September 21, 1973

SAFETY EVALUATION REPORT

BY THE

DIRECTORATE OF LICENSING

U.S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

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ABBREVIATIONS

a-c	alternating current
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
AEC	United States Atomic Energy Commission
AFP	auxiliary feed pump
AISC	American Institute of Steel Construction
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BIT	boron injection tank
$BTU/hr\text{-}ft^2$	British Thermal Units per hour per square foot
CA	wake factor
CARCFS	containment air recirculation cooling and filtration system
cc	cubic centimeter
cfs	cubic feet per second
CFR	Code of Federal Regulations
CSS	containment spray system
DBA	design basis accident
d-c	direct current
Δk, $\Delta \rho$	reactivity change
Δt	temperature change or difference

DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
ECCS	emergency core cooling system
ESF	engineered safety features
°F	degrees Fahrenheit
FSAR	final safety analysis report
ft	feet
ft^2	square feet
ft^3	cubic feet
fps	feet per second
fq	total peaking factor
fz	axial peaking factor
g	gravitational acceleration, 32.2 feet per second per second
GDC	AEC General Design Criteria for Nuclear Power Plants
gpm	gallons per minute
HEPA	high efficiency particulate air
IEEE	Institute of Electrical and Electronics Engineers
in	inch
km	kilometer
kV	kilovolt
kW	kilowatt
kW/ft	kilowatts per foot
lb	pound
LOCA	loss-of-coolant accident

LPZ	low population zone
m	meter
m^2	square meters
mph	miles per hour
m/s	meters per second
MLLW	mean low low water
MSL	mean sea level
MWe	megawatts electrical
MWt	megawatts thermal
mrem	one thousandth of a rad equivalent man
NaOH	sodium hydroxide
NDT	nil ductility transition
NOAