal are provided by the Containment Spray System, shown in Plant Drawings 9321-F-27353 and 27503 [Formerly Figures 6.2-1A and –B], whose components operate in sequential modes. These modes are:

a) Spray a portion of the contents of the Refueling Water Storage Tank into the entire containment atmosphere using the containment spray pumps. During this mode, the spray stream removes iodine from the containment atmosphere by a washing action.

b) 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 Chapter 14.

The principal components of the Containment Spray System, which provides containment cooling and iodine removal following a Loss-of-Coolant Accident consist of two pumps, sodium tetraborate baskets located in containment, spray ring headers, nozzles, and the necessary piping and valves. The containment spray pumps [deleted] 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 utilizes the two 100% capacity recirculation pumps, two residual 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.

Injection Phase

The spray system will be actuated by the coincidence of two sets of two-out-of-three high-high level containment pressure signals. This starting signal will start the pumps and open the discharge valves to the spray headers. If required, the operator can manually actuate the entire system from the Control Room, and periodically, the operator will actuate system components to demonstrate operability. Additionally, two current indicators, one per containment spray pump motor, monitor their operation once they are started.

Any “P” signal received without an accompanying “T” (i.e., “S”) signal is considered to be a spurious “P” signal. The “P” signal is actuated by the Phase B, (high-high level) high containment pressure; the “T” (i.e., “S”) signal is actuated by the Phase “A” (high level) high containment pressure.

In order to terminate and reset the containment spray system, the following steps need to be performed:

1) Depress both <u>CS RESET Buttons</u>;
2) Stop the operating CS Pumps(s);
3) Close the opened discharge valve(s) (866A/866B), and

To meet the containment sump conditions, the changeover from the injection phase to recirculation phase will be initiated after the injection of approximately 195,800 gallons of the Refueling Water Storage Tank capacity. Recirculation will commence after the changeover is completed and the remaining available 95,800 gallons will be injected into the Containment via one containment spray pump. [Deleted] This 95,800 gallon volume allows for operator action to complete the transfer to recirculation before the RWST is completely depleted. Final emptying of the RWST can be accomplished via the Containment Spray Pumps after recirculation has begun.

Sodium tetraborate is stored at elevation 46” inside the containment building. During the injection phase the level of the boric acid solution from containment spray, safety injection and the coolant lost from the reactor coolant system will rise above the sodium tetraborate baskets. The sodium tetraborate will dissolve into the solution, increasing the solution pH above 7.0 to enhance long-term iodine retention in the solution and to minimize corrosion.

<u>Recirculation Phase</u>

When the Refueling Water Storage Tank is exhausted, or sufficient sump level is obtained recirculation spray flow will be initiated. The operator can remotely open the stop valves on either of the two spray recirculation lines. Throttle valves in the injection lines to the core split the recirculation flow so that at least 662 gpm is delivered to the core and the remainder to the spray headers. With this split flow, decay heat can be removed by boil-off and the containment pressure maintained below design.

After the four-hour containment scrubbing operations, it is expected that spray flow could 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 exchanger once the sump water becomes heated. The heat removal capacity of three-out-of-five fan coolers is sufficient to remove the corresponding energy addition to the vapor space resulting from steam boil-off from the core, assuming flow into the core from one recirculation pump at the beginning of recirculation, without exceeding containment design pressure; hence, it is not expected that continued spray operation for containment heat removal would be required. If, however, the containment pressure was observed to increase, then recirculation to the spray header may be resumed by operator action as described above.

Cooling Water

The cooling water supply for the residual heat exchangers is discussed in Section 6.2.

Change-Over

The sequence for the change-over from injection to recirculation is also discussed in Section 6.2.

Remotely operated valves of the Containment Spray System which are under manual control (that is, valves which 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 this is shown visually on the board. In addition, an audible annunciation alerts the operator as to the condition.

Charcoal Filter Dousing

A dousing system is provided for the carbon filter bank of each fan cooler unit of the Containment Air Recirculation Cooling and Filtration System. Each dousing system can be supplied with water from the containment spray headers as shown in Plant Drawing 9321-F-27353 [Formerly Figure 6.2-1A]. The dousing system was designed to be started manually by the operator following indication of a fire in a carbon filter bank if high temperature conditions were to occur as a result of a failure of a fan. Further details of the dousing system are given in Section 6.4

Prior to initial operation, the lines connected to the Containment Spray Pumps were cleaned by means of a temporary strainer located on the suction side of the containment spray pumps. This insured an adequate clean supply of water was available prior to initial plant operation. These pump strainers were subsequently removed. During plant operation, the Refueling Water Storage Tank (RWST) can be purified by means of the RWST Purification System. This is permitted under administrative controls (i.e., an operator familiar with the operational restrictions of the RWST Purification System who is in contact with the Control Room). This system is connected to Residual Heat Removal (RHR) Pump suction line. The RWST is purified by pumping the water through the Spent Fuel Pit Demineralizer and the Spent Fuel Pit Filter before returning it to the RWST. Also, the RWST Purification pump suction piping originally contained a strainer which was subsequently removed. Thus, the water for the Containment Spray System, therefore, will be of adequate quality to prevent clogging of the spray nozzles for the 1) Fan Cooler Unit carbon filter dousing system and 2) the spray ring headers located near the containment building dome. No other provisions are provided for filtering the RWST water. Drain valves are also provided upstream of the carbon filter isolation dousing valves to prevent the unwanted passage of water from entering into the carbon filters during isolation valve testing.

During the post-LOCA recirculation mode of Safety Injection, Residual Heat Removal, and Containment Spray Systems operation, there are no provisions for filtering the water supplied to the carbon filter dousing system. While it is possible that one or more dousing nozzles could become clogged, a filter is not used for the following reasons:

a) The dousing system is placed into operation manually, only after indication of a fire.

b) A filter in the dousing water supply pipe to any of the five carbon filter units could itself become clogged and shut off all dousing water to that unit.
c) Within each carbon filter unit, there is a redundancy of spray nozzles since there is some overlapping of sprays.
d) Should one spray nozzle become clogged and the nearby carbon over- heats, a local fire could not spread since surrounding charcoal is being doused.

Components

All associated components, piping, structures and power supplies of the Containment Spray System were designed to seismic Class I criteria.

All components inside containment are capable of withstanding or are protected from differential pressures which may occur during the rapid pressure rise to 47 psig in ten (10) seconds. 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

Pumps

The two containment spray pumps are of the horizontal, centrifugal type driven by electric motors. The design head of each 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. Each pump motor is direct-coupled and large enough to meet the maximum power requirements of the pump. Their operation is monitored via the current indicators in the Control Room. The materials of construction suitable for use with the spray solution 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 with 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 Chapter 9.

Heat Exchangers

The two residual heat exchangers of the Safety Injection System, which are used during the recirculation phase, are described in Section 6.2

Spray Nozzles

The spray nozzles, which are of the hollow cone, ramp bottom design, are not subject to clogging by particles ¼ inch or less 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 cylindrical plate strainer top-hat modules before leaving the Recirculation or the Containment Sump. The spray nozzles are stainless steel and have a 3/8 inch diameter orifice. The nozzles are connected to four 360 degree ring headers (alternating headers connected) of radii 8' 2" (El. 228.5'), 25' 4" (El. 223.5'), 42' 3" (El. 218.5'), and 59' 6" (El. 213.5'). There are 315 nozzles distributed on the four headers. These nozzle and header arrangements result in maximum area coverage with either branch of the system operating alone, while assuring minimum overlap of spray trajectories in the minimum flow case. (See Chapter 14)

Spray Additive Tank

The spray additive tank was retired in place based on the use of sodium tetraborate 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 sodium hydroxide (NaOH) to sodium tetraborate by License Amendment 236 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.

Spray Additive Eductors

In the original plant design, the means of adding NaOH to the spray liquid was provided by a liquid jet eductor, a device which uses the kinetic energy of a pressure liquid to entrain another liquid, mix the two, and discharge the mixture against a counter pressure. The pressure liquid in this case was the spray pump discharge flow through the recirculation line of the pumps which entrained the NaOH solution and discharged the mixture into the suction of the spray pumps. The two eductors were designed to provide a spray pH between the limits stated in Section 1.3.1 of Appendix 6D during the injection phase. In the current plant design, the spray additive tank is isolated such that only the eductor motive flow path (through the pump recirculation line) is active and the eductors are no longer credited for pH control.

[Deleted]

PH adjustment of the Emergency Core Cooling Sump solution to a maximum pH of 7.4 and a minimum pH of 7.1 (See Appendix 6D) is initially performed by the sodium tetraborate baskets located in containment.

There is a sampling line on the discharge of the recirculation pumps that permits periodic remote sampling of the sump fluid. The primary means for adjusting the sump pH, following the LOCA, is through the use of the chemical mixing tanks and the charging pumps.

Valves

The valves for the Containment Spray System were designed in accordance to the specifications discussed for the valves in the Safety Injection System (Section 6.2).

Piping

The piping for the Containment Spray system was designed in accordance to the specifications discussed for the piping in the Safety Injection System (Section 6.2).

The system was designed for 150 psig at 300 F on the suction side and 300 psig at 300 F on the discharge side of the spray pumps.

Motors for Pumps and Valves

The motors inside and outside containment for the Containment Spray System were designed in accordance with the specifications discussed for motors in the Safety Injection System. (See Section 6.2)

Electrical Supply

Details of the normal and emergency power sources are presented in the discussion of the Electrical Systems, Chapter 8.

Environmental Protection

The spray headers are located outside and above the reactor and steam generator concrete shield. Another 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.

Material Compatibility

Parts of the system in contact with the spray solution are stainless steel or an equivalent corrosion resistant material.

All exposed surfaces within the Containment have coatings which are not subject to interaction under exposure to the containment spray solution, with the exception of small amounts of aluminum associated with the nuclear flux instrumentation.

An evaluation of materials compatibility with the long term storage conditions of the original containment spray additive (NaOH) is given in Appendix 6E. Appendix 6E is retained for historical purposes.

Post-accident chemistry changes due to the elimination of the spray additive tank and the installation of sodium tetraborate baskets were evaluate and it was determined that this change has little effect on the compatibility of materials located in containment, that come in contact with the initial spray and recirculation spray solution. This evaluation is documented in WCAP-16596-NP. An analysis of the materials exposed to the post-accident containment environment using the original containment spray additives (NaOH) solution is presented in Appendex 6D. Appendix 6D has been updated where appropriate to include post-accident buffer change to sodium tetraborate and much of the information for the original NaOH additive has been retained for historical purposes.

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. These chemistry changes with sodium tetraborate do not affect the environmental qualifications of equipment located within the containment required to mitigate the consequences of design basis accidents as documented in Appendix 6D, Section 9.0 and Appendix 6F, Section 5.0.

Sodium Tetraborate Baskets

Sodium tetraborate is stored in eight baskets at elevation 46' in the containment building. During the injection phase the baskets will be flooded, allowing the sodium tetraborate to dissolve into the fluid for pH control. The eight baskets are constructed of stainless steel and are seismically qualified and mounted.

6.3.3 Design Evaluation

Range of Containment Protection

For up to the first 25 minutes following the maximum Loss-of-Coolant Accidents (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 spray into the Containment for up to an additional thirty minutes. [Deleted] 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; flows are adjusted so that sufficient cooled recirculated water is delivered to keep the core flooded as well as providing flow to one spray header. Flow is sufficient to maintain the containment pressure below the design value, if required.

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:

- Both containment spray pumps (and one of the two spray valves in the recirculation path)
- All five containment cooling fans (discussed in Section 6.4)
- One containment spray pump and any three out of the five containment cooling fans.

For design basis accidents in which failure of any single diesel generator is assumed, the resulting equipment configuration is also adequate to satisfy containment cooling requirements.

During the injection and recirculation phases, the spray water is raised to the temperature of the Containment by 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. Heat transfer calculations, based upon 1000 micron droplets, show that thermal equilibrium is reached in a distance of approximately five feet. Thus the spray water reaches essentially the saturation temperature.

At containment design pressure (47 psig), at least 2450 gpm of borated water is injected into the containment atmosphere by one spray pump. At containment design temperature (271°F), the total heat absorption capability of one spray pump is approximately 205 x 10^6 Btu/hr based on the addition of 110^o F refueling water. The IP3 Stretch Power Uprate analysis were performed using 110^{o}F refueling water.

When recirculation is initiated, approximately 95,800 gallons of refueling water is left available in the Refueling Water Storage Tank for spray pump usage. This supply is reserved to allow switchover to recirculation pumps and to provide containment pressure relief. When the Refueling Water Storage Tank is empty, or the sufficient sump level is obtained, the recirculation pumps supply the flow to the spray headers. Spraying of water from the sump into the containment atmosphere with one recirculation pump, after cooling to 134.7°F with a residual heat exchanger, results in a heat removal rate of 1.63 x 10^8 Btu/hr at design temperature. This heat removal balances decay heat after 5000 seconds. The prior 2 sentences are a description of the capability of the original system design, at the original plant operating conditions. The performance of the Containment Spray System (at current operating conditions) in containment pressure reduction is discussed in Chapter 14.

In addition to heat removal, the spray system is effective in scrubbing fission products from the containment atmosphere. However, quantitative credit is taken only for absorption of reactive and/or soluble forms of inorganic iodine in the analysis of the hypothetical accident (Section 14.3). A discussion of the effectiveness of containment spray as a fission product trapping process is contained in Appendix 6A. 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 containment.

Any of the combinations of equipment (spray pumps and fans) required for containment heat removal will provide sufficient iodine trapping capability to ensure that post-accident fission product leakage (based on Alternate Source Term) does not result in exceeding the dose limits of 10 CFR 100. This is evaluated in Section 14.3.

<u>System Response</u>

The starting sequence of the containment spray pumps and their related emergency power equipment was designed so that delivery of the minimum required flow is reached in 43 seconds.

The starting sequence is:

Sequence	Seconds (max)
1) Initiation of safety injection signal, including instrument lag*	2
2) Starting of emergency diesel generators	10
3) Starting of containment spray pumps	8 or 13
4) Acceleration and pipe fill time	32
Total – from event initiation	52 or 57

NOTE: If no LOOP, subtract 10 seconds.

Motor control centers are energized and valves are opened at the same time as the pumps are started. As described in Section 14.3, a delay of 60 seconds is assumed for the starting of the containment spray.

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 of one of the two residual heat removal heat exchangers. As described in Section 6.2.3, 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 Chapter 14 reflects the single failure analysis.

*NOTE: To reduce inadvertent Safety Injection System Actuation due to instrumentation lags in the engineered safeguards system high steamline flow, low average temperature T_{avg}/Low steamline pressure coincidence circuitry, a time delay will be installed in each train (a maximum time delay of 6 seconds will meet the acceptance criteria for a steam line rupture).

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 to the Containment Air Recirculation Cooling and Filtration System for both the cooling and iodine removal functions. 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 exchangers may be diverted to the containment spray headers or the high head safety injection pumps. Minimum flow requirements are set for the flow being sent to the core and for the flow being sent to the containment spray headers such that at least 662 gpm is sent to the core. Sufficient flow instrumentation is provided so that the operator can perform appropriate flow adjustments with the remote throttle valves in the flow path as shown in Plant Drawing 9321-F-27503 [Formerly Figure 6.2-1B].

Normal and emergency power supply requirements are discussed in Chapter 8.

Shared Function Evaluation

Table 6.3-5 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.

Containment Spray Pump NPSH Requirements

The NPSH for the containment spray pumps is evaluated for injection operation. The beginning fill-up period of the injection phase gives the limiting NPSH requirements. The NPSH available is determined from the elevation head and vapor pressure of the water in the RWST and the pressure drop in the piping to the pump. Sufficient NPSH margin is available to prevent cavitation of the CS pumps under all operating conditions.

6.3.4 Minimum Operating Conditions

The Technical Specifications establish limiting conditions regarding the operability of the system when the reactor is above MODE 5.

6.3.5 Inspections and Tests

Inspections

All components of the Containment Spray System are inspected periodically to demonstrate system readiness.

The pressure containing systems are 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 are inspected for leaks.

Pre-Operational Testing

Offsite Work

These components in the system were subjected to offsite test work:

a) Spray pumps
b) Spray nozzles
c) Eductors

The spray pumps were subjected to conventional acceptance tests and the performance characteristics plotted to illustrate that the pumps met the design specification.

As part of the development work in support of Westinghouse plant equipment, a nozzle of the type used in the spray system was subjected to a performance test to demonstrate and prove the nozzle characteristics (e.g., flow/pressure drop, droplet size, spread of spray, etc.).

As part of the quality assurance program, a random 25% of the nozzles installed at the Indian Point 3 site were given a general performance test.

The eductors were produced and tested in two stages.

A prototype was made to check nozzle calculations prior to manufacture of the stainless steel units

A performance test was made by the manufacturer on one of the finished stainless steel units to confirm the capacity at the specified conditions. A sugar-water solution was used to simulate the 30% sodium hydroxide suction fluid.

Onsite Test Work

The aim of onsite testing was to:

Demonstrate and prove that the system is adequate to meet the design pressure conditions; outside the Containment this involved part radiographic inspection and part hydro-testing; inside the Containment the spray headers were subjected to 100% radiographic inspection

Demonstrate that the spray nozzles in the containment spray header are clear of obstructions by passing air through the test connections

Verify that the proper sequencing of valves and pumps occurred on initiation of the containment spray signal and demonstrate the proper operation of all remotely operated valves

Verify the operation of the spray pumps; each pump was run at shut-off 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

Demonstrate the operation of the spray eductors. The eductor and spray additive system were 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 drain down 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 flow for routine testing of the system, the above was made with the spray additive tank isolated and the eductor drawing water through the RWST/eductor suction test line.

Operational Testing

The aim of the periodic testing is 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.

Verify the operation of the spray pumps. Each pump is run at shut-off and the mini-flow directed through the normal path back to the Refueling Water Storage Tank.

Demonstrate the operation of the spray eductors. Periodic testing of the spray eductors is no longer required based on retirement of the spray additive tank and replacement of the sodium hydroxide buffer with sodium tetraborate buffer by License Amendment 236. The remaining motive portion of the spray eductor is tested as part of the spray pump miniflow line during spray pump testing.

The operational testing of the Safety 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.

TABLE 6.3-1

CONTAINMENT SPRAY SYSTEM-CODE REQUIREMENTS

Component	Code
[Deleted]	
Valves	ANSI B16-5 (1955)
Piping (including headers and spray nozzles)	ANSI B31.1 (1955)

TABLE 6.3-2

CONTAINMENT SPRAY SYSTEM DESIGN PARAMETERS

PUMPS

Quantity	2
Design pressure, discharge, psig	300
Design pressure, suction, psig	300
Design temperature, F	150
Design flow rate, gpm	2600
Design head, ft	450
Maximum flow rate, gpm	3154
Shutoff head, ft	490
Motor HP	400
Type of Pump	Horizontal-Centrifugal

EDUCTORS

Quantity	2
Eductor Inlet (motive)	Injection Phase
Operating Fluid	Water (with >2400 but <2600 ppm boron)
Operating Pressure, psig	195
Operating Temperature	Ambient
Flow Rate, gpm	112 (design), $\leq$150(analyzed)
Discharge Head (including static pressure, friction loss, and discharge elevation), psig	0.4 to 16.5

Eductor Suction (Inactive – SAT flow path was isolated due to retirement of the NaOH spray additive system)

TABLE 6.3-3

[Deleted]

TABLE 6.3-4

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 as incredible.
B. Pumps		
Containment Spray Pump	Fails to start	Two provided. Evaluation based on operation of one pump in addition to three out of five con-tainment cooling fans operating during injection phase.
Recirculation Pump	Fails to start	Two provided. Evaluation based on operation of one pump and no containment cooling fans operat-ing during recirculation phase.
Conventional and Nuclear Service Water Pumps	Fails to start	Six provided. Operation of three pumps during recirculation required.
Component Cooling Pumps	Fails to start	Three provided. Operation of two pumps during recirculation required.
Auxiliary Component Cooling Pump	Fails to start	Four provided. Two required to operate.
Automatically Operated Valves: (Open on coincidence of two – 2/3 high containment pressure signals)		
1. Containment Spray Pump Discharge Isolation Valve (Valves 866A & 866B)	Fails to open	Two parallel path, each with one pump and one valve are provided. Operation of one path is 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.

(valves 822A & 822B)		
Valves Operated from Control Room for Recirculation		
1) Recirculation isolation (valves 1802A & 1802B)	Fails to open	Two valves in parallel, one valve is required to open.
2) Containment spray header isolation valve from residual heat exchangers (valves 889A & 889B)	Fails to open	Two valves provided. Operation of one required.
3) Residual heat removal pump recirculation online (valves 743 & 1870)	Fails to close	Two valves in series, one required to close.
4) Residual heat removal pump discharge line (valve 744 and check valve 741)	Fails to close	Two valves in series, one required to close (one valve is a check valve).

TABLE 6.3-4
(Cont.)

SINGLE FAILURE ANALYSIS – CONTAINMENT SPRAY SYSTEM

Component	Malfunction	Comments and Consequences
Automatically Operated Valves (Close from the Control Room on injection to recirculation changeover)		
1) Isolation valves at spray pump discharge (motor operated valves 866A & 866B, check valves 867A & 867B, and manual valves 869A & 869B)	Fails to close	Check valve in series with one motor operated valve provided for each line. In addition, a manually operated isolation valve with IVSWS is provided in each line.
Valves Operated from Control Room for Charcoal Filter Dousing		
Isolation valves at filter unit (valves 880A & B, 880C & D, 880E & F, 880G & H, 880J & K)	Fails to open	Two valves provided for each of the five units. Operation of one valve per unit required.

6.3-5

SHARED FUNCTIONS EVALUATION

Component	Normal Operating Function	Normal Operating Arrangement	Accident Function	Accid
[Deleted]				
Containment Spray Pumps (2)	None	Lined up to spray headers	Supply spray water to containment atmosphere	Lined heade

NOTE: Refer to Section 6.2 for a brief description of the Refueling Water Storage Tank, recirculation pumps, conventional service water pumps, component cooling pump, 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.

6.4 CONTAINMENT AIR RECIRCULATION COOLING AND FILTRATION SYSTEM

6.4.1 Design Bases

The General Design Criteria presented and discussed in this section are those which were in effect at the time when Indian Point 3 was designed and constructed. These general design criteria, which formed the bases for the Indian Point 3 design, were published by the Atomic Energy Commission in the Federal Register of July 11, 1967, and subsequently made a part of 10 CFR 50.

The Authority has completed a study of compliance with 10 CFR Part 20 and 50 in accordance with some of the provisions of the Commission's Confirmatory Order of February 11, 1980. The detailed results of the evaluation of compliance of Indian Point 3 with the General Design Criteria presently established by the Nuclear Regulatory Commission (NRC) in 10 CFR 50 Appendix A, were submitted to NRC on August 11, 1980, and approved by the Commission on January 19, 1982. These results are presented in Section 1.3.

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 of 7/11/67)

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 Section 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.

The Containment Air Recirculation Cooling and Filtration System was designed to recirculate and cool the containment atmosphere in the event of a Loss-of-Coolant Accident and thereby ensures that the containment pressure will not exceed its design value of 47 psig at 271 F (100% 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 and Filtration System was designed 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 post-accident 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) Both containment spray pumps (and one of the two spray valves in the recirculation path)
3) Any three out of the five containment cooling fans and one of the containment spray pumps

For design basis accidents in which failure of any single diesel generator is assumed, the resulting equipment configuration is also adequate to satisfy containment cooling requirements.

Inspection of Containment Pressure-Reducing System

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 of 7/11/67)

Design provisions were made to facilitate access for periodic visual inspection of all the important components of the Containment Air Recirculation Cooling and Filtration System.

Testing of Containment Pressure-Reducing Systems Components

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 of 7/11/67)

The Containment Air Recirculation Cooling and Filtration System was designed so that the components can be tested periodically and, after any component maintenance, testing can be conducted for operability and functional performance.

The air recirculation and cooling units, and the service water pumps, which supply the cooling units, are in operation on an essentially continuous schedule during plant operation, and no additional periodic tests are required.

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 of 7/11/67)

Means were provided to test initially the full operational sequence of the air recirculation system, including transfer to alternate power sources.

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 systems, such as ducts, filters, fans, and dampers (GDC 62 of 7/11/67)

Access is available for periodic visual inspection of the Containment Air Recirculation Cooling and Filtration System components, including fans, cooling coils, dampers, filter units and ductwork.

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 performance. (GDC 63 of 7/11/67)

The charcoal filters of the filtration system are bypassed during normal operation by closed dampers. The dampers in a non-operating unit can be periodically tested by actuating the controls and verifying deflection by instruments in the Control Room. Since the fans are normally in operation, no additional periodic fan tests are necessary.

Testing Air Cleanup Systems

Criterion: A capability shall be provided to the extent practical for insite periodic testing and surveillance of the air cleanup systems to ensure (a) filter bypass paths have not developed and (b) filter and trapping materials have not deteriorated beyond acceptable limits. (GDC 64 of 7/11/67)

Representative sample elements in each of the activated carbon filter plenums are removed periodically during shutdowns and tested on the site to verify their continued efficiency. After reinstallation, the filter assemblies are tested in place by aerosol injection to determine integrity of the flow path.

Testing of Operational Sequence of Air Cleanup Systems

Criterion: A capability shall be provided to test initially under conditions as close to design as practical, the full operational sequence that would bring the air cleanup systems into action, including the transfer to alternate power sources and the design air flow delivery capability. (GDC 65 of 7/11/67)

Means were provided to test under conditions of close to design as practicable the full operational sequence that would bring the Containment Air Recirculation Cooling and Filtration System into action, including transfer to the emergency diesel generator power source.

Performance Objectives

The Containment Ventilation System, Chapter 5, which all of the components of the Containment Air Recirculation Cooling and Filtration System (with the exception of the moisture separators, HEPA filters and charcoal filters) are a part of, was 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. The fans and cooling units continue to remove heat after the Loss-of-Coolant Accident and reduce the containment pressure to near atmospheric pressure within the first 24 hours after the accident.

A second function of the Containment Air Recirculation Cooling and Filtration System is to remove fission products from the containment atmosphere should they be released in the event of an accident. The filtration capacity of the system is sufficient to reduce the concentration of fission products in the containment atmosphere following a loss of reactor coolant, to levels ensuring that the 2 hour and 30 day radiation (TEDE) doses will be limited to within the guidelines of 10 CFR 100 limits. Details of the site boundary dose calculation are given in Chapter 14 along with the equipment configurations resulting from a presumed loss of one diesel generator.

The air recirculation filtering capacity used to satisfy the original design basis, was determined for the following conditions:

1) Containment leak rate of 0.1% per day for 24 hours and 0.05% per day after 24 hours

2) Conservative meteorology corrected for building wake effects

3) A 70% efficiency for filtration of organic iodine. (This assumes credit for the demonstrated ability to filter organic forms of iodine at high relative humidity with impregnated charcoal.)

4) Fission product release to the Containment per TID 14844 at a power level of 3216 MWt. This assumes no credit for safety injection in limiting fission product release.

5) Partial effectiveness of the filtration equipment. This assumes two of the five installed carbon filter assemblies are unavailable at the time of the loss of coolant.

In addition to the design bases specified above, the following objectives were met to provide the engineered safety features functions:

1) Each of the five fan-cooler units is capable of transferring heat at the rate of 49.0 x 10^6 Btu/hr from the containment atmosphere at the post-accident design conditions, i.e., a saturated air-steam mixture at 47 psig and 271° F. This heat transfer rate is that assigned to the fan-cooler units in the analysis of containment and related heat removal system capability in Chapter 14.

   The establishment of basic heat transfer design parameters for the cooling coils of the fan-cooler units, and the calculation by computer of the overall heat transfer capacity are discussed in Chapter 14. 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 post-accident 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 and high efficiency particulate air (HEPA) filters rated for 8000 cfm unit flow. The latter are capable of 99.97% removal efficiency for 0.3 micronparticles at the post-accident conditions.

4) Each of the five air handling units is capable of supplying air to separate carbon-bed filter units following an accident for fission product iodine removal. The design flow rate through each air handling unit is 69,500 cfm during normal operation and 34,000 cfm during accident conditions. The design flow rate through each carbon filter assembly is 8,000 cfm, at a face velocity of approximately 50 fpm. The remainder of

the flow bypasses the filter assemblies. The carbon filter units are designed to remove at least 70% of the incident radioactive iodine in the form of methyl iodide (CH_3I). These are the iodine removal efficiencies assumed in the analysis of containment capability to retain fission product iodine under the post-accident design conditions in Chapter 14.

In addition to the above design bases, the equipment was designed to operate at the post-accident conditions of 47 psig and 271° F for three hours, followed by operation in an air-steam atmosphere at 20 psig, 219° F for an additional 21 hours. The equipment design will permit subsequent operation of an air-steam atmosphere at 5 psig, 152 F for an indefinite period. See Appendix 6F for details of the IP3 Equipment Qualification Program.

All components are capable of withstanding or are protected from differential pressures which may occur during the rapid pressure rise to 47 psig in ten (10) seconds.

Portions of other systems which share functions and become part of this containment cooling system when required were designed to meet the criteria of this section. 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).

Where portions of these systems are located outside of containment, the following features were incorporated in the design for operation under post-accident conditions:

a) Means for isolation for any section
b) Means to detect and control radioactivity leakage into the environs, to the limits consistent with guidelines set forth in 10 CFR 100.

6.4.2 System Design and Operations

The flow diagram of the Containment Air Recirculation Cooling and Filtration System is shown on Plant Drawing 9321-F-40223 [Formerly Figure 6.4-2].

Individual system components and their supports meet the requirement for Class I (seismic) structures and each component is mounted to isolate it from fan vibration.

Containment Cooling System Characteristics

The air recirculation system consists of five 20% capacity air handling units, each including motor, fan, cooling coils, moisture separators, HEPA filters, carbon filters with spray and fire detection, dampers, 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 moisture separators, HEPA filters and activated carbon filter assembly is normally isolated from the main air recirculation stream. Part of the air flow (air-steam mixture) is bypassed through the filtration section of the units (moisture separators, HEPA filters, and carbon filter assembly) to remove volatile iodine following an accident.

Each fan was designed to supply 69,500 cfm at approximately 6.3" s.p. (0.075 lb/ft^3 density) during normal operation and 34,000 cfm, at approximately 8.6" s.p. (0.175 lb/ft^3 density), during accident operation.

The fans are direct driven, centrifugal type, and the coils are plate fintube type. Each air handling unit is capable of removing 49.0 x 10^6 Btu/hr from the containment atmosphere under accident conditions. A flow of 1400 gpm of service (cooling) water is supplied to each unit during accident conditions. The design maximum river water inlet temperature is 95^{o}F.

Duct work distributes the cooled air to the various containment compartments and areas. During normal operation, the flow sequence through each air handling unit is as follows; inlet dampers, cooling coils, fan, discharge header. Roughing filters were installed up-stream of the cooling coils during plant clean-up. Any time these filters are used, they must be removed prior to exceeding cold shutdown. In lieu of using roughing filters, cooling coil thermal performance is assured by alternate means. (Reference Generic Letter 89-13)

In the event of an accident, the flow is split into two parts: a minimum of 8000 cfm passes through the filtration section consisting of moisture separation, HEPA filters, and carbon filter assembly, and the remainder of the flow bypasses the filtration section of the units and passes through the cooling coils with the filtration flow. The bypass flow control is accomplished via a damper that fails closed to a pre-set position for accident operation.

Plant Drawing 636F269 [Formerly Figure 6.4-1] is an engineering layout drawing of an air handling unit, showing the arrangement of the above components in the unit. Plant Drawing 9321-F-40253 [Formerly Figure 6.4-3] shows the location of the five units on the intermediate floor (elevation 68' -0").

Actuation Provisions

A tight closing damper isolates the filtration section of the units from the normal operating components. Upon loss of air pressure to the damper control cylinder, the damper and accident filtration inlet door opens to permit air to flow through this section. The damper and door are fail safe open via weights and spring, respectively.

Upon either manual or automatic actuation of the safety injection safeguards sequence, the accident damper and door are tripped to the accident position. Accident position is also the “fail-safe position.” Electrically operated environmentally qualified three-way solenoid valves are used with the dampers and door to control the instrument air supply (control air).

The containment pressure is sensed by six separate pressure transmitters located outside the Containment. Containment pressure is communicated to the transmitters through three 1" stainless steel lines penetrating the containment vessel. A high containment pressure signal automatically actuates the safety injection safeguard sequence (see Section 6.2.2), which trips the valves to the accident position.

The fans are part of the engineered safety features and either all five or at least three out of the five fans will be started after an accident, depending on the availability of emergency power. (See Chapters 8 and 14)

Overload protection for the fan motors is provided at the switchgear by over-current 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.

Redundant flow switches in the system, operating both normally and post-accident, indicate whether air is circulating in accordance with the design arrangement. Abnormal flow alarms are provided in the Control Room.

Flow Distribution and Flow Characteristics

The location of the distribution ductwork outlets, with references to the location of the air handling unit return inlets, ensures that the air is directed to all areas requiring ventilation before returning to the units. The arrangement is shown in Plant Drawing 9321-F-40223 [Formerly Figure 6.4-2].

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 assure that the discharge air will mix with the atmosphere.

The air discharge inside the periphery of the shield wall circulates and rises above the operating floor through openings around the steam generators where it will mix air displaced from the dome area. This mixture returns to the air handling units through floor grating located at the operating floor directly above each air handling unit inlet. The temperature of this air is 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° F and have a density of 0.175 pounds per cubic foot. Part of the water vapor will condense on the cooling coil, and the air leaving the unit will be saturated at a temperature slightly below 271° F. The fluid also enters the moisture separators at approximately 271° F and saturated (100% R.H.) condition.

The purpose of the moisture separators is to remove the entrained moisture to protect the HEPA Filters from excessive pressures due to water buildup during accident operation. The fluid flows through the HEPA filter and into the carbon filters, and to the cooling coils picking up some sensible heat from the fan and fan motor before flowing through into the distribution header. This sensible heat will increase the dry-bulb temperature slightly above 271° F and will decrease the relative humidity slightly below 100%.

With a flow rate of 34,000 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 4 containment volumes per hour.

Carbon Filter High Temperature Detection and Dousing System

The five carbon filter assemblies are provided with high-temperature detectors and associated alarms in the Control Room. Each carbon filter assembly is also provided with a spray system for water dousing upon an indication of a fire. See Figure 6.4-4 and Section 6.3.2.

Capability for detecting and alarming the presence of fires and localized hot spots in the carbon filters is provided by a system of temperature switches. Each carbon filter plenum (containing one bank of 12 adsorbers) is provided with temperature switches. These switches are uniformly distributed for good coverage. The temperature switches are factory-set for 400° F, (which is below the carbon ignition temperature of 644° F) and they are wired in parallel to a common alarm in the Control Room. Thus closing of a single switch will actuate the alarm to indicate a high temperature condition in the filter plenum.

The water dousing system provided with each carbon filter plenum was designed to drench the adsorbers thoroughly in the extremely unlikely event of a carbon fire during the post-accident recovery. Water for this system is obtained from the main headers of the containment spray system through a separate 2 inch stainless steel line to each filter plenum. There are two normally closed motor operated valves in parallel in each 2 inch line.

The Containment Spray System is automatically actuated and will be running in the event of a Loss-of-Coolant Accident (injection phase). Upon indication of a fire in a filter unit, the operator manually initiates filter dousing by actuating the parallel-connected isolation valves for each filter assembly. Because of the piping arrangement either of the two spray pumps can be used to feed the dousing lines. The dousing flow (approximately 12 gpm per fan cooler unit) was sized to wet the carbon completely and remove the decay heat of the adsorbed iodine thereby preventing heating to the ignition temperature. The system was designed to so containment spray at slightly reduced flow can continue simultaneously with filter dousing. Provisions were made for testing of the dousing nozzles through an air hose connection.

During the recirculation phase of core cooling, operation of the dousing system is the same as above except that water to the spray headers is supplied from the discharge of the residual heat removal heat exchangers.

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 Chapter 9.

The cooling water discharges from the cooling coils to the discharge canal and is monitored for radioactivity by routing a small bypass flow from each through a line monitored by two adjacent-to-line radiation detection assemblies. Note that for a fan cooler unit (FCU) cooling coil failure, assumed to occur concurrently with a large break LOCA, radiological accessibility to identify and isolate the failed FCU will be possible prior to initiation of external recirculation. Upon indication of radioactivity in the effluent, each cooler discharge line is monitored individually to locate the defective cooling coil which when identified would remain isolated, and operation would continue with the remaining units. The service water system pressure at locations inside the Containment is 15 to 20 psig, which is below the containment design pressure of 47 psig. However, since the cooling coils and service water lines are completely closed inside the Containment, no contaminated leakage is expected into these units.

Local flow indication is provided outside containment for service water flow to each cooling unit. Abnormal flow alarms are provided in the Control Room. Service water common inlet temperatures, and all outlet temperatures are displayed at the critical function monitoring system (CFMS).

During normal plant operation, flow through the cooling units is throttled for containment temperature control purposes by a valve on the common discharge header from the cooling units. Two additional independent, full flow, isolation valves open automatically in the event an engineered safeguards actuation signal to bypass the control valve. 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.

Environmental Protection

All system control and instrumentation devices required for containment accident conditions were located to minimize the danger of control loss due to missile damage. Flow switches in the ductwork system, operating both normally and post-accident, indicate whether air is circulating in accordance with the design arrangement. Abnormal flow alarms are provided in the Control Room. Post-accident monitoring of certain parameters are qualified for a post-accident environment.

All fan parts, valve shaft and disc seating surfaces and ducts in contact with the containment fluid are protected against corrosion. The fan motor enclosures, electrical insulation and bearings were designed for operation during accident conditions. See Appendix 6F for qualification testing.

Verification tests under the combined environmental effects of high humidity, pressure, temperature, radiation and applicable chemical concentration of the assembled system (as opposed to tests being performed separately) would require the entire Containment Building or a prototype to be adapted to the test conditions.

No significant information would result from such a test beyond that already obtained from the testing of individual components. The combination of the individual component test results assures the performance of the containment air recirculation system under accident conditions.

All of the air handling units are located on the intermediate floor between the containment vessel and the primary compartment 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.

Components

Moisture Separators

The moisture separators were designed to protect the HEPA Filters from adverse pressures due to water buildup following a Loss-of-Coolant Accident[3]. The water flow rate entering the moisture separators is approximately 0.31 gpm per moisture separator (8 per unit) and the moisture separator effluent has essentially zero moisture content.

Each bank was designed for horizontal air flow, and is composed of 8 elements. Each element or separator is 24 in x 24 in x 4 in (minimum) thick and is mounted in a steel support frame.

A steel drain trough was incorporated for each horizontal tier of separators to collect and remove the water that is recovered from the air stream. 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, air-tight seals were provided between the floor, walls, plenum, and around the perimeter of each moisture separator. The tight seal is accomplished by gaskets, and adhesive which can withstand a temperature of 300°F. The thickness of the gaskets is ¼ in for the separator elements and do not extend into the media area when installed.

The moisture separator elements are of fire resistant construction, and consist of mats of stainless steel wire mesh. Non-stainless steel parts used in the construction are protected against corrosion by painting with one (1) coat each of Carbon Zinc No. 11 and Phenoline 305. The separator element frames are stainless steel.

Roughing Filters

The roughing filters remove the large particles from the air stream before it contacts the cooling coils. The roughing filters were installed during plant clean up. These are efficient for removing large particles. Under normal air flow, they offer a resistance to air flow of approximately 0.2 inches of water.

As in the case for all components of the air handling recirculating system, the bank was designed for horizontal air flow. The filter media is of fire retardant construction composed of a fiber mat support screen and fasteners.

HEPA (Absolute) Filters

The high efficiency particulate air (HEPA) filters are capable of 99.97% removal efficiency for 0.3 micron particles at the post-accident design conditions. All materials of construction of these filters are compatible with the spray solution in the post-accident environment with the conditions they are exposed to where the moisture separators upstream protected the HEPA filters. See Appendix 6D, Section 8.3.

The filter media is made of glass fiber (meets MIL-F-51079) and can withstand the incident ambient steam/air temperature conditions and 100% relative humidity. Filter frames were made of stainless steel. The filters meet MIL-STD-282, MIL-F-51068C, MIL-F051079A, and UL 586.

Fan-Motor Units

The five containment cooling fans are of the centrifugal, non-overloading direct drive type.

Each fan can provide a minimum flow rate of 34,000 cfm when operating against the system resistance of approximately 8.6" s.p. existing during the accident condition (0.175 lb/ft^3 density, a containment pressure of 47 psig, and a temperature of 271° F).

The reactor containment fan cooler motors are Westinghouse, totally enclosed water cooled, 225 horsepower, induction type, 3 phase, 60 cycle 720 rpm, 440 volt with ample insulation margin. Significant motor details are as follows:

a) Insulation

Class F (NEMA rated total temperature 155° C) or Class H (NEMA rated total temperature 180° C. It is impregnated and varnish dipped to give a homogeneous insulation system which is highly impervious to moisture. Internal leads and the terminal box-motor interconnection are given special design consideration to assure that the level of insulation matches or exceeds that of the basic motor system. At incident ambient and load conditions (271° F and 225 hp), the motor insulation hot spot temperature is not expected to exceed 127° C. The Fan Cooler Motors and their lubrication are environmentally qualified for use inside the containment building as documented in their respective EQ files.

b) Heat Exchanger

An air to water heat exchanger is connected to the motor to form an entirely enclosed cooling system. The heat removal capability under LOCA conditions is 110,868 Btu/hr at saturation conditions (271° F, 47 psig). Air movement is through the heat exchanger and is returned to the motor. Two vent valves permit 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 as the containment pressure is reduced by the motor heat exchanger. The drain is piped to the containment cooler drain system.

The motor cooling coils have tubes of AL-6X stainless steel with continuous copper plate-type turbex fins. Water boxes made of 904L stainless steel provide for the water supply and discharge which are common with the containment cooler water system, i.e., supplied from the nuclear service water header. A two pass water flow design counter to the air flow is employed.

c) 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 is provided which will alarm in the Control Room.

Conduit (Connection) Box

The motor leads are brought out of the frame through a seal and into a cast iron sealed explosion-proof type of conduit box.

Factory Tests

In addition to the usual quality control tests which were performed to give assurance that the motors meet design specifications, special tests were performed to demonstrate that insulation margins were built in as specified. The completely wound stators have been given a special high potential test to ground. The stators were immersed in water, meggered, given a high potential test while immersed, and baked. After passing the water tests, the rotor was baked, given a final coating dip and were baked again.

Carbon Filters

The carbon filters were fabricated with stainless steel frames filled with activated carbon, which is tested in accordance with ASTM D3803-1989, per the IP3 response to Generic Letter 99-02. The cell construction insures compacted carbon beds of uniform density and thickness.

The design flow rate through each carbon filter unit is 8,000 cfm, at a face velocity of approximately 50 fpm. These units were designed to remove at least 70% of the incident radioactive iodine in the form of methyl iodine (CH_3I). (1) (2)

Fan Cooling Coils

The coils were fabricated of cooper plate fin vertically oriented on AL6X (Allegheny-Ludlum) tubes. The heat removal capability of the cooling coils is 49.0 x 10^6 Btu/hr per air handling unit at saturation conditions (271°F, 47 psig).

The design internal pressure of the coil is 150 psig at 300 °F and the coils can withstand an external pressure of 47 psig at a temperature of 271°F without damage.

Each recirculating unit consists of eight (8) coil units mounted in two banks of four (4) coils high. These banks are located one behind the other for horizontal series air flow, and the tubes of the coil are horizontal with vertical fins.

Each coil assembly consists of one bank of six row deep coils. Each of the two banks contain four Westinghouse Sturtevant designation WC-36114 (36" high by 114" long) coils. The coils are stacked four high to a bank. The total coil assembly (two banks of coils) is 3½ feet wide. There are 12 rows of tubes in the horizontal flow direction and a total of 96 rows of tubes in the vertical direction. Cooling water flow is 1/3 velocity. Cooling coils have 8.5 fins per inch of tube length. (For normal operation, the coils will remove 2.3 x 10^6 Btu/hr.)

Local flow indication is provided outside containment for service water flow to each cooling unit. Abnormal flow alarms are provided in the Control Room. Service water common inlet temperatures, and all outlet temperatures are displayed at the critical function monitoring system (CFMS). Alarms indicating abnormal service water flow and radioactivity are provided in the Control Room.

The coils are provided with drain, pans and drain piping to prevent flooding during accident conditions. This condensate is drained to the Containment Sump. (See Section 6.7.)

Ducting

The ducts were 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, 3.0 psi. The ducts were designed and are supported to withstand thermal expansion during an accident.

Ducts are of welded and flanged construction. All longitudinal seams were welded. Where flanged joints were used, joints are provided with gaskets suitable for temperatures to 300 °F.

Ducts were constructed of galvanized sheet metal.

Dampers

Dampers are held in their operating position by gravity weight and air cylinders. A leak tight damper prevents leakage of air into the charcoal filter compartment during normal operation thereby preventing carbon deterioration. The damper and blow-in door fail to the open position to assure flow through the carbon filters during the accident condition.

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 and Filtration System is given in Chapter 5.

6.4.3 Design Evaluation

Range of Containment Protection

The Containment Air Recirculation Cooling and Filtration System provides the design heat removal capacity and the design iodine removal capability for the containment following a Loss-of-Coolant Accident assuming that the core residual heat released to the Containment as steam. The system accomplishes this by continuously recirculating the air-steam mixture: 1) through cooling coils to transfer heat from containment to service water, and 2) through activated carbon filters to transfer methyl iodide to the filters from the air-steam mixture.

The performance of the Containment Recirculation Cooling and Filtration System in pressure reduction and iodine removal is discussed in Chapter 14.

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) All five containment cooling fans.
2) Both containment spray pumps (and one of the two spray valves in the recirculation path).
3) Any three out of the five containment cooling fans and one containment spray pump.

For design basis accidents in which failure of any single diesel generator is presumed, the resulting equipment configuration is also adequate to satisfy containment cooling and filtration requirements.

System Response

The starting sequence of the last of the five containment cooling fans (at design conditions five of the fans and one of the nuclear service water pumps operate during normal power operations for containment ventilation) and the related emergency power equipment were designed so that delivery of the minimum required air flow to the carbon filters and cooling water flow is reached in 58 seconds. In the analysis of the containment pressure transient, Section 14.3, a delay time of 50 seconds was assumed.

The starting sequence is:

Sequence	Seconds
1)* Initiation of safety injection signal including instrument lag	2
2) Starting of emergency diesel generators	10
3) Starting of containment cooling fan	15 or 23
4) Acceleration time (estimated)	10
TOTAL – from event initiation	37 or 45

*NOTE: If no LOOP, subtract ten seconds.

The valves are actuated to safeguards position by the safety injection signal.

Single Failure Analysis

A failure analysis was 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 Chapter 14 is consistent with the single failure analysis.

Loss of a fan motor in a unit should not result in ignition of the carbon. Ignition should be prevented by backflow induced by the operating fans. If, during normal operation, an increase in the carbon filter temperature were to occur, the high temperature detectors would initiate an alarm and the operator would cause the affected bank to be sprayed.

Reliance on Interconnected Systems

The Containment Air Recirculation Cooling and Filtration 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 post-accident period for the containment cooling function.

Shared Function 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.

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. 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 “clean up” as that interior air continually recirculates through the heat exchanger.

It will be noted that the motor insulation hot spot temperature is not expected to exceed 127 C even under incident conditions. Normal life could be expected with a continuous hot spot of 155 C.

During the lifetime of the plant, these motors perform the normal heat removal service and as such are only loaded to approximately 90-100 hp, which is less than half the rated horsepower.

The bearings were designed to perform in the incident ambient temperature conditions. However, it should 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 (see Appendix 6F). The heat exchanger system of preventing moisture from reaching the winding keeps the winding in much more favorable conditions. In addition, it should 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. Additionally, the motors are furnished with insulation voltage margin beyond the operating voltage of 440 volts.

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 was designed using a conservative 0.001 fouling factor.

To prove the effectiveness of the heat exchanger in inhibiting large quantities of the steam air mixture from impringing on the winding and bearings, a full scale motor of the exact same type as described was subjected to prolonged exposure of accident conditions, which included high pressure and temperature, 100% relative humidity, and chemical spray. The test exposed the motor to a steam air mixture as well as boric acid and alkaline spray at approximately 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 further assure that the unit performed as designed. The post-testing inspection showed no degradation of the motor components.

Carbon Filter Performance

The design flow rate through each carbon filter bank is 8000 cfm, at a face velocity of approximately 50 fpm. The bed thickness of 2 inches provides a superficial residence time of 0.2 sec. Under the design conditions of temperature, pressure, and humidity, and with moisture uptake limited to less than 1 gram of water per gram of dry charcoal, the expected penetration of incident 1_2 vapor is less than 0.1%.

An evaluation of the effectiveness of charcoal filters in removing organic iodide from the containment atmosphere is presented in Appendix 6C.

6.4.4 Minimum Operating Conditions

The Technical Specifications establish limiting conditions regarding the operability of the air recirculation units when in MODES 1, 2, 3, and 4.

6.4.5 Inspection and Testing

Inspection

Access is available for visual inspection of the containment fan coolers and recirculation filtration components, including fans, cooling coils, dampers, filter units and ductwork. Provision was made for ready removal of the filters for inspection and testing.

Technical Specifications (TS) require charcoal and HEPA filter testing to demonstrate operability any time a fire, chemical release or work done on the filters could alter integrity. Technical Specification surveillance testing is based upon a maximum flow of 8,800 cfm giving a minimum safety factor of 1.87 for methyl iodide removal efficiency while allowing 1% bypass. 50.59 Evaluation 98-3-017 HVAC demonstrates, for purpose of TS implementation, that welding is not a fire, a chemical release or work that could alter filter integrity. The 50.59 evaluation also demonstrates that organic components from painting and similar activities could not alter filter integrity until the organic components are above 10% by weight and concludes that filter testing shall be performed when the organic components are greater than or equal to 2.5% by weight organics. Administrative controls are required to evaluate the percent by weight of organics when activities that could generate organics are conducted.

Testing

Component Testing

The HEPA filters used in the containment fan cooler system were specified to operate in the post-accident containment environment. Each filter was subjected to standard manufacturer's efficiency and production tests prior to shipment.

These included flow resistance tests and the Standard Efficiency Penetration Test requiring that penetration does not exceed 0.03 percent for 0.3 micron diameter homogeneous diocylphathalate (DOP) particles.

Evaluation tests were performed on sample filters constructed from the filter medium to demonstrate retention of strength under wet conditions, and to demonstrate the effectiveness of the moisture separator for protecting the HEPA filter as follows:

1) The filter was exposed to a flow of wet steam (at 280 F, 50 psig, and 100% R.H.) and water spray (with 2500 ppm boron, pH of 10) in a test facility which simulated the actual filter installation. The water was injected ahead of the filter with a nozzle designed to produce a fine spray. Free (unentrained) moisture was removed by means of a moisture separator upstream of the HEPA filter but no provisions were made for removal of entrained moisture entering the HEPA filter.

2) The filter pressure drop was measured to demonstrate that its resistance to flow under the simulated accident conditions did not significantly increase.

Only filters of a type which have been certified to have passed these tests were accepted for initial use or replacement in the fan coolers application.

Any of the activated carbon filter adsorbers in the air handling units can be removed and tested periodically for effectiveness in removing methyl iodine forms. In addition, periodic in-place testing of the filtration assemblies is made by injection of a DOP aerosol in the air stream at the filter inlet to verify the leak-tightness of individual filter elements and their frame seals. The activated charcoal used shall have an ignition temperature not less than 300 ^{o}F.

The in-place testing of HEPA filters with DOP aerosol is performed to demonstrate gasket and media integrity, and overall bank efficiency, rather than an investigation of individual pinhole leaks in the filter media. Test procedures are available at the plant site for inspection.

Large filter installations are tested to within 20% of the full rated flow. Besides limiting the quantity of DOP to be introduced into the ventilation system and containment, this is the flow rate at which filter imperfections would most readily be noticeable. At higher flow rates, the turbulent flow through pinhole leaks and other imperfections becomes proportionally less than the laminar flow through the media. Filters therefore increase in efficiency with increasing air flow rates. When an in-place test, carried out in accordance with NRC requirements, shows an unacceptable efficiency, leakage paths can be detected by passing the aerosol through the system, and probing the downstream side of the bank of filters and mounting frame with a probe connected directly to the photometer.

Carbon filters will not be contaminated with DOP, and will be removed from the system before any testing takes place.

For small charcoal filter installations, filter bank efficiencies are determined using Freon 112, in accordance with the procedures described in DP1082 "Standardized Nondestructive Test of Carbon Beds for Reactor Confinement Applications." For large installations, the use of this procedure would necessitate the release of excessive amounts of Freon 112 within the Containment. Due to problems of possible fluoride formation, it is desirable to keep freon contamination to a minimum.

Consequently, instead of introducing Freon into a fully operating ventilation system, carbon filter installations are tested a few cells at a time. The procedure is to use a small temporary portable blower and duct on the inlet side, while checking for leakage on the downstream side of the installation with a halogen leak detector. Any Freon pickup which may occur in the section of the filter under test will be released following the completion of the test and will have no effect on filter performance.

The dampers and blow-in door on each air handling unit can be operated periodically to assure continued operability.

System Testing

Each fan cooling unit was tested after installation for proper flow and distribution through the duct distribution system. Four fan cooling units are used during normal operation. (Five will only be required for normal operation at design conditions, i.e., when the service water inlet temperature is above 85°F, and this condition is expected to exist only for relatively short periods, if at all.) The fan not in use can be started from the Control Room to verify readiness. The dampers and blow-in door directing flow through the carbon filter banks are tested only when the fan is not running.

After reinstallation following testing, the carbon filter units are tested in place by aerosol injection to determine integrity of the flow path.

Operational Sequence Testing

The test described in Section 6.2.5 serves to demonstrate proper transfer and sequencing of the fan motor supplies from the diesel generators in the event of loss of power. A test signal is used to demonstrate proper damper motion and fan starting prior to installation of the carbon filters. The test verifies proper functioning of the vane-switch flow indicators.

Verification of Heat Removal Capability

Since river water is circulated through the containment fan coolers and since the fans are used under both normal and accident conditions, provisions were made for verifying that the fan cooler heat removal capability does not degrade below that assumed in the containment integrity evaluation.

Instrumentation provided to verify heat removal capability is:

1) An Environmentally qualified RTD is installed on the inlet line to provide indication on the critical function monitoring system (CFMS).

2) Flow measurement of each fan cooler service water effluent is provided by an indicating flow transmitter installed in each line. The transmitter actuates a common annunciator alarm in the Control Room upon the decrease of flow in any fan cooler line.

   In addition, the flow indicator provides for manual balance of flow rates in all five fan coolers.

3) In the event of fan cooler coil service water out-leakage, the head of water will increase in a stand pipe weir which collects condensate runoff from each of the fan cooler, motor heat exchanger and demister (moisture separator).

   The increase of head is measured by a differential pressure transmitter. The current output signal is connected to an alarm unit which actuates a control room annunciator. Through the use of a weir level indicator and selector switch, the operator can determine the location of the leakage.

4) The containment building ambient temperature is controlled by manually modulating the service water flow through the fan coolers.

   The indicating range is 40° - 400°F. Average temperature indication is available at the QSPDS display and at the CR Supervisory Panel. Individual temperatures for each RTD are displayed at the CFMS. An increase in ambient temperature indicates fan cooler failure or service water discharge control valve malfunction. Either cause can be easily checked. To ensure reliability of the temperature instruments, perform a channel check daily and a channel calibration every 24 months.

References

1) "Connecticut-Yankee Charcoal Filter Tests," CYAP 101, (December 1966).

2) Ackley, R.D. and R. E. Adams, "Trapping of Radioactive Methyl Iodide from Flowing Steam-Air: Westinghouse Test Series," ORNL-TM-2728, (December 1969).

3) Reactor Containment Fan Cooler System Technical Manual, Nuclear Technology Division of Westinghouse Electric Corporation, PE-1275, (May 1982).

4) Attachment I to IPN-89-046, "Proposed Change to Technical Specifications to Increase the Design Bases Ultimate Heat Sink Temperature."

TABLE 6.4-1

SINGLE FAILURE ANALYSIS – CONTAINMENT AIR RECIRCULATION COOLING AND FILTRATION SYSTEM

Component	Malfunction	Comments and Consequence
A. Containment Cooling Fan	Fails to start	Five provided. Evaluation based on three fans containment spray pump operating during the in configurations are evaluated in Chapter 14.
B. Nuclear Service Water Pumps	Fails to start	Three provided. Two required for operation for contai
C. Automatically Operated Valves: (Open on automatic safeguards sequence)		
1) Carbon filter compartment Damper and Blow-in door	Fails to open	Five filters provided. Evaluation based on three filte containment spray pump in operation during the inject
2) Nuclear service water discharge line isolation Valve	Fails to open	Two provided. Operation of one required.

TABLE 6.4-2

SHARED FUNCTION EVALUATION

Component	Normal Operating Function	Normal Operating Arrangement	Accident Function	Accident Arrangement
Containment Cooling Fan Unit (5)	Circulate and cool containment atmosphere	Up to five fan units in service	Circulate and cool containment atmosphere	Three to five fan units in service
Nuclear Service Water Pumps (3)	Supply river cooling water to fan units	One or two pumps in service	Supply river cooling water in service components	Two or three pumps
Carbon Filter Units (5)	None	Isolated from normal fan discharge flow	Remove iodine from containment atmosphere	Lined up to receive fan discharg

6.5 ISOLATION VALVE SEAL WATER SYSTEM

6.5.1 Design Bases

The Isolation Valve Seal Water System assures the effectiveness of the containment isolation valves 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 reliable means for injecting seal water between the seats and stem packing of the globe and double disc types of isolation valves, and into the piping between closed gate valves and diaphragm type isolation valves. This system operates to limit the fission product release from the Containment.

Although no credit is taken for operation of this system in the calculation of offsite accident doses, 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 Seal Water System are discussed in Section 6.5.5.

See Section 5.2, Containment Isolation System for containment isolation diagrams, tabulation of isolation valve parameters 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, post-accident, by air from the Penetration and Weld Channel Pressurization System.

6.5.2 System Design and Operation

System Description

The Isolation Valve Seal Water System flow diagram is shown in Plant Drawing 9321-F-27463 [Formerly Figure 6.5-1]. System operation is initiated either manually or by any automatic safety injection 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 leakage from the Containment through valve seats and stem packing. The water is introduced at a pressure slightly higher than the containment peak accident pressure. The high pressure nitrogen supply used to maintain pressure in the seal water tank does not require any 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 assuring that if leakage does exist, it will be from the seal water system into the Containment.

The following lines would be subjected to pressure in excess of the isolation Valve Seal Water System design pressure (150 psig) in the event of an accident, due to operation of the recirculation pumps:

1) Residual heat removal loop return 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
5) Residual heat removal pump miniflow line
6) Residual heat removal loop outlet line.

Lines 1, 2, and 6 are isolated by double disc gate valves, while 3, 4 and 5 are each isolated by two valves in series. These valves 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 during power operation, and 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 pressurized from the system's own supply of high pressure nitrogen cylinders through pressure control valves. Design pressure of the tank and injection piping* 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.

In certain lines approximately three inches and larger, double disc gate valves are 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 discs 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 discs. The seal water pressure in excess of the potential accident pressure eliminates any outleakage past the first isolation point.

For smaller lines, isolation is provided by two globe valves in series (inboard and outboard) outside containment, with the seal water injected into the pipe between the valves. The valves are oriented such that the seal water wets the stem packing and plugs as follows:

1) On process lines ingressing containment (incoming lines) IVSWS will wet both the stem packings and plugs on both the inboard and outboard valves.

2) On process lines egressing containment (outgoing lines) IVSWS will wet both the stem packing and plug on the inboard valve, but only wet the plug on the outboard valve. One exception would be the Steam Generator Blowdown CIVs where both the inboard and outboard valves stem packings and plugs are wetted by IVSWS.

*NOTE: The injection piping runs and nitrogen supply piping are fabricated using 3/8 inch O.D. tubing, which is capable of 2500 psig service.

When the valves are closed for containment isolation, the first isolation point is the valve plug in the valve closest to Containment. One exception would be the RCP seal injection line CIVs where both the valve plug and stem packing act as isolation points. In a number of the smaller lines, isolation is provided by two diaphragm valves in series, with the seal water injected into the pipe between the valves.

The normally acceptable leakage across both the seat and stem packing of any gate or globe valve is 10 cc/hr/inch of nominal pipe diameter. Tests on these valves have indicated that much lower leakage rates can be expected. However, design of the Isolation Valve Seal Water System is based on the conservation assumption that all isolation valves are leaking at five times the acceptable value, or 50 cc/hr/inch of nominal pipe diameter. In addition, should one of the isolation valves fail to seat, flow through the failed valve will be limited to approximately 100 times the maximum acceptable leakage value, or 1000 cc/hr/inch of nominal pipe diameter, by the resistance of the seal water injection path. A water seal at the failed valve is assured by proper slope of the protection line, or a loop seal, or by additional valves on the side of the isolation valves away from the Containment.

The seal water tank is sized to provide at least a 24 hour supply of seal water under the most adverse circumstances, i.e., isolation valves leaking at the design rate of 50 /cc/hr/ inch, plus the failure of the largest containment isolation valve to seat and leakage at the maximum rate of 1000 cc/hr/inch. The seal water volume required to satisfy these conditions is approximately 144 gallons. A 176 gallon seal water tank is provided. If all of the isolation valves seat properly, as expected, the tank volume is sufficient for approximately 2½ 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: the primary water storage tank and the city water system. The tank is instrumented to provide local indication of pressure and water level; low water level, low pressure and high pressure are alarmed on the Waste Disposal/Boron Recycle Panel on El. 55 of the PAB. Any of these local alarms will be annunciated in the Control Room.

Seal Water Actuation Criteria

Containment isolation (Section 5.2) and seal water injection are accomplished automatically for the 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 provided for lines that could communicate with the containment atmosphere and be void of water following a Loss-of-Coolant Accident. These lines are as follows:

- Pressurizer Steam and Liquid Space Sample lines
- Excess Letdown Heat Exchanger Cooling Water supply and return lines
- Letdown line
- Reactor Coolant System sample line
- Containment vent header
- Reactor coolant drain tank gas analyzer line
- Station air line

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 post-accident service. These lines are as follows:

- Pressurizer relief tank gas analyzer line
- Pressurizer relief tank makeup line
- Safety Injection System test line
- Reactor coolant drain tank pump discharge line
- Steam generator blowdown lines
- Steam generator blowdown sample lines
- Demineralized water to Containment
- Accumulator sample line
- Containment sump pump discharge

Manual containment isolation and 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 manual seal water injection assures a long term seal. These lines are as follows:

- Reactor coolant pump seal water supply lines
- Reactor coolant pump seal water return line
- Charging line
- Safety injection header
- Containment spray headers
- Reactor coolant pump cooling water supply and return lines

Manual containment isolation and manual seal gas injection are provided except as noted 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. These lines are as follows:

- Bypass line from residual heat exchanger outlet to safety injection pump suction
- Residual heat removal loop return line
- Residual heat removal loop sample line (automatic isolation)
- Safety injection line from boron injection tank
- Recirculation pump discharge sample line
- Residual heat removal pump miniflow line
- Residual heat removal loop outlet line

Seal water injection is not necessary to insure the integrity of isolated lines in the following categories:

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, reactor coolant drain tank, the instrument air header, the weld channel pressurization air lines, and the pressurizer pressure deadweight calibrator line.

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.

Lines that are designed for post-accident service as part of the engineered safety features. The only line in this category is the containment sump recirculation line. This line is connected to a closed system outside containment.

Special lines such as the fuel transfer tube, containment purge ducts and the containment pressure relief line. 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.

Components

All associated components, piping and structures of the Isolation Valve Seal Water System are designed to Class I seismic 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 to accordance with the ANSI Code for Pressure Piping (Power Piping Systems) B31.1.

6.5.3 Design Evaluation

The isolation Valve Seal Water System (IVSWS) 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 analysis of the consequences of the accident, provides an additional means of conservatism in ensuring that leakage is minimized. No detrimental effect on any other safeguards systems will occur should the seal water system fail to operate.

Post-accident access for a total of 22 manual IVSWS valves fed from line No 539 and 542 is provided. (Refer to Plant Drawing 9321-F-27463 [Formerly Figure 6.5-1]) Operation of these valves in an acceptable radiation field area during post-accident plant operating conditions is possible.

System Response

Automatic containment isolation will be completed within approximately two seconds following generation of the Phase A containment isolation signal. This is the approximate closing time of the air operated containment isolation valves (Section 5.2). This closing time is a nominal value only, and is not used as a valve stroke performance criteria nor as an input to any accident analysis or off-site dose calculation. Since the Isolation Valve Seal Water System is actuated by this signal, automatic seal water injection will be in effect within this nominal time period.

Subsequent generation of the Phase B isolation signal on containment high pressure (spray actuation signal) will close a number of motor operated isolation valves with an approximate closing time of 10 seconds (Section 5.2). This closing time is a nominal value only, and is not used as valve stroke performance criteria nor as an input to any accident analysis or off-site dose calculation.

Seal water (or Nitrogen) injection flow is manually initiated to these valves as well as the remainder of the containment isolation valves that receive a manually initiated closure signal at the appropriate time following the loss-of-coolant accident.

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.

Reliance on Interconnected Systems

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 in the control room of seal water tank pressure and level.

Shared Function Evaluation

Table 6.5-3 is an evaluation of the main components discussed previously and 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 when the reactor is critical.

6.5.5 Inspections and Tests

Inspections

The system components are all located outside the containment and can be visually inspected at any time.

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.

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; low water level, low pressure and high pressure will be annunciated locally on the Waste Disposal Panel.
The system is not in service during the Containment Integrated Leakage Rate Test.

Operational Sequence Testing

The capacity of the system to deliver water in accordance with the design 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, F	200
Operating pressure, psig	45 – 100
Operating temperature, F	Ambient
Code	ASME UPV (Sect. 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	Isolation Valve Seal Water Storage Tank (1)	N_2 Supply Bottles (3)
Normal Operating Function	None	None
Normal Operating Arrangement	Lined up to seal water injection piping	Lined up to seal water tank
Accident Function	Source of water for sealing isolation valves	Source of N_2 to maintain seal water pressure
Accident Arrangement	Lined up to seal water injection piping	Lined up to seal water tank

6.6 CONTAINMENT PENETRATION AND WELD CHANNEL PRESSURIZATION SYSTEM

6.6.1 Design Basis

The WCCPPS is incorporated into the design of Indian Point 3 as an engineered safety feature. Its purpose is to provide pressurized gas to all containment penetrations and most liner inner weld seams such that in the event of a LOCA, there would be no leakage through these potential leakage paths from the containment to the atmosphere. Spaces between selected isolation valves are also served by the WCCPPS. By maintaining the WCCPPS at some pressure level above the peak accident pressure, any postulated leakage would be into the Containment rather than out of the Containment.

Although the WCCPPS is an engineering safety feature, no credit is taken for its operation in calculating the amount of radioactivity released for offsite dose evaluations. For Indian Point 3, offsite dose calculations were performed to demonstrate compliance with 10 CFR 100 guidelines and the results were well within those guidelines. In those calculations, it was assumed that the Containment leaked at a rate of 0.1% per day of Containment free volume for the first 24 hours and 0.045% per day of Containment free volume thereafter.

6.6.2 System Design and Operation

System Description

The containment Penetration and Weld Channel Pressurization System is shown in Plant Drawing 9321-F-27263 [Formerly 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 most inner liner weld channels. The system maintains a pressure in excess of containment calculated peak accident response pressure continuously during all reactor operations thereby ensuring that there will be no out-leakage of the containment atmosphere through the penetrations and most liner welds during an accident. Following a design basis accident, the system will maintain pressure greater than the post accident containment pressure for 24 hours. Typical piping and electrical penetrations are described in Chapter 5.

The primary source of air for this system is the instrument air system (Chapter 9). 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 plant air system acts as a backup to the instrument and control air system added reliability. One plant air compressor is available for backup.

A standby source of gas pressure for the system is provided by a bank of nitrogen cylinders (see Table 6.6-1). The associated nitrogen system will automatically deliver nitrogen at a slightly lower pressure than the normal regulated air supply pressure. 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 automatically be maintained by the nitrogen supply. This assures reliable pressurization under both normal and accident conditions.

The backup gas supply is sized such that over the 24 hour period following a LOCA, WCCPPS pressure starts above the peak containment pressure, and then is continually maintained above the post-LOCA containment pressure profile.

The design basis leakage rate from the WCCPPS is 0.2% of containment free volume per day, with 0.1% of containment free volume per day leaking into the containment and an identical amount leaking to the environment.

Once the air receiver pressure decays to a 45 psig zone pressure, an automatic transfer to the backup nitrogen cylinders occurs. The WCCPPS zone is then fed nitrogen until the nitrogen accumulators reach equilibrium pressure with the air receiver. Once this equilibrium pressure is reached, both air and nitrogen mix to supply the leaking WCCPPS zone header until the air receiver and nitrogen accumulators are depleted.

Containment penetration 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 6.6-2), is sized to supply air to its pressurized zone for a period of at least one hour, based on a leakage rate of 0.2% of the containment free volume per day from the affected zone (0.1% leakage into the containment and 0.1% 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 will be supplied to the affected zones. Together the air receivers and nitrogen bank are sized to provide a 24 hour supply of gas to the system, again based on a total leakage rate from the pressurization system of 0.2% of the containment free volume in 24 hours. There are three nitrogen cylinders in the bank each 24" OD by 20' 6½" long. The nitrogen supply will also automatically assume the pressurization gas load in the event that an air receiver fails.

A pressure relief valve set at 175 psig (sized for 1250 scfm at 10% accumulation) protects the system from failure of the pressure reducing valve in the line to each zone from the bank of nitrogen cylinder. Each zone of piping is also protected by a relief valve designed to open at 82 psig. Pressure control valves, isolation valves and check valves are located outside of the containment for ease of inspection and maintenance. Failure of any of these components does not lead to 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 assure 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 assuring that 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. Pressurization of the zone is verified by closing off the air supply line and opening the flow-through test line valve to observe the escape of the pressurizing medium. Containment penetration and WCCPPS atmosphere may be sampled by opening the flow through test valves.

Pressure Indication

In order to ensure that the station operators are aware at all times that all penetrations and liner weld seam channels are pressurized, the following instrumentation is provided:

1) The following pressurized zones are equipped with local pressure gages, mounted outside the containment for ready accessibility and available for regular reading. The accuracy of these gages is within 2% of the full scale reading.
    a) Each piping penetration, except piping penetration 0'0'
    b) Each electrical penetration
    c) The spaces between the two isolation (butterfly) valves in the purge supply and exhaust ducts
    d) The two spaces between the three isolation (butterfly) valves in the containment pressure relief line
    e) The double-gasketed space on the outside hatch of each of the two personnel air locks
    f) The sump drain line valve enclosure
    g) Spaces between the Post Accident Containment Vent containment isolation valves

2) 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 and identification lights in the control room. Examples of the zones so equipped are:
    a) Each liner seam weld channel that remains connected to the Weld Channel Pressurization System
    b) The double-gasketed space on each inside hatch of the personnel air locks
    c) The double-gasketed space on the equipment door flange
    d) The pressurized zones in the spent fuel transfer lube
    e) Shroud rings over penetration to containment liner weld-piping, and electrical penetrations

The actuating pressure for each pressure switch is set just above incident pressure and just below the nitrogen supply regulator setting.

Personnel Air Lock Interlock

Continuous pressurization of air-lock door double-gasketed barriers, and the protection of the pressurization header against air loss are assured by a set of interlocks. One interlock on each air-lock door prevents 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 assure 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 re-pressurized.

Containment Purge Line Interlock

The containment ventilation purge penetration butterfly valves are also interlocked to prevent the opening of either valve until the pressurization line to the space between the valves has been isolated. Isolation of the pressurization line to each purge duct pressurized zone can be accomplished remotely from the Control 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 on each pressurization line to the ventilation purge ducts to assure that air consumption, even on failure of an interlock, will not result in loss of pressure to the other zones connected to the same pressurization header.

The containment pressure relief line isolation valves (three butterfly valves in series), and the two pressurized spaces formed between these valves, are provided with similar interlocking to prevent the opening of any of the butterfly valves until the adjacent intervalve space has been depressurized. The pressurization lines to these spaces are also equipped with flow restricting orifices, and alarm lights in the containment identify open valves or low intervalve space pressure.

Containment Inleakage

With a continuous inleakage to the Containment from the penetration and liner weld joint channel pressurization system of 0.1% 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 through the use of the two containment auxiliary charcoal filter units, each complete with roughing filters, HEPA filters, and charcoal filters (see Chapter 5).

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. A narrow range pressure indicator is provided on the local fan panel to assist in operation of the building pressure relief fan. The range is –5 to +5 psig.

Components

All associated components, piping, and structures, of the Containment Penetration and Weld Channel Pressurization System are designed to Class I seismic criteria.

The piping and valves for the system are designed in accordance with the ANSI Code for Pressure Piping (Power Piping Systems), B31.1.

For a description of the instrument and control air compressors and the plant air compressor, see Section 9.6.

The three nitrogen cylinders provided meet the requirements of 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 an accident, provides an additional means for ensuring that leakage is minimized, if not altogether eliminated. No detrimental effect of any other safety features system will be felt should the pressurization system fail to operate.

System Response

WCCPPS is not single failure proof, as the WCCPPS zones are not redundant. A nitrogen or air regulator failure may render a zone, or in some instances the entire system, incapable of performing its design function (i.e., pressurize the space between containment isolation valves, weld channels and containment penetrations at a pressure greater than the containment accident pressure profile for 24 hours post accident).

This can be tolerated for the following reasons. While one of the design basis functions for WCCPPS is to minimize offsite releases, WCCPPS is not needed to meet the requirements of 10 CFR 100. In addition, no other safeguards systems are dependent on the operation of WCCPPS. As such, a WCCPPS failure would not create undue risk to the health and safety of the public.

To account for active failures, two parallel WCCPPS supply valves are provided for certain containment isolation lines that are normally or intermittently open during operation. These containment isolation valves automatically close on a containment isolation signal. Opening of one of the two WCCPPS supply valves in each line is sufficient to accomplish pressurization gas injection upon closing of the containment isolation valves.

Shared Functions Evaluation

Table 6.6-4 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 when the plant is above MODE 5.

6.6.5 Inspections and Tests

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.

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. Therefore, remedial action can be taken before the limit is reached. For those liner welds that are no longer continuously pressurized, a leak would not be identified during plant operation. The integrity of these welds is verified by integrated leak rate testing.

In order to provide facility for testing the larger penetrations, branch pressurizing lines are provided from one of the zones to:

1) The double-gasketed space on each hatch of the Personnel Air Lock.
2) The double-gasketed space at the Equipment Hatch flange.
3) The pressurized zones in the spent fuel transfer tube.
4) The spaces between the two butterfly valves in the purge supply exhaust ducts.
5) The two spaces between the three butterfly valves in the containment pressure relief line.
6) The spaces between double containment isolation valves in the steam jet air ejector return line to containment and in the containment radiation monitor inlet and outlet lines.
7) The spaces between the isolation valves for the Post Accident Containment Vent lines and VC Hydrogen Analyzer lines.

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 selection of operating limits for air consumption of the pressurization system are based upon the integrated containment leak rate test acceptance criterion 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 included in the Technical Specifications.

A flow sensing device is located in each of the headers supplying makeup air to the four pressurization zones. Signal output from each of the four flow sensors is applied to an integrating recorder located in the Control Room. Output from all sensors is also applied to a summing amplifier which drives a total flow recorder. High flow alarms are also derived in the recording channel, to alert the operator in the Control Room. The flow measurement range is 0-15 SCFM with an accuracy of +/- 1% of full scale. Since a flow of 0.2% of the containment volume per day at 47 psig is approximately 3.6 Ft^3/minute, the sensitivity of the flow meters is well within the maximum leakage of the pressurization system.

TABLE 6.6-1

CONTAINMENT PENETRATION AND WELD CHANNEL PRESSURIZATION SYSTEM
EMERGENCY NITROGEN STORAGE

Number	3
Volume, (each) ft^3	7333*
Material	ASTM A-372-CL IV
Design pressure, psig	2450
Design temperature, F	200
Operating pressure, psig	2200
Operating temperature, F	100
Code	ASME UPV (SECT. VIII)

*NOTE: Each nitrogen cylinder has a volume of 51 ft^3 and can store 7333 standard cubic feet of dry nitrogen at 2200 psig at 70° F.

TABLE 6.6-2

CONTAINMENT PENETRATION AND WELD CHANNEL PRESSURIZATION SYSTEM AIR RECEIVERS

Number	4
Volume, (each) ft^3	360
Material	ASTM A-285-C
Design pressure, psig	140
Design temperature, F	200
Operating pressure, psig	100
Operating temperature, F	100
Code	ASME UPV (SECT. VIII)

TABLE 6.6-3

SINGLE FAILURE ANALYSIS – CONTAINMENT
PENETRATION AND WELD CHANNEL PRESSURIZATION SYSTEM

Component	Malfunction	Comments
Instrument Air Control Air Compressor	Fails to maintain pressure	One of two instrument and control air compresso operate.
Pressure Reducing Valve for each zone	Fails to maintain pressure	On valve failure, flow is limited to acceptable valu critical pressure drop orifice. Under low flow con pressurization of system downstream of valve is valve.

TABLE 6.6-4

SHARED FUNCTIONS EVALUATION

Component	Normal Operating Function	Normal Operating Arrangement	Accident Function	Acc
Instrument and Control Air Compressors (2)	Supply air to plant instrument and controls and to penetrations and weld channels	2 air compressors in operation	Supply air to penetra-tions and weld channels	1 ai ope
Plant Air Compressor (1)	Supply air to station air headers	1 air compressor in operation	"	1 ai ope
N_2 Cylinders (3)	Backup source of N_2 to maintain penetration and weld channel pressure	Lined up to Penetration and Weld Channel Pressurization System	Backup source of N_2 to maintain penetration and weld channel pressure	Line and Pres
Air Receivers (1) and Dryers (3)	Primary source of air for penetrations and weld channels	Lined up to Penetration and Weld Channel Pressurization System	Primary source of air for penetrations and weld channels	Line and Pres

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

The General Design Criteria presented and discussed in this section are those which were in effect at the time when Indian Point 3 was designed and constructed. These general design criteria, which formed the bases for the Indian Point 3 design, were published by the Atomic Energy Commission in the Federal Register of July 11, 1967, and subsequently made a part of 10 CFR 50.

The Authority completed a study of compliance with 10 CFR Parts 20 and 50 in accordance with some of the provisions of the Commission's Confirmatory Order of February 11, 1980. The detailed results of the evaluation of the compliance of Indian Point 3 with the General Design Criteria presently established by the Nuclear Regulatory Commission (NRC) in 10 CFR 50 Appendix A, were submitted to NRC on August 11, 1980, and approved by the Commission on January 19, 1982. These results are presented in Section 1.3. The NRC has concluded that the current IP3 leakage detection system capability is adequate to continue to support the technical bases cited in the NRC's March 10, 1986, SE approving Leak Before Break (LBB) for the IP3 Primary Coolant Loop piping. This position was further clarified in the IP3 Supplement to Safety Evaluation re: Leakage Detection Systems (TAC No. MB3328, dated 01/30/02).

Monitoring Reactor Coolant Leakage

Criterion: Means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary. (GDC 16 of 7/11/67)

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. This equipment provides 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. The containment sump system with incorporation of a CR alarm (VC Sump Pump running) or operator actions to increase monitoring of the processing system (i.e. sump flow monitor once every 4 hours), provides the capability of detecting a 1 gpm leak within four hours.

These methods are designed to monitor leakage into the Containment atmosphere and as such do not distinguish between identified and unidentified leaks.

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 radio-activity releases to the environs of the plant have not been excessive. (GDC 17 of 7/11/67)

The containment atmosphere, the plant ventilation exhaust (including exhausts from the Fuel Storage Building, Primary Auxiliary Building, and Waste Holdup Tank Pit), the containment fan-coolers 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.

<u>Principles of Design</u>

The principles for design of the leakage detection systems can be summarized as follows:

1) Increased leakage could occur as the result of failure of pump seals, valve packing glands, flange gaskets or instrument connections. The maximum single 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 shall not produce spurious annunciation from normal expected leakage rates but shall reliably annunciate increasing leakage.

3) Increasing leakage rate shall be annunciated in the control room. An exception is for the VC sump where compensatory operator action could be used if the CR alarm is unavailable as long as the sensitivity of the credited RCS leak detection system is maintained (i.e., 1 GPM within 4 hours). Operator action will be required to isolate the leak in the leaking system.

For Class I 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 monitor or by one of the area radiation monitors. Similarly, leakage to other systems which do not normally contain radioactive fluids would result in an increase in the activity level in that system.

2) For closed systems, 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 operation, or increased operation, of the associated sump pumps.

Details of how these methods are utilized to detect leakage from Class I systems other than the Reactor Coolant System are given in the following sections and summarized in Table 6.7-1.

The Authority has established a program to identify and reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident (NUREG – 0578). Leak test results for these systems are presented in Table 6.7-2.

6.7.1.2 Systems Design and Operation

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.

Reactor Coolant System

During normal operation and anticipated reactor transients, the following methods are employed to detect leakage from the Reactor Coolant System:

Containment Air Particulate Monitor (R-11)

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 counter – filter paper detector assembly. The filter paper collects all particulate matter greater than 1 micron in size on its constantly moving surface, which is viewed by a photomultiplier plastic scintillator 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 and 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 and transmitted to a microprocessor which converts the detector output to digital and analog signals for display, generates alarms and communicates with the Radiation Monitoring System cabinet in the Control Room. Lead shielding is provided for the radiogas detector to reduce the background radiation level to where it does not interfere with the detector's sensitivity.

The activity is indicated on digital displays and recorded by a stripchart recorder. High-activity alarm indication is displayed on the control room annunciator, and the radiation monitor microprocessor. Local and control room alarms provide operational status of supporting equipment such as pumps, motors and flow and pressure controllers.

The containment air particulate monitor is the most sensitive instrument of those available for detection of reactor coolant leakage into the Containment. The measuring range of this monitor is given in Section 11.2.

The sensitivity of the air particulate monitor 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 experience. (See Appendix 6B.) Where containment air particulate activity is below the threshold of detectability, operation of the monitor with stationary filter paper would increase leak sensitivity to a few cubic centimeters per minute.

Using a source term based on six months after start up through the end of the operating cycle with little fuel defect, varying ambient background level, the least conservative detection capability for R-11 is not expected to exceed a value of 2 gpm within 4 hours. Varying detector background, RCS activity level and failed fuel conditions are contributors to changes in R-11 detection capability.

Containment Radioactive Gas Monitor (R-12)

This channel measures the gaseous beta radioactivity in the Containment by taking the continuous air samples from the containment atmosphere, after they pass through the air particulate monitors, and drawing the samples through a closed, sealed system to a gas monitor assembly.

Each sample is constantly mixed in the fixed, shielded volume, where it is viewed by a plastic scintillator coupled to a heated photomultiplier. The samples are then returned to the Containment.

The detector is in a completely enclosed housing. Lead shielding is provided to reduce the background radiation level to a point where it does not interfere with the detector's sensitivity. A preamplifier is mounted at the detector skid.

The detector outputs are transmitted to a microprocessor which converts the detector output to digital and analog signals for display, and communicates with the Radiation Monitoring System cabinet in the Control Room. The activity is indicated by a digital display and recorded by a stripchart recorder. High-activity alarm indications are displayed on the control room annunciator and the radiation monitoring microprocessor. Local and control room alarms annunciate the supporting equipments' operational status.

The containment radioactive gas monitor is inherently less sensitive. Varying detector background, RCS activity level and failed fuel conditions are contributors to changes in R-12 detection capability. As the detector background increases, either the time to detect a 1gpm leak goes higher or the detectable RCS leak rate is greater than 1 gpm within the specified time frame. The measuring range of this monitor is given in Section 11.2.

R 11/12 and its associated equipment will provide 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 the change within the Containment and the equipment provided is capable of monitoring the change.

The NRC has concluded that the current IP3 leakage detection system capability is adequate tom continue to support the technical bases cited in the NRC's March 10, 1986, SE approving Leak Before Break (LBB) for the IP3 Primary Coolant Loop piping. This position was further clarified in the IP3 Supplement to Safety Evaluation re: Leakage Detection Systems (TAC No. MB3328, dated 01/30/02).

Humidity Detector

The humidity detection instrumentation offers another means of detection of leakage into the Containment. Although this instrumentation has not nearly the sensitivity of the air particulate monitor, it has the characteristics of being sensitive to vapor originating from all sources within the Containment, including the reactor coolant and steam and feedwater systems. Plots of containment air dew point variations above a base-line maximum established by the cooling water temperature to the air coolers should be sensitive to incremental increases of water leakage to the containment atmosphere on the order of 0.25 gpm per F degree of dewpoint temperature increase.

The sensitivity of this method depends on cooling water temperature, containment air temperature variation and containment air recirculation rate.

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. 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 near 120° F DB and 92° F WB (approximately 36% 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:

$$D = L\,[1-\exp(-Q / Vt)]$$

Where:

D	=	Change in drainage rate after initiation of increased leakage rate (gpm)
L	=	Change in evaporated leakage rate (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% of a new equilibrium value in approximately 200 minutes after the start of the leak. Detection of the increasing condensation rate, however, would be possible within 5 to 10 minutes.

The condensate measuring device consists essentially of a vertical 6 inch diameter standpipe with a 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 30 gpm can be measured by observing the height of the water level above the crest notch of the weir. This water head can be converted to a proportional flow rate by means of a calibration curve. A high level alarm, set above the established normal (baseline) flow, is provided for each unit to warn the operator when operating limits are approached.

All indicators, alarms, and controls are located in the Control Room. This method provides a backup to the radiation monitoring methods.

Component Cooling Liquid Monitors (R-17A and R-17B)

These channels continuously monitor 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, Each scintillation counter is located in an in-line well downstream of the component cooling heat exchangers. The detector assembly output is amplified by a preamplifier and transmitted to the Radiation Monitoring System cabinets in the Control Room. The activity is indicated on a meter and recorded. High activity alarm indications are displayed on the control board annunciator in addition to the Radiation Monitoring System cabinets.

The measuring range of this monitor is given in Section 11.2

Condenser Air Ejector Gas Monitor (R-15)

The channel monitors the discharge from the air ejector exhaust header of the condensers for gaseous radiation, which is indicative of a primary to secondary system leak. The gas discharge is routed to the turbine roof vent. In order to quantify the amount of radiation release from the condensers into the atmosphere, flow measuring instrumentation is installed into the same line with an output to the critical function monitoring system (CFMS). On high radiation level alarm, this gas discharge is diverted to the containment.

The detector output is transmitted to a microprocessor which converts the detector output to digital and analog outputs for display, generates alarms and communicates with the Radiation Monitoring System cabinet in the Control Room. The activity is indicated by a digital display and recorded by a stripchart recorder. High activity alarm indications are displayed on the control room annunciator and the radiation monitoring microprocessor.

A gamma sensitive Sodium Iodide (NaI) crystal scintillator/ photomultiplier tube is used to monitor the gaseous radiation level. The radiation monitor consists of a 3" pipe section in series with the steam jet air ejector exhaust line, a thin-walled sealed well (perpendicular to and penetrating the 3" pipe) which houses the NaI/PM assembly, and ample lead shielding to reduce background radiation interference to an acceptable level.

Steam Generator Liquid Sample Monitor (R-19)

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 mixed to a common header and the common sample is continuously monitored by one of two separate scintillation detectors. Upon indication of a high radiation level, each steam generator is individually sampled in order 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 minute).

The sensitivity range of this monitor is given in Table 11.2-7.

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 level so it does not interfere with the detector's maximum sensitivity. 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 crystals outside the Reactor Coolant System and the evaporation process leaves them behind.

If an accident involving gross leakage from the Reactor Coolant System occurred it could be detected by the following methods:

Pump Activity

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 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 pump reaches a high speed limit, an alarm is actuated.

A break in the primary system would result in reactor coolant flowing into the Containment, reactor vessel, and/or recirculation sumps. Leakage to these sumps would be indicated by the frequency of operation of the containment or recirculation pumps. Since the building floor drains preferentially to the containment sump, the operating frequency of the containment sump pumps would be more likely to indicate the leak than the operating frequency of the recirculation or reactor vessel sump pumps.

The containment sump contains two redundant level loops consisting of a transmitter and sensor inside containment, and a recorder, indicator and power supply in the control room.

An alarm will annunciate on the control room supervisory panel if the water level reaches the overflow point in the section of the sump specifically designed to preferentially collect Containment Building leakage.

The containment sump contains two (2) sump pumps which are actuated by separate pump float switches. These pumps discharge the water to the waste holdup tank outside Containment. Located on this discharge line outside containment is the flow meter and totalizer, which indicates on the Primary Auxiliary Building waste disposal panel the flow from the pumps and a cumulative measure of the amount of water being discharged from containment. The cumulative volume is trended by the control room operators to identify any abnormal increases in leakage on a daily basis. In addition, indicating lights on the waste disposal panel indicate when the containment sump pumps are running. This panel is periodically operated and monitored by the auxiliary operator who reports directly to the control room operator. The containment sump system with incorporation of a CR alarm (VC Sump Pump running) or operator actions to increase monitoring of the processing system (i.e. sump flow monitor once every four hours), provides the capability of detecting a 1 gpm leak within four hours.

The recirculation sump contains two redundant level loops consisting of a transmitter and sensor inside containment, and a recorder, indicator and power supply in the control room. Loss of both of these level indications requires a plant shutdown in accordance with Technical Specifications. The recirculation pumps, which discharge into the Reactor Coolant System, are required for a LOCA and require an immediate plant shutdown in accordance with Technical Specifications if they become inoperable.

The reactor pit contains a level sensor which annunciates two alarms at two separate levels on the control room supervisory panel. These alarms annunciate at different levels when the pit accumulates water prior to the level reaching the in-core instrumentation tubing for the reactor vessel. In addition, the running of the first sump pump illuminates an indicating light on the Control Room supervisory panel. At the present time, during normal plant operation, there is no means to test operability of either the level indication or pumps since this sump is normally maintained dry.

The containment sump contains redundant level indication. Loss of both of these level indications requires a plant shutdown in accordance with the Technical Specifications. Even if both level indications were inoperable, the level probe at the top of the sump would still provide an annunicated alarm.

<u>Liquid Inventory</u>

Gross leaks might be detected by unscheduled increases in the amount of reactor coolant makeup water which is required to maintain the normal level in the pressurizer.

A large tube side to shell side leak in the non-regenerative (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 high radiation and temperature alarms actuated by monitors at the component cooling water pump suction header.

A high level alarm for the component cooling water surge tank and high radiation and temperature alarms actuated by monitors at the component cooling pump suction header could also indicate a thermal barrier cooler coil rupture in a reactor coolant pump. However, in addition to these alarms, high temperature and high flow on the component cooling outlet line from the pump would activate alarms.

Leakage might also be indicated by a rise in the normal containment and/or recirculation sump levels. High level in either of these sumps is indicated in the Control Room. Since the building floor drains preferentially to the containment sump, the containment sump level transmitter would most likely be actuated prior to the level transmitter in the recirculation sump.

The maximum leak rate from an unidentified source that will be permitted during normal operation is 1 gpm.

Leakage directly into the Containment indicates the possibility of a breach in the coolant envelope. The limitation of 1 gpm for a source of leakage not identified is sufficiently above the minimum detectable leakage rate to provide a reliable indication of leakage. The 1 gpm limit is well below the capacity of one coolant charging pump (98 gpm).

The relationship between leak rate and crack site has been studied to detail in WCAP-7503(1), Revision 1, February 1972. This report includes the following information:

1) The length of a through-wall crack that would leak at the rate of the proposed limit, as a function of wall thickness.

2) The ratio of that length to the length of a critical through-wall crack, based on the application of the principles of fracture mechanics.
3) The mathematical model and data used in such analyses.

Leak rate detection is not relied upon for assuring the integrity of the primary system pressure boundary during operation. The conservative approach which is utilized in the design and fabrication of the components which 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 in-service 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 of 1 gpm which 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.

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 humidity detector and the condensate measuring system 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 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 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 within the Primary Auxiliary Building. Detection of a leaking residual heat removal pump is thus possible by means of the radiation monitors provided on the Primary Auxiliary Building plant ventilation system which exhausts these pump compartments.

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 shutdown personnel can enter the Containment to check visually for leaks. 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 which 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 these sumps have redundant level indication in the control room. As the building floor drains preferentially to the containment sump, the level transmitter in this sump would most likely be actuated first.

Should a large tube side to shell side leak develop in a residual heat exchanger or the seal heat exchanger of a residual heat removal pump break, the water level in the component cooling surge tank would rise and the operator would be alerted by a high water alarm. Radiation and temperature monitors at the component cooling water pump header will also signal an alarm.

Leakage from both of the residual heat removal pumps is drained to a common sump equipped with a sump pump. The sump pump starts automatically and transfers this leakage to the waste holdup tank; indication and alarm for high level in this tank is made on the waste disposal panel. This would provide indication of gross leakage (i.e., a seal failure from a residual heat removal pump).

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.

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) at the component cooling water pump discharge header.

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 radiation and temperature alarm actuated by monitors at the component cooling water pump header.

If the external recirculation loop is used, leakage from the section outside the Containment would be directed by floor drains to the auxiliary building sump and / or sump tank. From here, it is automatically transferred by pumps to the waste holdup tank. Indication and alarm for high level is made on the waste disposal panel. This would serve to alert the operator of the leakage.

Component Cooling Loop

Leakage from the component cooling loop inside the Reactor Containment could be detected by the humidity detector and/or the condensate measuring system (see section on Reactor Coolant System leak detection for a description of these systems).

Visual inspection inside the Containment is possible.

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 redundant level indication in the Control Room. As the building floor drains preferentially to the containment sump, the level transmitter in the sump would be more likely to signal the occurrence of leakage.

If the leakage is from a part of the component cooling loop outside the Containment, it would be directed by floor drains to the auxiliary building sump and/or sump tank. The auxiliary building sump pumps and/or sump tank pumps then automatically transfers the leakage to the waste holdup tank. Indication and alarm for high level is made on the waste disposal panel. This would serve as a means of leak detection for this part of the system.

Leakage of Component Cooling Water into the Service Water System through the Component Cooling Water Heat Exchangers can be detected by a radiation monitor (R-23) which monitors the Service Water return line from the CCW Heat Exchangers. A high radiation alarm is annunciated in the Control Room.

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 NaI scintillation detectors. 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 minute). This method of fan cooler unit (FCU) or FCU motor cooler leak detection and isolation may be performed up until the entry into external recirculation. This is sufficient time to detect and isolate the leak, since the passive failure of the cooling coil is assumed to occur concurrently with the LOCA.

The measuring range of these monitors is given in Section 11.2.

Gross leakage from the Service Water System due to a faulty cooling coil in the Containment Air Recirculation Cooling and Filtration System can be detected by stopping the fans and continuing the cooling water flow. Any significant cooling water leakage would be seen as flow into a collecting pan.

Leakage from a component in the Service Water System in the Primary Auxiliary Building will be directed by floor drains to the Primary Auxiliary Building sump tank. Pumps will then transfer this leakage to the waste holdup tank. Indication and alarm for high level is made on the waste disposal panel, and would serve as a means of leak detection.

Service Water System leaks inside Containment which are directed to the containment sump and/or the recirculation sump are handled and detected in the same manner as Reactor Coolant System leaks.

6.7.1.3 Pre-Operational Testing

Since initial operation was substantially without benefit of the containment air particulate monitor and radioactive gas monitor to indicate leaks, the performance of the humidity detector and the condensate measuring system was verified as follows:

The radiogas monitor, air particulate monitor, and humidity detectors were tested prior to plant installation. After installation, the radiogas and air particulate monitors were checked for operability as part of the system checkout of the Radiation Monitoring System utilizing the build-in check source provided in each detector. Sensitivity of the condensate measuring systems and humidity detectors were verified as part of the Indian Point 2 Test Program. These tests demonstrated system sensitivities.

During the startup test program at Indian Point 2, a reactor coolant leak of known magnitude was simulated inside the containment vessel, and the performance of the humidity detector and condensate measuring system was observed. The leak was simulated by introducing steam into the Containment at a known rate 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 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 began, as indicated by the condensate measuring devices, was recorded and compared with the calculated value based on 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. As the design of this system was verified in Indian Point 2, it was not necessary to repeat the test on Indian Point 3.

Operability of Technical Specification required radiogas and air particulate monitors is checked periodically in accordance with Technical Specification requirements. Simple comparative checks between the six humidity detectors or between the five condensate measuring systems readily confirm operability of these detection systems.

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) isolation of 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 which 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 avert uncontrolled leakage from the primary and auxiliary coolant loops.

The leak detection sensitivity of the radiation monitors during plant operation depends upon the primary coolant activity level and the normal baseline leakage. Reliable indications of coolant leakage above baseline are assured when the coolant activity and baseline leakage result in containment atmospheric activity levels within the sensitivity ranges of the radiation monitors. The sensitivity ranges of these monitors, and examples of their leak detection response times for various design coolant activity levels are given in Section 6.7.1.2.

When the containment atmospheric activity level is below the threshold of detectability of the radiation monitors, a primary coolant leak would be detected by the other redundant leak detection systems that monitor nonradioactive parameters (humidity, condensate runoff, liquid inventory, containment sump system).

6.7.2.3 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 activity or the radioactive gas activity exceed preset levels on the containment air particulate and radioactive gas monitors, respectively, the containment purge supply and exhaust duct valves and pressure relief line valves are closed.

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 through a blower.

A high radiation alarm actuated by the steam generator liquid sample monitor initiates closure of the isolation valves in the blowdown lines and sample lines.

If a leak should develop from the Reactor Coolant System to the component cooling loop, a high radiation alarm will actuate in the control room. If the leak is large, the component cooling surge tanks will fill and overflow to the waste hold-up tanks 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 1 is discussed in Appendix 6B.

Evaluation of Potential Leakage from the Reactor Coolant System

In considering potential leakage from the Reactor Coolant System containing primary coolant at high pressure, four categories should be considered:

- I - Leakage to the reactor coolant drain tank.
- II - Leakage to the pressurizer relief tank.
- III - Leakage to the containment environment.
- IV - Leakage to the interconnecting systems.

For clarity, each of these paths are discussed in turn.

I – Paths Directed to the Reactor Coolant Drain Tank (RCDT)

The routes directed to the Reactor Coolant Drain Tank may be summarized as follows:

1) Reactor Coolant System Loop Drains
2) Accumulator Drains
3) Auxiliary System Equipment Drains
4) Excess Let-down
5) Valve Leakoffs
6) Reactor Coolant Pumps Seal Leakage
7) Reactor Flange Leak-off

Of these paths, (1) through (4) do not present a leakage load on the RCDT during normal operation; leakage from the high pressure systems is not expected because of the use of double isolation valves. Path (5) through (7) merit some discussion.

Valve Leak-offs

Source – there are some twenty-six valves provided with leak-offs in the containment. Of these valves, only two valves in the Reactor Coolant System, one valve in the Chemical and Volume Control System and four valves in the Safety Injection System will normally have their valve stem packing subjected to pressure.

342	First isolation valve in the letdown line is normally fully open.
535 536	Pressurizer relief isolation valves which are normally fully open.
894A 894B 894C 894D	Accumulator isolation valves are normally open. The only leakage would be borated non-radioactive water.

Estimated Leakage – In the original FSAR, total leakage of reactor coolant fluid during normal power operation was conservatively estimated by assuming the use of the valve backseat for valves 342, 535, and 536 shown above as well as for valves 571A, 571B, 571C, and 571D (since removed by the RCS RTD Bypass Manifold Removal modification). Since backseats are capable of limiting leakage to less than 1.0 cc per hour per inch of stem diameter, assuming no credit for valve packing, it was assumed that the valves would leak at this rate. Hence for these original seven valves, a total leakage of 7 cc/hr was assumed. Consistent with industry practice, backseats are not used at IP3. The valve packing is used to minimize stem leakage. Actual RCS leakage is monitored and controlled to Technical Specification limits.

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 the panel. Any alarm on the WDS/BR panel causes annunciation of a single window on the main control board.

Reactor Coolant Pump Seals

Source – Charging flow is directed to the reactor coolant pumps via a seal-water-injection filter. It enters the pumps 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 25-30 psig and its temperature is increased from 130°F to about 136°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 our reactor coolant pumps) flow to a common manifold and then via a filter through the Seal Water Heat Exchanger (where the temperature is reduced to about 130°F) to the volume control tank.

The leak-off system between the No. 2 and No. 3 seals is considered to be part of the Reactor Coolant System. The leak-off system collects leakage passed by the No. 2 seal, 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 flow will be 3 gph per pump. This is the value specified in the Reactor Coolant Pump Equipment Specification.

Indication to Operation – Level instrumentation on the standpipes is provided to alert the operator to abnormal conditions. The standpipe consists of a pipe with an orificed overflow at the mid-point, 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 mid-point overflow. Excessive leakage will “back-up” in the standpipe until it overflows out the top. A level switch in the upper standpipe actuates an annunciator indicating excessive flow. A level switch in the lower standpipe causes annunciation of the opposite condition which could result in undesirable dry operation of the No. 3 seal.

Reactor Vessel Flange Leak-off

Source – The reactor vessel flange and head are sealed by two metallic o-rings. To facilitate leakage detection, a leak-off connection was placed between the two o-rings and a leak-off connection was placed beyond the outer o-ring. Piping and associated valving was provided to direct any leakage to the reactor coolant drain tank.

Leakage – During normal operation, the leakage will be negligible since it was 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.

II – Paths Directed to Pressurizer Relief Tank (PRT)

Source – The PRT condenses and cools the discharge from the pressurizer safety and relief valves. Discharge from smaller 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 CVCS letdown station relief valve.

Leakage – During normal operation, the leakage to the pressurizer relief tank is negligible since the valves were designed for essentially zero leakage at the normal system operating pressure, as specified in the respective valve equipment specifications.

Indication to Operator – For each valve, temperature detectors are provided 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 (see discussion for Category IV below). To further assist the operators in evaluating pressurizer relief tank conditions, there is a pressure recorder which takes pressure fluctuation data from pressurizer relief tank pressure transmitter and plots it in the CCR.

All of the pressurizer power operated relief valves and their associated motor-operated block valves have been provided with an acoustic monitoring system for position indication. Should there be any significant leakage from any of these valves, this system initiates an alarm in the control room.

III – Releases to the Containment Environment

Source – The main contributors of leakage to the containment environment may be listed as follows:

1) Valve stem leakages
2) Reactor Coolant Pump No. 3 seal leakage
3) Weld leakages
4) Flange leakages

Valve Stem Leakage

With exception of the pressurizer spray valves, the modulating valve within the containment are provided with leakoff connections which in turn are piped to the reactor coolant drain tank. Of the remaining valves which 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 provided with backseats which are capable of limiting leakage to less than one cubic centimeter per hour per inch 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 was originally estimated to be approximately 50 cc/hr.

- Modulating valves PCV-455A and PCV-455B were originally installed with intermediate packing leak-off lines piped to the RCDT. The original assumption was that this configuration would result in a maximum leakage to the RCDT of 6 cc/hr/valve through the packing. A modification resulted in replacement of the standard packing with a live loaded packing configuration to mitigate the potential of any leakage and capping of the existing leakoff line as no longer required. For conservatism, in the unlikely event of failure of the live-load packing, this assumption as to leakage quantity remains. However, the leakage path would now be to the containment environment in lieu of the RCDT.

- CH-342 was originally supplied with a lantern ring and a leakoff line that was routed to the Reactor Coolant Drain Tank (RCDT) in the event of packing failure. A live load packing configuration has been incorporated in the design of this valve and, consequently, the leakoff line has been retired in place and any potential leak path is to the Containment environment. (Reference NSE 98-3-156 CVCS, Rev. 1)

As a general rule, open valves are not backseated. The plant relies on packing to minimize stem leakage. Actual leakage is monitored and controlled to Technical Specification limits.

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 charging water and was anticipated to be of the order of 100 cc/hr per pump. This is the value specified in the Reactor Coolant Pump Equipment Specification. The No.3 seal leak-off is diverted to the local open drains and is thus released to the containment environment.

Weld Flanges

The welded joints throughout the system were subjected to extensive non-destructive testing; leakage through metal surfaces and welded joints is very unlikely.

Flange Joints

There are a number of flanged joints in the system; all of which will be 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 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.

Leakages – The main contributors to leakage to the containment environment are considered to be (1) and (2); experience with operating reactors has shown that following the normal pre-operational testing, leakage from these sources is negligible.

Conclusion

On the basis of the above, the analysis of the situation indicates a total leak rate to the containment environment of the order 450 cc/hr. For design purposes 50 lb/day (i.e., 1000cc/hr was assumed.

IV – Leakage to Interconnecting Systems

Each of the interconnecting systems are dealt with in turn.

SYSTEM	**DISCUSSION**
CVCS	This is a normally operating interconnected system redundancy for isolating purposes if required.
SS	In the event of sample valves failing close or seat, adequate redundancy is provided by containment isolation valves; the piping between the sets of valves is designed for RCS pressure.
RHR Hot Leg	Two isolation valves are provided; in the unlikely event of Connection leakage past the two valves, interconnecting piping is provided to enable pressure relief via the RHR loop relief valve to the pressurizer relief tank.
RHR Cold Leg	In the unlikely event of leakage past two sets of check valves into the RHR loop, pressure relief will take place via the RHR loop relief

	valves to the pressurizer relief tank.
SIS High Head Pump	In the event of leakage past two sets of check injection lines valves in any high head branch line, pressure relief will take place to the PRT via the relief valve in the SIS test line.
SIS Accumulator	Provisions have been made to check the Connections leak tightness of the leak tightness of the accumulator check valves. The implications of leakage past these valves are discussed in Section 6.2

*NOTE: The configuration of these RHR, SI and Accumulator connections is shown on Plant Drawing 9321-F-27503 [Formerly Figure 6.2-1A] and consists of in series check valves. Al of these check valves are categorized as Reactor Coolant System (RCS) Pressure Isolation (PIVs) and are listed in the Table 6.7-3. Periodic testing of these valves for loss leakage is required by Technical Specifications and reduces the probability of an inter-system LOCA (Reference 1). These tests implement the requirements set fourth in Reference 3 regarding the testing of SIS check valves and provide the basis for the rescission of Item A.5 of Reference 2.

On the table 6.7-3 pressure isolation valves S1-857A&G, Q&R, S&T, and U&V are presented as matched sets. These pairs of valves are configured in series on the HHSI non-BIT header cold leg branch lines, and deliver flow to the RCS through the Accumulator/RHR injection lines, with each high head branch connecting upstream (low pressure side) of SI-897 valves present the first pressure isolation valve barrier and are leak tested individually. The upstream pairs of 857 valves combine to form a second barrier and are tested as a pair. No credit is taken for leak tight integrity of an individual valve that is tested as a pair. This testing position was approved by the NRC in the Reference 4 supplemental evaluation.

Although leakage of primary fluid to the secondary system via the steam generator primary/secondary boundary is not expected during normal operation because of the conservative design of the U-tubes in the steam generator, any such leakage 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. (See Section 11.2)

6.7.2.4 Residual Heat Removal Loop

High containment air particulate beta and/or gamma activity or high radioactive gas activity will result in an alarm being activated by either the containment air particulate or radioactive gas monitors, respectively. The containment purge supply and exhaust duct valves and pressure relief line valves are closed. This prevents the release of radioactivity to the atmosphere outside the nuclear plant.

If a leak should develop from the residual heat removal loop into the component cooling loop, a high radiation alarm will actuate in the control room. If the leak is large, the component cooling surge tanks will fill and overflow to the waste hold-up tanks in the Primary Auxiliary Building.

Gross leakage from the portion 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.

6.7.2.5 Recirculation Loop

The containment purge supply and exhaust duct valves and pressure relief line valves are closed 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 was 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 lift and the effluent would be discharged to the waste holdup tank in the 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. Gross leakage from the external recirculation loop which does not flow into another closed loop will be drained to the auxiliary building sump and/or sump tank. From there it is pumped to the waste holdup tank.

6.7.2.6 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 and/or sump tank. Outside the containment, major leakage would be drained to the primary auxiliary building sump. From there it is pumped to the waste holdup tank.

Other provisions made for leakage from the component cooling loop are discussed in Section 9.3

6.7.2.7 Service Water System

Gross leakage from the service water system in the Primary Auxiliary Building will be directed by floor drains to the primary auxiliary sump tank, located outside containment. Pumps will then transfer this leakage to the waste holdup tank. A service water leak through the containment fan cooler units could result in containment flooding. The sump water level monitors would detect this flooding. The containment fan cooler condensate drains from the cooling coils and/or cooling coil leakage are collected and flow into a vertical standpipe slotted Weir System. The flow rate from the fan units are measured, based on the water depth flowing over the Weir.

If the drainage rates for all five units are nearly the same, it can be concluded that their water is condensate from the containment atmosphere. A particular unit with a high drainage rate, indicates a possible leak in one of the cooling coils. In series with each transmitter signal is an alarm for “C.B. Fan Cooler Cond. High Level.” The affected unit will be identified by individually monitoring the drainage flow from each unit using a rotary selector switch.

The containment sump under normal operation collects water from various drains within containment. This water is then pumped to the waste holdup tank when the sump level reaches the actuation level on the sump pump float switches. The sump pump flow meter measures instantaneous flow (an indication of proper pump performance) while the totalizer measures cumulative flow, which is used to indicate changes in sump accumulations.

When the level in the containment sump increases to the float actuation level, one sump pump will start. If the level continues to increase, the second pump starts. There is constant containment sump level indication provided in the Control Room via two independent level indicators. Also, through use of the containment sump flow meter and totalizer, any increase in sump accumulation because of a leak would be detected. If the containment sump level should approach an overflow condition, either because the two pumps cannot keep up with the leak or due to failure of both pumps, an alarm will annunciate in the Control Room via an additional level indicator installed at the top of the containment sump. At this point, water will overflow into the normally empty recirculation sump which will be indicated by the level indication in the control room.

Water collecting in the reactor pit is pumped out via two sump pumps, each pump discharging into an individual check valve that joins a common header and discharges into the containment sump. The reactor pit is normally kept dry. Level alarms in the reactor pit will annunciate in the Control Room if water should accumulate in this area. Also, an indicating light located in the control room will indicate when reactor pit sump pump number 31 is running.

References

1. WASH 1400

2. NRC Letter, A Schwencer to NYPA, Enclosing a Confirmatory Order dated February 11, 1980.

3. NRC Generic Letter 80-14, February 23, 1980, LWR Primary Coolant System Pressure Isolation Valves.

4. NRC letter Dated August 20, 1993, “Supplemental Safety Evaluation of the Second 10 Year Interval Inservice Testing Program and Associated Relief Request for Indian Point Nuclear Generating Unit No. 3 (TAC No. M85108).”

TABLE 6.7-1

CLASS 1 FLUID SYSTEMS FOR WHICH NO SPECIAL LEAK DETECTION IS PROVIDED

System	Remarks on Leakage Detection (Items a, b, and c are found in th of Section 6.7.1)
1. Residual Heat Removal (RHR)	Refer to items a, b, c, and Section 6.7.1.2
2. Component Cooling	Refer to item c and Section 6.7.1.2
3. Service Water	Refer to item c and Section 6.7.1.2
4. Auxiliary Feedwater	Visual
5. Waste Disposal	Auxiliary building sump pump operation and refer to item a.

TABLE 6.7-2

RESULTS OF LEAK TESTS OUTSIDE CONTAINMENT

System		Leak Rate
Volume Control Tank		0
Residual Heat Removal System		31.5 cc/min
Safety Injection System	3 cc/min	
Primary Sample System	1 drop/hr	
Post Accident Containment Sampling System		0
Modified Post Accident Containment Sampling System	10 cc/min	

*NOTE: The above data are provided as information only.

TABLE 6.7-3

REACTOR COOLANT SYSTEM (RCS)
PRESSURE ISOLATION VALVES (PIVs)

SI-838-A	SI-857D	SI-857M	SI-895B
SI-838-B	SI-857E	SI-857N	SI-895C
SI-838-C	SI-857F	SI-857P	SI-895D
SI-838-D	SI-857H	SI-857Q&R[(1)]	SI-897A
SI-857 A&G[(1)]	SI-857J	SI-857S&T[(1)]	SI-897B
SI-857B	SI-857K	SI-857U&W[(1)]	SI-897C
SI-857C	SI-857L	SI-895A	SI-897D

*NOTE: (1) See Section 6.7.2.3, Part IV for a discussion of these pairings.

6.8 HYDROGEN RECOMBINATION SYSTEM

On April 14, 2005, the NRC issued Unit 3 License Amendment 228 (Reference 12) 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.

In addition, Technical Specification requirements for the hydrogen monitoring instruments (HCMC-A and –B) were eliminated and replaced by a licensee commitment to maintain the monitors as reliable and functional through a preventive maintenance program. This license amendment also resulted in the Regulatory Guide 1.97 categorization for these instruments being changed from Category 1 to Category 3.

6.8.1 Design Bases

The design bases for the hydrogen control following a postulated Loss-of-Coolant Accident are as follows:

1) The system shall prevent the hydrogen concentration in the containment volume from exceeding 3% by volume following a design basis accident.

2) The system shall be capable of performing its design function in the containment environment following a design basis accident, i.e., withstand the accident and be capable of beginning operation as required when the containment pressure is near ambient.

3) The system shall be designed to withstand the design basis earthquake and still be capable of operation.

4) The system shall be sufficiently redundant and independent to the extent that no single active or passive failure can negate the minimum requirements of operation.

5) The system shall be testable during normal operating conditions of the plant.

6.8.2 System Design and Operation

System Description

The electric hydrogen recombiner systems installed at Indian Point 3 are engineered safety features to control the hydrogen generated in the containment following a Loss-of-Coolant Accident. The redundant systems are designed to seismic Class I Standards.

Two full rated, redundant and independent systems are provided. Each recombiner is powered from a separate safety related MCC. Each is capable of maintaining the ambient H_2 concentration at or below three volume percent (v/o).

Each recombiner system consists of a control panel located in the Control Room, a power supply cabinet located in the lower electrical cable tunnel, at elevation 34 ft., and a recombiner located on the operating deck at elevation 95 ft. in the Containment. The electric hydrogen recombiners are located in the southeast and southwest quadrants of the containment approximately 90° apart in the same location as the old flame type recombiners they are replacing. There are no moving parts or controls inside the containment. Heated air within the unit causes airflow by natural convection. The recombiner is a completely passive device.

To regulate the power supply to the recombiner, the power supply cabinet contains an isolation transformer and a controller. This equipment will not be exposed to the post-LOCA environment. The controls for the power supply are located in the Control Room and are manually actuated.

Each hydrogen recombiner consists of the following components:

1) A preheater section, consisting of a shroud placed around the central heaters to take advantage of heat conduction through the central walls, for preheating incoming air.

2) An orifice plate to regulate the rate of airflow through the unit.

3) A heater section, consisting of four banks of metal-sheathed electric resistance heaters, to heat the air flowing through it to hydrogen-oxygen recombination temperatures.

4) An exhaust chamber, which mixes and dilutes the hot effluent with containment air to lower the temperature of the discharge stream.

5) An outer enclosure to protect the unit from impingement by containment spray.

The recombiner unit is manufactured of corrosion-resistant, high-temperature material. The electric hydrogen recombiner uses commercial-type electric resistance heaters sheathed with Incoloy-800 which is an excellent corrosion-resistant material for this service. The recombiner heaters operate at significantly lower power densities than similar heaters used in commercial practice.

<u>System Operation</u>

Each recombiner is operated from its control panel located in the Central Control Room. Emergency operating procedures direct that the hydrogen concentration in the containment be monitored (by manual sampling or with the hydrogen analyzers) following a LOCA or high containment pressure condition and that the hydrogen recombiners be actuated in time to prevent reaching a hydrogen concentration of 4.0 volume percent. System operating procedures provide instructions for the operator to manually put the recombiners in service from the control panel. The recombiner, power supply panel and control panel are shown on Plant Drawings 9321-F-30064 and -30065 [Formerly Figures 6.8-2 and 6.8-2A]. The power panel for the recombiner contains an isolation transformer and a controller to regulate power into the recombiner. This equipment is not exposed to the post-LOCA containment environment.

To control the recombination process, the correct power input to bring the recombiner above the threshold temperature for recombination is set by adjusting a potentiometer located on the control panel. The correct power required for recombination depends upon containment atmosphere conditions and is determined when recombiner operation is required. For equipment test and periodic checkout, a temperature controller is provided on the control panel to automatically bring the recombiner to the recombination temperature.

The containment atmosphere is heated within the recombiner in a vertical duct, causing it to rise by natural convection. As it rises, replacement air is drawn through intake louvers downward through a preheater section which will temper the air and lower its relative humidity.

The preheated air then flows through an orifice plate, sized to maintain a 100 SCFM flowrate, to the heater section. The airflow is heated to a temperature above 1150°F, the reaction temperature for the hydrogen-oxygen reaction. Any free hydrogen present reacts with atmospheric oxygen to form water vapor. After passing through the heater section, the flow enters a mixing section, which is a louvered chamber where the hot gases are mixed and cooled with containment atmosphere before the gases are discharged directly into the containment. The air discharge louvers are located on three sides of the recombiner. To avoid recirculating previously processed air, no discharge louvers are located on the intake side of the recombiner. (Reference 10)

Tests have verified that the hydrogen-oxygen recombination is not a catalytic surface effect associated with the heaters but occurs due to the increased temperature of the process gases. As the phenomenon is not a catalytic effect, saturation of the unit cannot occur (References 1 and 9).

Instrumentation

The recombiners do not require any instrumentation inside the containment for proper operation after a LOCA. The recombiners are started manually after a LOCA. The sampling system is used to obtain containment atmosphere samples that indicate when the recombiners or the venting system should be actuated. Control measures can be initiated when the hydrogen concentration reaches 3.0 volume-percent.

The thermocouples and temperature transmitters located in the thermocouple splice box are used for testing and calibration only. Their failure will not affect the safety function of the recombiner.

Power Supply

Supply power for the electric recombiners is provided from safety related 480 MCC's 36C and 36B which are backed up by emergency diesel generators 31 and 32, respectively.

In order to prevent overloading of the diesel generators, the electric hydrogen recombiners will be deenergized on loss of offsite power or on a safety injection (SI) signal. Manual operator action will be required to restart the recombiners once adequate diesel generator capacity is available.

Post Accident Containment Atmosphere Sampling System

Following an accident, containment atmosphere is monitored for hydrogen concentration. A two channel redundant system is provided. Samples are taken from the plenum chambers of the containment recirculation fan units. Train A monitor takes suction from the plenum chambers of fan units 32 and 35. The train B monitor takes suction from the plenum chambers of fan units 31, 33 and 34.

Sampling should begin with the first 7 hours following diagnosis of a LOCA or MSLB inside containment. (Reference 11)

To assure that stratification effects or sample errors would not permit all or parts of the containment to hold hydrogen in excess of the lower flammable limit (4.1 v/o) when the measured concentration is 3.0 v/o, the following checks were made: It was determined that the minimum reliable air circulation capacity by three of the main recirculation fans within the containment could recirculate the entire containment air volume at an average rate of 4.8 times an hour (or 210,000 cfm capacity based upon pressure decay to ambient conditions for fan operation). But the calculated hydrogen generation rate during the first day post accident is 17,100 SCF yielding a ratio of air circulation to hydrogen generation in excess of 17,7000:1. Due to the decreased rate of hydrogen generation with time, the ratio increases to an even greater value before the hydrogen concentration in the containment reaches two volume percent. At a conservatively predicted generation rate, 60 hours are required to produce hydrogen in the amount of two percent of containment volume. During this same period, the entire atmosphere of the containment would have been recirculated, on the average, 288 times. Furthermore, the air handling system is designed to promote the interchange of air in all regions of the containment to avoid the possibility of accumulation of hydrogen in stagnant pockets or strata. For example, in the highest part of the containment dome (above the top spray ring), minimum air recirculation provides one air change approximately every 61 seconds. For these reasons it is concluded that the stratification error is negligible.

Based on the foregoing discussion, it is concluded that the three volume percent design concentration for operating the recombiner provides more than adequate margin for error associated with sampling the containment atmosphere. The calculated containment hydrogen concentration does not reach three volume percent until 10 days post accident, so it is highly unlikely that any significant concentration gradient will exist in the containment when the recombiner is started. Furthermore, since tests have been run with a full scale recombiner system at hydrogen concentrations up to and including 4.0 volume percent hydrogen, a hydrogen concentration between 2 and 3.5 volume percent at the recombiner suction would have no adverse effect on the recombiner operation.

6.8.3 Design Evaluation

The analysis of post-LOCA hydrogen production and accumulation in the containment is presented in Section 14.3.7. To determine the effectiveness of the recombiner, it is assumed that it will be activated before the containment hydrogen concentration reaches the design limit of 3 volume-percent. Starting the recombiner at below 3% provides substantial margin in time to reach the lower flammability concentration of 4.1%. The capacity of the recombiner, working in a 3% hydrogen environment, is at least 3 SCFM of hydrogen gas.

The results of the Regulatory Guide 1.7 analysis indicate that 3% hydrogen occurs at approximately 5.5 days (Figure 14.3-79) and the corresponding aggregate hydrogen production rate is approximately 1.6 SCFM (Figure 14.3-75). This production rate is well within the capacity of the recombiner. Further, because the hydrogen production rate decreases with time, the recombiner can easily accommodate hydrogen concentrations greater than 3%. Thus, starting a recombiner before the containment hydrogen concentration reaches 3% will ensure that the concentration remains well below the lower flammability limit.

Hydrogen stratification in the containment post-LOCA is minimized by the operation of the containment fan coolers. The containment coolers circulate air within the containment volume (Section 6.8.2). A containment sampling line is located near the inlet of each fan cooler. Assuming that 3 of 5 fan coolers are operating and that the flow rate per unit is 34,000 cfm (design flow rate during accident conditions) results in an average air flow of approximately 18.8 containment volumes per day per unit.

The recombiners are located in an open area of the containment on the 95' elevation operating deck.

The calculated average hydrogen concentration in containment reaches three volume percent in approximately 5.5 days (Safety Guide 7 basis, Figure 14.3-79). Based on this relatively small hydrogen production rate (average less than 5 SCFM, approximately 1.6 SCFM at 5.5 days, Figure 14.3-75) and the large air mixing rate described above, the bulk of the containment volume is expected to be well mixed, and no significant hydrogen concentration gradients are expected at either the hydrogen sampling points or at the recombiner locations.

Personnel Doses

The control panel for the hydrogen recombiner is located in the Control Room. The control room is designed to provide radiation protection for the operator following a design basis event. Doses to the control room operators following a large-break LOCA are evaluated in Section 14.3.5. The calculated doses are well within the limits specified in 10 CFR 50, Appendix A, General Design Criterion 19, i.e., 5 rem whole body (TEDE).

6.8.4 Tests and Inspections

The electric hydrogen recombiners underwent extensive testing in the Westinghouse development program. These tests encompassed the initial analytical studies, laboratory proof-of-principle tests, and full-scale prototype testing. The full scale prototype tests include the effect of:

- Varying hydrogen concentrations
- Alkaline spray atmosphere
- Steam effects
- Convection currents
- Seismic effects

A detailed discussion of these tests is provided in references 1 through 9.

Operational tests and inspections are performed to verify the operation of the control system and the ability of the heaters to achieve the required temperature. In addition, a channel calibration of all recombiner instrumentation and control circuits is performed every 24 months.

References

1) Wilson, J. F., “Electric Hydrogen Recombiner for Water Reactor Containments,” WCAP-7709-L (Proprietary), July 1971, and WCAP-7820 (Nonproprietary), December 1971.

2) Wilson, J. F., “Electric Hydrogen Recombiner for PWR Containments – Final Development Report,” WCAP-7709-L Supplement 1 (Proprietary), and WCAP-7820, Supplement 1 (Nonproprietary), April 1972.

3) Wilson, J. F., “Electric Hydrogen Recombiner for PWR Containments – Equipment Qualification Report,” WCAP-7709-L Supplement 2 (Proprietary), and WCAP-7820, Supplement 2 (Nonproprietary), September 1973.

4) Wilson, J. F., “Electric Hydrogen Recombiner for PWR Containments – Long Term Tests,” WCAP-7709-L Supplement 3 (Proprietary), and WCAP-7820, Supplement 3 (Nonproprietary), January 1974.

5) Wilson, J. F., “Electric Hydrogen Recombiner for PWR Containments,” WCAP-7709-L Supplement 4 (Proprietary), and WCAP-7820, Supplement 4 (Nonproprietary), April 1974.

6) Wilson, J. F., “Electric Hydrogen Recombiner Special Tests,” WCAP-7709-L Supplement 5 (Proprietary), and WCAP-7820, Supplement 5 (Nonproprietary), December 1975.

7) Wilson, J. F., “Electric Hydrogen Recombiner IEEE 323-1974 Qualification,” WCAP-7709-L Supplement 6 (Proprietary), and WCAP-7820, Supplement 6 (Nonproprietary), October 1976.

8) Wilson, J. F., “Electric Hydrogen Recombiner for LWR Containments Supplemental Test Number 2,” WCAP-7709-L Supplement 7 (Proprietary), and WCAP-7820, Supplement 7 (Nonproprietary), August 1977.

9) Wilson, J. F., “Qualification Testing for Model B Electric Hydrogen Recombiner,” WCAP-9346 (Proprietary) and WCAP-9347 (Nonproprietary), July 1978.

10) Electric Hydrogen Recombiner Model B Technical Manual, NYPA File 439-100058911.

11) NSE 97-3-269 HR

12) Letter from P. Milano, NRC to M. Kansler, Entergy: “Issuance of Amendment (228) Eliminating Requirements for Hydrogen Recombiners and Hydrogen Monitors”, dated April 14, 2005.

TABLE 6.8-1

DESIGN DATA FOR HYDROGEN RECOMBINERS

Quantity	2
Power (each) (maximum/minimum) (kW)	75/50
Model	B
Capacity (each) (minimum) (SCFM)	100
Heaters (per recombiner)	
Number	4 banks
Maximum heat flux (Btu/h-ft^2)	2850
Maximum sheath temperature (°F)	1550
Gas Temperatures	
Inlet (°F)	80-155
Outlet of heater section (°F)	1150 to 1400
Exhaust (°F)	Approximately 50° above ambient
Materials	
Outer structure	Type 300 series SS
Inner structure	Incology 800
Heater element sheath	Incology 800
Base skid	Type 300 series SS
Weight (lb)	4500
Codes and Standards	American Society of Mechanical Engineers Section IX, Underwri National Electric Manufacturers Association, National Fire Prote Institute of Electrical and Electronic Engineers 279, 308, 323, 34
Design Data for Power Supplies	
Quantity	2
Electrical Requirements	3 phase, 60 Hz, 480 VAC
Power (max.)	90 kW

APPENDIX 6A

IODINE REMOVAL EFFECTIVENESS EVALUATION OF CONTAINMENT SPRAY SYSTEM

1.0 INTRODUCTION

The Containment Spray System is an engineered safety system employed to reduce pressure and temperature in the containment following a postulated Loss-of-Coolant Accident. For this purpose, subcooled water is sprayed into the containment atmosphere through a large number of nozzles from spray headers located in the containment dome.

Because of the large ratio of spray drop surface area to containment volume, the spray system also serves as a removal mechanism for fission products postulated to be dispersed in the containment atmosphere. The source term used for the large-break LOCA assumes major core degradation and is defined in Regulatory Guide 1.183[(4)] as being a release of gap activity (noble gases, iodines, and alkali metal nuclides) over a half-hour period followed by a core melt that releases additional activity in those three nuclide groups plus additional nuclides over a 1.3 hour duration. The iodine activity is assumed to be primarily in the particulate form (cesium iodide) with small fractions of the iodine in the elemental and organic forms. Nuclides other than the iodines and noble gases are all modeled as being in the particulate form. The sprays are effective at removing elemental iodine and particulates from the containment atmosphere but the organic iodine and the noble gases are not subject to removal by the sprays.

2.0 CONTAINMENT SPRAY IODINE REMOVAL MODEL

Spray removal coefficients for particulates and for elemental iodine were calculated for both the injection and recirculation modes of containment spray operation. Calculation of the containment spray removal coefficients is based on the models documented in NURG-0800, SRP Section 6.5.2[(1)]. The removal coefficients were credited in the Large Break LOCA radiological consequences as documented in Section 14.3.5.

2.1 Elemental Iodine Removal

Elemental iodine removal during the spraying of a fresh solution is highly dependent on the rate at which the fresh solution surface is introduced into the containment atmosphere. The elemental iodine removal coefficient is given by:

$$\lambda_s = \frac{6K_gTF}{VD}$$

Where:
- λ_s = Elemental spray removal coefficient, hr^{-1}
- K_g = Gas-phase mass transfer coefficient, ft/min
- T = Time of fall of the spray drops, min
- F = Volume flow rate of sprays, ft^3/hr
- V = Containment sprayed volume, ft^3
- D = Mass mean diameter of the spray drops, ft

The gas phase mass transfer coefficient taken from BNL – Technical Report A-3778[2], is 3 meter/min or 9 84 ft/min. The average spray droplet fall time of 0.167 minutes was determined based on a spray fall height of 118.5 ft and a spray flow rate of 2,200 gpm. The volumetric flow rate of the sprays is 17,646 ft^3/hr. The containment sprayed volume is 2,088,000 ft^3. The mass mean diameter of the spray droplets was calculated to be 0.003675 ft.

From these inputs an elemental iodine removal coefficient of 22.7 hr^{-1} was calculated. The value used in the large break LOCA radiological consequences was reduced to 20 hr^{-1}, which is the upper limit specified in SRP 6.5.2 [1].

During the recirculation spray mode the spray flow rate is reduced to 962 gpm with a resulting reduction in iodine removal coefficient to 9.9 hr^{-1}. However, during recirculation the spray solution will gradually become loaded with elemental iodine which will limit the capacity of the spray to remove airborne iodine. The spray removal coefficient would be inversely proportional to the DF achieved for elemental iodine. Thus, when recirculation spray is first credited there is still so little elemental iodine in the sump solution that it would be appropriate to use the removal coefficient of 9.9 hr^{-1}. But, when the DF approaches its defined limit, the removal coefficient would be only a small fraction of its original value. The impact of this varying nature of the removal coefficient can be approximated by setting the removal coefficient to one half of the calculated value (5.0 hr^{-1}). This has the effect of reducing credit for removal of elemental iodine early in the recirculation phase but increasing it in the latter stages of spray removal.

2.2 Elemental Iodine Decontamination Factor

In SRP 6.5.2[1], the maximum removal of elemental iodine is that of a decontamination factor (DF) of 200. The DF is related to the total release to the containment atmosphere which takes place over a 1.8 hour period (based on the source term model provided in Regluatory Guide 1.183[4].

The elemental iodine decontamination factor for the containment atmosphere achieved by the containment spray system is determined by the following equation:

$$DF = 1 + [V_s/(V_{c-}V_s)]\ (PC)$$

Where:
DF = Decontamination
V_s = Volume of liquid in the containment sump, ft^3
PC = Partition coefficient for iodine in water
V_c = Containment net free volume, ft^3

The post LOCA sump volume is calculated to be 386,000 gallons. The partition coefficient for iodine in water is conservatively assumed to be 10,000 based on a sump pH of 7.0. The net free containment volume is 2,610,000 ft^3.

The calculated spray decontamination factor is slightly greater than 200. This is reduced to 200 as specified above for use in calculating the large break LOCA radiological consequences. The elemental iodine decontamination factor of 200 is reached at approximately 166 minutes in the calculation of the LOCA doses. No credit for elemental iodine removal by containment sprays is taken after 166 minutes.

2.3 Particulate Iodine Removal

The spray removal coefficient for particulates is given by:

$$\lambda_p = \underline{3hFE}$$

2VD

Where: λ_p = Particulate spray removal coefficient, hr^{-1}
H = Drop fall height, ft
F = Volume of flow rate of sprays, ft^3/hr
V = Containment sprayed volume
E/D= Ratio of a dimensionless collection efficiency E to the average spray drop Diameter D.

The spray fall height is 118.5 ft; the volumetric flow rate of the sprays during the injection phase is 2,200 gpm (17,646 ft 3/hr). The containment sprayed volume is 2,088,000 ft^3. From SRP 6.5.2[(1)], the E/D ratio is 10 m^{-1} until a DF of 50 is attained (i.e., when the aerosol activity released from the core is reduced by a factor of 50) and is a factor of ten lower after a DF of 50 is achieved.

From these inputs the particulate iodine removal coefficient was calculated to be 4.6 hr^{--1} for a particulate iodine DF less than or equal to 50. The injection phase of the containment spray system is in operation until 45 minutes into the accident. With the activity release from the core having a duration of 1.8 hours[(4)], most of the particulates are not released to the containment atmosphere until after the spray injection phase is over. During the spray recirculation phase the volumetric spray flow is reduced to 962 gpm (7,716 ft^3/hr) and the resulting removal coefficient for particulates is 2.0 hr^{-1} (after a DF of 50 is reached, the removal coefficient drops to 0.2 hr^{-1}).

References

1. NUREG-0800, Standard Review Plan, Section 6.5-2, “Containment Spray as a Fission Product Cleanup System,” Revision 2, U.S. Nuclear Regulatory Commission, December 1988.

2. BNL – Technical Report A-3788, “Fission Product Removal Effectiveness of Chemical Additives in PWR Containment Sprays,” Davis, Nourbakhsh, and Khatib-Rahbar, dated 8/12/86.

3. Deleted

4. Regulatory Guide 1.183, “Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, “ July 2000, U.S. Nuclear Regulatory Commission.

APPENDIX 6B

PRIMARY SYSTEM LEAK DETECTION INTO INDIAN POINT 1 CONTAINMENT VESSEL

[Historical Information Only]

CAUTION

Appendix 6B shows the leak detection methodology to **Indian Point 1.** The assumptions and constants contained in this appendix **should not be used for Indian Point 3 leak detection calculations**.

1.0 INTRODUCTION

Small leaks 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 is the radioactive air particulate monitor which continuously samples the air in the containment cooling system. The purpose of the containment cooling system is to maintain proper ambient temperatures for equipment in the containment vessel. This system takes air from the upper elevations of the vessel and recirculates it through cooling coils on the suction side of the supply fan. This air is then discharged at a rate of 40,000 cfm. The turnover rate of air in the containment vessel as a result of this system is approximately once every hour. By sampling air from the discharge of the containment cooling system supply fan, leak rates as small as 0.3 gph (20 cc/minute) could be detected.

Another detector, the radiogas monitor, sampling air from the same position as the air particulate monitor, continuously analyzes air from the containment cooling system for gaseous radioactivity. This monitor is capable of detecting a leak rate of about 100 gph (6500 cc/minute).

In addition to measuring changes in the radioactivity of the containment vessel, dew point sensors continuously sample 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 gph (250 cc/minute) 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 taken 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 taken manually to back up these detectors were accurate to within a factor of 2.

Other methods for detecting and locating primary system leakage include 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 manually taken samples for tritium radioactivity in condensed moisture from the containment vessel.

2.0 SAMPLE CALCULATIONS

To determine the leak rate utilizing measurements from the instrumentation discussed in Paragraph 1.0 the following method must be applied:

Assumptions

The calculations are based on the assumption that:

1) Uniform mixing in the Containment occurs within one hour after initiation of the leak when one cooling fan is in service at a flow of 40,000 cfm.

2) The smallest significant change for the radiogas monitor which reflects the presence of a leak is 1 count per second (cps), which is equivalent to an increase in activity of 3×10^{-7} µc/cc of air.

3) The smallest significant change for the particulate monitor which reflects the presence of a leak is 8 cps, which is equivalent to an increase in activity of 8×10^{-9} µc/cc of air.

4) A period of eight hours is used to evaluate these changes, which provides time for checking the instrumentation and determining the cause of the leak. The eight hour period is predicated for determining the magnitude of small leaks; large leaks would be evaluated much sooner.

Basic Data Used for Calculation

1) Containment volume: 1.8×10^{-6} ft^3 (5.05×10^{10}cc)

2) Normal containment environment:
    a) Average temperature: 120° F
    b) Dewpoint temperature 70° F
    c) Water content: 0.016 lbs of water/lb of dry air

3) Normal radioactivity in the containment cooling system:
    a) Radiogas: 2.5 cps (7.5×10^{-7} µc/cc)
    b) Particulate: 16 cps (1.6×10^{-3} µc/cc)

4) Normal primary coolant radioactivity after one hour:
    a) Radiogas: 5×10^{-3} µc/ml of H_2O
    b) Particulate: 5×10^{-2} µc/ml of H_2O

Calculations

Dewpoint

The smallest leak that can be detected will result in an increase in the dewpoint reading from 70 F to 74 F. The water content of the containment atmosphere at a 74 F dewpoint would be 0.018 lbs of water per lb of dry air.

Letting

X = the leak rate into the Containment in gph
h_a = the water content as a dewpoint of 70 F
h_b = the water content as a dewpoint of 74 F
V_c = the volume of the Containment in ft^3

	pa	=	the density of the containment atmosphere in lb/ft^3
	t	=	the evaluation period, and
	k	=	8.3 lbs/gallon for water

Then:

	X	=	$(h_b\text{-}h_a)$ Vc p_a/tk
or	X	=	(0.018 – 0.016) (1.8 x 10^6)(0.081 x 109/121)(8)(8.3)
		=	3.95 gph (100 gpd)

Radiogas Activity

For the smallest significant change for the radiogas monitor (1 cps) the corresponding leak rate could be determined as follows:

Let	Y	=	the leak rate into the Containment in gph
	C_g	=	the radiogas activity increase (3.0 x 10^{-7} µc/cc of air)
	V^c	=	the volume of the Containment if cc
	T	=	the evaluation period
	I^g	=	the primary coolant radioactivity after one hour, and
	k	=	3.8 x 10^3 ml/gal for water

Then:

	Y	=	C_gV_c/tI_gk
or	Y	=	(3.0 x 10^{-7})(5.05 x 10^{10})/(8)(5 x 10^{-3})(3.8 x 10^3)
		=	99.8 pgh (2400 gpd)

Particulate Activity

For the smallest significant change for the particulate monitor (8 cps) the corresponding leak rate could be determined as follows:

Let

	Z	=	the leak rate into the Containment in gph
	C_p	=	the particulate activity increase (8 x 10^{-9} µc/cc of air)
	V_c	=	the volume of the Containment in cc
	t	=	the evaluation period
	I_p	=	the primary coolant radioactivity after one hour, and
	k	=	3.8 x 10^3 ml/gal for water

Then:

	Z	=	C_pV_c/tI_pk
or	Z	=	(8 x 10^{-9})(5.05 x 10^{10})/(8)(5 x 10^{-2})(3.8 x 10^3)
		=	0.265 gph (6 gpd)

APPENDIX 6C

CHARCOAL FILTER REMOVAL OF METHYL IODIDE BY ISOTOPIC EXCHANGE

1. INTRODUCTION

It was postulated that radioactive iodine in organic forms, principally methyl iodide, exists in the containment atmosphere following a loss of coolant accident. Engineered Safety Features which can remove this radioactive methyl iodide are reactive sprays, charcoal filters by absorption, and impregnated charcoal filters by isotopic exchange. At the present time, no credit is taken for methyl iodide removal by sprays or charcoal filter absorption. Since isotopic exchange of radioactive methyl iodide with iodine impregnated charcoal filters is the only form of active removal considered, a model which accurately describes this process has been derived. It should be noted that the original charcoal filters installed at Indian Point 3 have been replaced. The carbon used in these replacement filters is co-impregnated with triethylene diamine (TEDA) and potassium iodide (KI) to enhance the ability to absorb organic radioiodine compounds. The bases of this evaluation are nevertheless valid and the results applicable. The installed Nuclear Grade Activated Charcoal is tested in accordance with ASTM D3803-1989, per the IP3 response to Generic Letter 99-02.

The isotopic exchange reaction between the radioactive methyl iodide in the containment atmosphere and the iodide impregnant on the charcoal filters is of the form:

$$CH_3I^* + Ic \Leftrightarrow CH_3I + IC^* \quad (1)$$

The equilibrium constant for this reaction is defined as:

$$Keq = [CH_3II]\ \{Ic^*]\ /\ [CH_3I^*]\ [Ic] = 1 \quad (2)$$

where

$[CH_3I]$	-	is the grams of iodine as methyl iodide in the containment atmosphere
$[CH_3I^*]$	-	is the curies of iodine as methyl iodide in the containment atmosphere
[Ic]	-	is the grams of iodine impregnant on the filter
[Ic*]	-	is the curies of iodine on the filter impregnant

2. MATHEMATICAL BASIS FOR THE MODEL

From a material balance on the balance on the containment atmosphere, the rate of change of specific activity of iodine as methyl iodide (in the containment atmosphere) can be expressed as:

$$dC_1\ /\ dt = [-\lambda_F\ (C_{in} - C_{out}) - ([\lambda_D + \lambda_L\ C1] \quad (3)$$

where:

C_1	-	is the specific activity, defined as curies of a species per gram of the same species, of iodine as methyl iodide in the containment atmosphere.
λ_F	-	rate constant for isotopic exchange reaction (hr^{-1})

λ_D - decay rate (hr^{-1})
λ_L - containment leak rate (hr^{-1})
C_{in} - specific activity of methyl iodide in air stream entering filter
C_{out} - specific activity of methyl iodide in the air stream exiting from the filter

The radioactive iodine on the filter impregnant can expressed as:

$$dC_2/dt = [\lambda_F (C_{in'} - C_{out'}) - \lambda_D (C_2)] \qquad (4)$$

where

C_2 - is the iodine specific activity of the filter impregnant

The term $\lambda_F (C'_{in} - C'_{out})$ is the rate of change of specific activity of the impregnant by isotopic exchange in units of curies per gram of iodine impregnant per hour.

An analytical expression is obtained by integrating equations (3) and (4) with time and assuming:

a) The specific activity of the air stream exiting from the filter is in equilibrium with the specific activity on the filter.

b) The amount of stable iodine as methyl iodide at time zero remains constant with time.

3. APPLICATION OF CHARCOAL FILTER MODEL TO INDIAN POINT 3

A study was performed to evaluate the effects of the charcoal filter model on the removal of radioactive methyl iodide from the containment atmosphere and the buildup of radioactive iodine on the filter impregnant by isotopic exchange. This evaluation used the following parameters:

1) Plant Power – 3116 MWt

2) Containment Free Volume – 2.61 x 10^6 ft^3

3) 6.0 grams of iodine per MWt in the core after 830 days of operation

4) Core Inventories per MWt after 830 days of operation, as follows:
   I-131 – 2.51 x 10^4 curies/MWt
   I-132 – 3.81 x 10^4 curies/MWt
   I-133 – 5.63 x 10^4 curies/MWt
   I-134 – 6.58 x 10^4 curies/MWt
   I-135 – 5.10 x 10^4 curies/MWt

5) 2.5% of core iodine released to the containment atmosphere as methyl iodine

6) 36 charcoal filter cells with 2 pounds of iodine impregnant per cell, i.e., minimum safeguards

7) Containment leak rate schedule
   a) 0.001/day → 0-24 hrs
   b) 0.00045/day → 24-720 hrs

8) Air flow through charcoal filters – 24000 CFM

9) The iodide removal rate-constant, λ_F in Eqs. (3) and (4), is defined as the air flow through the filters (fraction of the containment volume per hour) times the filter efficiency for methyl iodides. A filter efficiency of 70% was assumed in this analysis.

   Test data from the charcoal manufacturer, for the original charcoal filters which were installed at Indian Point 3, impregnated with KI_3, show a removal efficiency of 98.9% for methyl iodide under conditions (130°C/90% RH) similar to post-LOCA conditions.

   Test data from two charcoal manufacturers for charcoal filters coimpregnated by KI and TEDA, of the type installed at Indian Point 3, show a slightly higher removal efficiency of 99%.6% for methyl iodides.

   For either types of impregnated charcoal filters, the assumption of 70% removal efficiency is thus conservative. Results of the analysis are presented in Figures 6C-1 through 6C-10.

4. DISCUSSION OF RESULTS

Figures 6C-1 through 6C-5 show the decrease of specific activity of iodine as methyl iodide in the containment and the buildup of radioactive iodine on the filter impregnant by isotopic exchange over a 30-day period. Figures 6C-6 through 6C-10 correspond to Figures 6C-1 through 6C-5 except that the latter five figures give the detailed specific activity breakdown for the 0-2 hour period.

5. CONCLUSIONS

It can be seen from the specific activity plots that all isotopes reach an equilibrium value between the filters and the containment atmosphere after which decay and leakage are the only iodine removal mechanism.

APPENDIX 6D

COMPATIBILITY OF MATERIALS UNDER EXPOSURETO THE POST-ACCIDENT CONTAINMENT ENVIRONMENT

1.0 DEFINITION OF POST-ACCIDENT 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:

a) The integrity of the materials of construction of engineered safeguards equipment when exposed to post Design Basis Accident (DBA) conditions, and

b) 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 DBA environmental conditions of temperature, pressure, radiation and chemical composition are described in the following sections. The time-temperature-pressure cycle used in the materials evaluation was most conservative since it considered only partial safeguards operation during the DBA. The containment spray and core cooling solutions considered herein include both the design chemical compositions and the design chemical compositions contaminated with deterioration products and 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 granular sodium tetraborate for pH control was implemented at Indian Point 3 (IP3) by License Amendment 236 (Reference 27) to address sump clogging issues raised by Generic Letter 2004-02 (Generic Safety Issue 191). Replacement of sodium hydroxide with sodium tetraborate was the most comparable alternative to sodium hydroxide for pH control. Appendix 6D was updated where appropriate to incorporate this change and much of the updated information was drawn from WCAP-16596-NP. Information for the original sodium hydroxide additive has been retained for historical purposes.

1.1 Design Basis Accident Temperature-Pressure Cycle

Figure 6D-1 presents 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 containment spray pumps is considered to inject 3000 gpm of boric acid solution into the Containment. When the Refueling Water Storage Tank is empty, the recirculation pumps supply a flow of 2400 gpm to the spray headers. Recirculation flow through one recirculation pump is cooled in the residual heat exchanger.

Figures 6D-2 and 6D-3 present materials evaluation test conditions for the Containment and core environments, respectively.

Evaluations of materials were performed, in general, for conditions either simulating the time-temperature conditions of Figure 6D-2 or conservatively considering higher temperatures for longer periods. The basis for each material evaluation is described with the discussion of its particular suitability.

1.2 Design Basis Accident Radiation Environment

Evaluation of materials for use inside containment included a consideration of the radiation stability requirements for the particular materials application. Figures 6D-4 and 6D-5 present the post DBA containment atmosphere direct gamma dose rate and integrated direct gamma dose, respectively. These data were calculated on the basis of a core meltdown and by assuming the following fission product fractional releases consistent with the TID-14844 model:

Noble Gases	Fractional Release	1.0
Halogens	Fractional Release	0.5
Other Isotopes	Fractional Release	0.01

1.3 Design Chemical Composition of the Emergency Core Cooling Solution

Nuclear Regulatory Commission Branch Technical Position MTEB 6-1 (Reference 24) 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 ECC system design uses alkaline adjusted boric acid solution as the spray and core cooling fluid. Initially the injection and spray solution are not alkaline adjusted since the RWST contains only boric acid and not sodium tetraborate. It is not until recirculation from the sump, where the injected water has dissolved sodium tetraborate that the spray and core cooling fluid is alkaline adjusted.

Plant design 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. IP3 has since converted to sodium tetraborate granules by License Amendment 236 (Reference 27) to address sump screen clogging issues. 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 containment 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 4.6. As the initial spray solution and subsequently the recirculation solution comes in contact with the sodium tetraborate, the sodium tetraborate dissolves raising the pH of the sump solution to an equilibrium value above 7.0.

8096 pounds of sodium tetraborate is sufficient to assure a post-LOCA sump pH above 7.0 (Reference 25). Dissolution testing performed in support of the buffer change to sodium tetraborate shows that it will dissolve rapidly during the injection phase when submerged in the boric acid solution.

1.3.1 Alkaline Sodium Borate

Plant designs which utilize the containment spray solution for fission product iodine removal, as well as containment cooling, include provisions for injection of chemical additive (sodium hydroxide) to the Emergency Core Cooling System. Boric acid solution, containing approximately 2000 ppm boron, is pumped from the Refueling Water Storage Tank to the Containment System by means of the safety injection pumps, residual heat removal pumps and containment spray pumps.

The chemical additive tank contains sufficient sodium hydroxide solution such that when its contents, the Refueling Water Storage Tank contents, and the Reactor Coolant System fluid are mixed, the resulting pH will be between 7.9 and 10.0. During the initial 30 to 60 minutes of spraying, the spray solution may be at a pH of about 10.

Figure 6D-6 shows a plot of sodium hydroxide concentration versus pH for a 2500 ppm boron solution. Tentative limits of pH between 7.9 and 10 for the mixed spray solution are indicted on this figure.

For the purpose of materials evaluation in the design chemistry solution, the following concentration/time relationship was considered:

0 to 1 hour	:	pH = 10.0,	Boron 2500 ppm
1 hour to 12 months	:	pH = 9.0,	Boron 2500 ppm

The solutions were considered aerated through the entire exposure period.

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 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 which were identified as conceivably being present in the emergency core cooling solution during recirculation.

To identify the trace elements inside containment which 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 6D-1 presents a listing of the major periodic groups of elements. Elements which 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 latter sections.

2.0 MATERIALS OF CONSTRUCTION IN CONTAINMENT

All materials in the Containment were reviewed from the standpoint of insuring the integrity of equipment and to insure that deterioration products of some materials do not aggravate the accident condition. In essence, therefore, all materials of construction inside containment must exhibit resistance to the post-accident environment or, at worst, contribute only insignificant quantities of trace contaminants which have been identified as potentially harmful to vital safeguards equipment.

Table 6D-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 3.0 of this Appendix, 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 inside containment.

Table 6D-3 lists the Nuclear Steam Supply System aluminum inventory which is considered 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 600 series aluminum alloys are generally the major types found inside containment. This inventory reflects the determination to exclude as much as practicable the use of aluminum in the Containment.

3.0 CORROSION OF METALS OF CONSTRUCTION IN DESIGN BASIS ECC SOLUTION

Emergency core cooling components are austenitic stainless steel and hence, are quite resistant to corrosion by the alkaline sodium borate solution, as demonstrated by corrosion tests performed at Westinghouse and ORNL.[(1)] The general corrosion rate for type 304 and 316 stainless steel was found to be 0.01 mils/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 ORNL[2,3] to confirm the low values.

Extensive testing was also performed on other metals of construction which are found in the Reactor Containment. Testing was performed on these materials to ascertain their compatibility with the spray solution at design post-accident 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 5.0 of this Appendix. 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 – 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
Zr-4	0.004
Inconel 718	0.003
Copper	0.015
90 – 10 Cu-Ni	0.020
70 – 30 Cu-Ni	0.006
Galvanized Carbon Steel	0.051
Brass	0.010

Tests conducted at ORNL[2,3] also have verified the compatibility of various materials of construction with alkaline sodium borate solution. In tests conducted at 284 F, 212 F, and 130 F, stainless steels, Inconel, cupronickels, Monel, and Zircaloy-2 experienced negligible changes in appearance and negligible weight loss.

Corrosion tests at both Westinghouse and ORNL have shown copper suffers only slight attack when exposed to the alkaline sodium borate solution at DBA conditions. The corrosion rate of copper, for example, in alkaline sodium borate solution at 200 F is approximately 0.015 mil/month. [1] The corrosion of copper in an alkaline sodium borate environment under spray conditions at 284 F and 212 F has been reported by ORNL. Corrosion penetrations of less than 0.02 mil were observed after 24-hour exposure at 284 F (see Reference 3, Table 3.13), and a corrosion rate of less than 0.3 mil per month was observed at 212 C (see Reference 2, Table 3.6).

The corrosion of copper in the post-accident 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 ECC solution.

The corrosion rate of galvanized carbon steel in alkaline sodium borate (3000 ppm B, pH 9.3) is also low. Tests conducted in aerated solutions showed the corrosion rate to be 0.003 mil/month (0.046 $mg/dm^2/hr$) and 0.002 mil/month (0.036 $mg/dm^2/hr$) for temperatures of 200 F and 150 F, respectively. It can be seen, therefore, that the corrosion of zinc (galvanized) in alkaline borate solution is minimal and will not contribute significantly to the post-accident hydrogen buildup.

Consideration was given to possible caustic corrosion of austenitic steels by the alkaline solution. Data presented by Swandby[(4)] (Figure 6D-7) show that these steels are not subject to caustic stress cracking at the temperature (285 F and below) and caustic concentrations (less than 1 percent by weight) of interest. It can be seen from Figure 6D-7 that the stress cracking boundary minimum temperature as defined by Swandby coincides with a high free caustic concentration (~40%) and is considerably above the long term post-accident design temperature (~80 F). Further, from Figure 6D-7, a temperature in excess of 500 F is required to produce stress corrosion cracking at a sodium hydroxide concentration greater than 85%.

It should be noted, considering the possibility of caustic cracking of stainless steel, that the sodium hydroxide-boric acid solution is a buffer mixture wherein no free caustic exists at the temperatures of interest, even when the solution is concentrated locally through evaporation of water. Hence, the above consideration is somewhat hypothetical with regard to the post-accident environment.

4.0 <u>CORROSION OF METALS OF CONSTRUCTION BY TRACE CONTAMINANTS IN EMERGENCY CORE COOLING (ECC) SOLUTION</u>

Of the various trace elements which 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.

The use of mercury or mercury bearing items, however, is prohibited inside containment. This includes mercury vapor lamps, fluorescent lighting and instruments which employ mercury for pressure and temperature measurements and for electrical equipment. Potential sources of mercury, therefore, are excluded from containment and hence, no hazard from this element is recognized.

The possibility of chloride stress corrosion of austenitic stainless steels was also considered. It is believed that corrosion by this mechanism will not be significant during the post-accident period for the following reasons:

1. <u>Low Temperature of ECC Solution</u>

   The temperature of the ECC 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 indicates 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 percent by weight solutions of $MgCl_2$ with temperature decrease from 310 F to 272 F. Staehle and Latanision[(6)] present data which 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] which 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 my occur in the ECC solution.

2. <u>Low Chloride Concentration of ECC Solution</u>

It is anticipated that the chloride concentration of the ECC solution during the post-accident period will be low. Throughout plant construction, surveillance was maintained to ensure that the chloride inventory inside containment would be maintained at a minimum. Controls on use of chloride bearing substance in containment included the following:

a) Restriction in chloride content of water used in concrete
b) Prohibition of use of chloride in cleaning agents for stainless steel components and surfaces
c) Prohibition of use of chloride in concrete etching for surface preparation
d) Use of non-chloride bearing protective coatings inside containment
e) Restriction of 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] presented data of Staehle which 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 $MgCl_2$.

Instances of chloride cracking at representative ECC solution temperatures and at low solution chloride concentration have generally been on surfaces on which concentration of the chloride occurred. In the ECCS, 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% relative humidity.

3) <u>Alkaline Nature of the ECC Solution</u>

The ECC solution will have a solution pH greater than 7.0 after dissolution of the sodium tetraborate additive stored in containment. 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 C1 (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 tests runs by Scharfstein and Brindley showed the influence of solution pH on higher chloride concentrations up to 550 ppm Cl; 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 due to the hydrolysis yielding pH 8.8 and not to 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, and 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 post-accident environment is very low indeed.

Westinghouse corrosion tests (Reference 26) showed that at pH 7, 100 ppm CI, 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 (Reference 24) 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 section6D.1.3, the initial pH of the solution will be approximately 4.5 and will increase as the sodium tetraborate dissolves. Therefore, for some period of time the spray solution will be below 7.0. The sodium tetraborate baskets will be submerged within one hour as the RWST is emptied following a LOCA. This envelopes start of recirculation as shown in Table 14.3-55. Dissolution testing in 2500 ppm boric acid solution has shown that sodium tetraborate dissolves rapidly when submerged in the boric acid solution (Reference 23). Westinghouse corrosion tests (Reference 26) indicate that the minimum time to crack (100% crack of a 304 SS welded single U-bend) in a pH 4.5, 100 ppm CI 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 sample). Hence, cracking will not occur during the relatively brief injection spray and safety injection period.

During recirculation, as the solution washes over the containment structure 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 build up to critical concentrations (if they ever do). Since sodium tetraborate has been shown to dissolve rapidly in boric acid solution, the time required to adjust the sump solution pH to greater than 7.0 by dissolution of sodium tetraborate is less than 8 hours, therefore pH adjustment occurs 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 ECC solution during the post-accident period will not impair its operability from the standpoint of chloride stress corrosion cracking. The environment of low temperature, low chlorides and high pH (>7.0) which will be experienced during the post-accident period will not be conducive to chloride cracking.

5.0 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 post-accident condition temperatures with the liberation of hydrogen gas. A number of corrosion tests were conducted in the Westinghouse laboratories and at ORNL facilities. A review of applicable aluminum corrosion data is given in Table 6D-4 and on Figure 6D-8.

5.1. Aluminum Corrosion Products in Alkaline Solution

The corrosion of aluminum in alkaline solution expected following a design basis accident (DBA) has been shown to proceed with the formation of aluminum hydroxide [14, 15, 16] and the aluminate ion, as well as with the production of hydrogen gas.

The DBA conditions expected for Indian Point 3 include the establishment of an alkaline ECC solution having a total volume of liquid of 4.47×10^5 gallons after actuation of the Engineered Safety Features.

As mentioned above, aluminum is known to corrode in alkaline solutions to give a precipitate of $Al(OH)_3$ which, in turn, can re-dissolve in an excess of alkali to form a complex aluminate.

Van Horn[14] noted that the precipitation of $Al(OH)_3$ begins about pH4 and is essentially complete at pH7. A further increase in pH to about 9 causes dissolution of the hydroxide with the formation of the aluminate.

It can be seen, therefore, that the solubility of aluminum corrosion product is a function of the pH of the environment. Consistent with this, the corrosion of aluminum is also strongly dependent on the solution pH since when the corrosion products are dissolved from the metal surface, corrosion of the base metal can proceed more freely.

Figure 6D-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 DBA condition here would include the reaction of aluminum in alkaline solution to from aluminum hydroxide, i.e.,

$$2\,Al + 6\,H_2O \rightarrow 2\,Al\,(OH)_3 + 3\,H_2 \qquad (1)$$

and dissolution of the hydroxide to form the aluminate, i.e.,

$$Al\,(OH)_3 \rightleftharpoons AlO_2^{-} + H^{+} + H_2O \qquad (2)$$

A knowledge of the solubility product of the aluminum hydroxide in an alkaline solution allows the determination of the solubility expected for the hydroxide in the DBA environment.

Deltombe and Purbaix[17] have determined the solubility product of aluminum hydroxide. Using the value of 2.28 x 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 form Equation (2):

$$Al(OH)_3 \rightleftharpoons AlO_2^{-} + H^{+} + H_2O$$

$$K_{sp} = \left[AlO_2^{-}\right]\left[H^{+}\right]$$

$$2.28 \quad x \quad 10^{-11} = \left[AlO_2^{-}\right]\left[H^{+}\right]$$

at pH = 9.3

$$\left[AlO_2^{-}\right] = \frac{2.28 \quad x \quad 10^{-11}}{5 \quad x \quad 10^{-10}} = 4.6 \quad x \quad 10^{-2} \text{ moles/liter}$$

Therefore, the solubility of $A1(OH_3)$ in a pH 9.3 solution at 25 C (77^{o} F) is equal to 4.6 x 10^{-2} moles/liter or 3.0 x 10^{-2} lbs/gal. Expressed as aluminum, the solubility at these conditions is 1.05 x 10^{-2} lbs/gal.

The solubility of the aluminum corrosion products in the post-accident environment is a function of both solution pH and temperature. Figure 6D-10 presents plots of the corrosion product solubility, expressed in terms of aluminum, versus solution pH for temperatures of 77^{o} F and 150^{o} F. The change in solubility with temperature is found utilizing the relationship of the free energy of formation, temperature, and solubility product.

With the data available from Figure 6D-9 and Figure 6D-10 and with 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 3, there are 4.47 x 10^{5} gallons of ECC solution after actuation of the safety features. The total amount of aluminum present in the Containment is given in Table 6D-3. Table 6D-5 shows the corrosion of aluminum with time for the original (NaOH additive) design basis, pH 9.3, post-accident environment. The pH of the sump solution does not exceed 7.4 with sodium tetraborate used for solution pH control based on Reference 25.

Table 6D-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 system at the specific conditions. The last values in the table give the aluminum solubility margin after 100 days 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 ECC solution is not expected to be saturated with aluminum corrosion products.

Further, 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 sodium hydroxide addition and a pH of 8.5 to 10.0 in the solution. Use of sodium tetraborate reduces the long term solution pH to between 7.1 and 7.4. Figures 6D-9 and 6D-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. Corrosion testing at a pH of 8.0 (Reference 23) performed with several buffering agents showed corrosion to submerged aluminum is higher in the presence of sodium Tetraborate compared to sodium hydroxide but was not excessive. This testing rated sodium tetraborate and sodium hydroxide good foe overall resistance to corrosiveness.

It is concluded, therefore, that the corrosion products of aluminum will be in the soluble form during the post-accident period considered and hence, there is no potential for deposition on flow orifices, spray nozzles, or other equipment.

Behavior of Circulating Aluminum Corrosion Products

The solubility of aluminum corrosion products has shown that the entire inventory produced after 100 days exposure to the post DBA condition would remain in solution. The review also indicates that the ECC solution is only approximately 17 percent saturated at 77 F and less than 1 percent saturated at 150 ^{o} F.

It is of interest, however, to review the experience of facilities which have operated with insoluble aluminum corrosion products and to relate their conditions with those expected in the post-accident environment.

The most significant experience available was that of Griess(18) 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 ^{o} C in pH 9.3 sodium borate solution (0.15 M NaOH –0.28 M 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 inch orifice). The pH of the solution did not change because of the increase in the corrosion products.

Griess(18), in describing his observations with regards to aluminum corrosion product deposition potential, stated that:

a) No significant deposition was observed on that cooling coil installed in the solution
b) No significant deposition was observed on the heated surfaces of the facility
c) 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 ^{o} C and 100 ^{o} C were approximately 4.0 and 18.6 grams, respectively. The concentration of aluminum present in the recirculation stream, therefore, was approximately 0.2 and 1 gram/liter, respectively. This value is about a factor of about 5 above the aluminum concentration expected in the post-accident ECC solution at Indian Point 3 in a pH 9.3 (NaOH additive) solution after 100 days. Corrosion testing performed with several buffering agents showed corrosion to submerged aluminum is higher in the presence of sodium tetraborate compared to NaOH but was not excessive.

Hatcher and Rae [19] describe the appearance of turbidity in the NRU reactor and "propose" 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 years of full power operation on several occasions, and within the limit of accuracy of the measurements, reported at approximately 5%, no change in the thermal resistance had been observed.

It is concluded, therefore, from the work of both Griess and of Hatcher and Rae that the deposition of aluminum corrosion products on heat exchangers surfaces will not be significant in the post-accident environments even for the circumstances of insoluble product formation.

6.0 COMPATIBILITY OF PROTECTIVE COATINGS WITH POST-ACCIDENT ENVIRONMENT

The investigation of materials compatibility in the post-accident design basis environment also included an evaluation of protective coating for use inside containment.

The results of the protective coatings evaluation presented in WCAP-7198(12) showed that several inorganic zincs, modified phenolics, and epoxy coatings are resistant to an environment of high temperature (320^{o} F maximum test temperature) and alkaline sodium borate. Long-term tests included exposure to spray solution at 150 to 175 F for 60 days, after initially being subjected to the conservative DBA cycle shown in Figure 6D-3. The protective coatings which were found to be resistant to the test conditions, that is, which exhibited no significant loss of adhesion to the substrate for formation of deterioration products, comprise virtually all of the protective coatings recommended for use in containment. Hence, the protective coatings will not add deleterious products to the core cooling solution.

An additional evaluation was performed (Reference 22) which evaluated the contribution of zinc corrosion to the amount of hydrogen in the post-LOCA containment atmosphere. Based on simple modeling and engineering judgment, the evaluation shows that hydrogen recombiners will be in service well before any observable contribution can be made by zinc corrosion.

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 the substrate.

7.0 EVALUATION OF THE COMPATIBILITY OF CONCRETE-ECC 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 DBA environment.

The purpose of this study was to establish:

a) The extent of debris formation by solution attack of the concrete surfaces
b) The extent and rate of boron removal from the ECC 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 DBA short term, high temperature transient.

For these tests, the total surface area of concrete in the design containment which may be exposed to the ECC solution following a DBA was estimated at 6.3 x 10^4 square feet. This value includes both coated and uncoated surfaces. The ECC solution volume for a reference plant was considered at approximately 313,000 gallons and the surface to volume ratio from these values is approximately 29 in^2/gallon. The surface to volume ratios for the concrete-boron tests used were between 28 and 78 in^2/gallon of solution.

Table 6D-7 presents a summary of the data obtained from the concrete-boron test series.

Testing of uncoated concrete specimens in the post-accident environment showed that attack by both boric acid and the alkaline boric acid 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[13] who lists the following resistances of Portland Cement concrete to attack by various compounds:

Boric acid	- little or no attack
Alkali hydroxide solution under 10%	- little or no attack
Sodium borate	- mild attack
Sodium hydroxide over 10%	- 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 square inches per gallon 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 6D-11 shows a representation of the boron loss from the ECC solution versus time by a boron-concrete reaction following a DBA. The time period from 0 to 6 hours shows the loss during a conservative high temperature transient test, ambient to 320° F to 285° F. The data from 6 hours to 30 days is based 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.

8.0 MISCELLANEOUS MATERIALS OF CONSTRUCTION

8.1 Sealants

Candidate sealant materials for use in the Reactor Containment System were evaluated in simulated DBA environments. Cured samples of various sealants were exposed in alkaline sodium borate solution (pH 10.0, 3000 ppm boron) to a maximum temperature of 320° F.

Table 6D-8 presents a summary of the sealant materials tested together with a description of the panels’ appearance after testing. Three generic types 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 and allowed to cure well prior to testing.

The test results showed that the silicone sealants tested were chemically resistant to the DBA 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 ECC solution during the post DBA period and will maintain its structural integrity and elastic properties.

8.2 <u>PVC Protective Coating</u>

Tests were conducted to determine the stability of the polyvinylchloride protective coating, the type which might be used on conduit in the DBA environment. Samples of the PVC exposed to alkaline sodium borate solutions at DBA conditions showed no loss in structural rigidity and no change in weight or appearance.

A sample of PVC coated aluminum conduit (1" OD x 8" length) was irradiated by means of a Co-60 source a 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 indicted that some bond breakage had occurred in the PVC coating as evidenced by an 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 PVC surface. Considering a total surface area of PVC coating present in containment (approximately 500 ft^2) and the ECC solution volume of 313,000 gallons, the chloride concentration increase in the ECC solution due to irradiation of the coating would be approximately 0.01 ppm.

It was concluded, therefore, that PVC protective coating will be stable in the DBA environment.

8.3 <u>Fan Cooler Materials</u>

Samples of the following air handling system materials were exposed in an autoclave facility to the DBA temperature – pressure cycle:

a) Moisture separator pad
b) High efficiency particulate filter media
c) Asbestos separator pads
d) Adhesive for joining separator pads and HEPA filter media corners
e) Neoprene gasketing material

The materials were exposed in both the steam phase and liquid phase of a solution of sodium tetraborate (15 ppm boron) to simulate the concentrations expected downstream of the fan cooler cooling coils. Examination of the specimens after exposure showed the following:

a) 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

b) High efficiency particulate filter media maintained its structural integrity in both the liquid and steam phase. No apparent change.

c) Asbestos separator pads showed some slight color bleaching, however, both steam and liquid phase samples maintained their structural integrity with no significant loss in rigidity

d) Adhesive material for the HEPA/separator pad edges showed no deterioration or embrittlement and maintained its adhesive property

e) 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 in plant applications.

9.0 ENVIRONMENTAL REQUALIFICATIONS

An ongoing program of evaluating the environmental qualifications of safety related electrical equipment at Indian Point 3 has been in progress since early 1980.

Included in this program are evaluations of the following environmental parameters: function, service, location, operating time, temperature, pressure, relative humidity, chemical spray, radiation, aging, submergence, and qualifying method. This evaluation program is based on the provisions of: Code of Federal Regulations, Title 10, Part 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants" and Regulatory Guide 1.89, Revision 1, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants." Complete and auditable records are available and will be maintained at a central location. These records described the environmental qualification methods used for all safety related electrical equipment in sufficient detail to document compliance with the requirements of 10 CFR 50.49 and Regulatory Guide 1.89, Rev. 1.
Such records will be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified. In accordance with schedule requirements of 10 CFR 50.49, all components falling within the scope of this program will be qualified, replaced, or modified to ensure their operation.

The change in pH buffer from sodium hydroxide to sodium tetraborate was evaluated by Reference 28 for impact on existing equipment qualifications for IP3 EQ components. The report determined that EQ components that were qualified for a sodium hydroxide buffered boric acid solution remained qualified for a sodium tetraborate buffered boric acid solution.

References

1) Bell, M. J., J. E. Bulkowski, and L. F. Picone, "Investigation of Chemical Additives for Reactor Containment Sprays," Westinghouse Proprietary WCAP-7153, March 1968.

2) "ORNL Nuclear Safety Research & Development Program Bimonthly Report for July – August 1968," ORNL TM-2368, p. 78.

3) "ONRL Nuclear Safety Research & Development Program Bimonthly Report for September – October 1968," ORNL TM-2425, p. 53.

4) Swandby, R. K., Chemical Engineer 69, 186, November 12, 1962.

5) Hoar, T. P., and J. G. Hines, "Stress Corrosion Cracking of Austenitic Stainless Steel in Aqueous Chloride Solutions," Stress Corrosion Cracking and Embrittlement (ed. W. D. Robertons), John Wiley and Sons, 1956.

6) Latanision, R. M., and R. W. Staehle, "Stress Corrosion Cracking of Iron – Nickel Chromium Alloys," Department of Metallurgical Engineering, The Ohio State University.

7) Warren, D., Proceedings of Fifteenth Annual Industrial Work Conference, Purdue University, May 1960.

8) Edeleanu, C., JISI 173 (1963), 140.

9) Thomas, K. C., et al., "Stress Corrosion of Type 304 Stainless Steel in Chloride Environment," Corrosion, Volume 20 (1964), p. 89t.

10) Sharfstein, L. R., and W. F. Brindley, "Chloride Stress Corrosion Cracking of Austenitic Stainless Steel – Effect of Temperature and pH," Corrosion, Volume 14 (1958), p. 588t.

11) "ORNL Nuclear Safety Research & Development Program Bimonthly Report for March – April, 1969," ORNL TM-2588.

12) Picone, L. F., "Evaluation of Protective Coatings for Use in Reactor Containment," Westinghouse Proprietary WCAP-7198L, April 1968.

13) Orchard, D. F., "Concrete Technology, Volume 1," Contractors Record Limited, London, 1958.

14) Van Horn, K. C., "Aluminum, Volume I," American Society of Metals (1967).

15) Sundararajan, J. and T. C. Rama Char, Corrosion 17 (1961), 39t.

16) Cotton, F. A., and G. Wilkinson, Advanced Inorganic Chemistry, Interscience Publishers, 1962.

17) Deltombe, E. and Purbaix, M., Corrosion 14 (1968), 49t.

18) Griess, J. C., et al., "Corrosion Studies," ORNL Nuclear Safety Research and Development Program Bi-Monthly, July-August 1968. USAEC Report ORNL-TM-2368, pp. 76-81.

19) Hatcher, S. R. and Rae H. K., Nuclear Science and Engineering 10 (1961), p. 316.

20) SECL-91-313, "Safety Evaluation for Indian Point 3 RWST Boron Increase," Westinghouse Electric Corp.

21) IP3 Technical Specification Amendment 119, June 2, 1992.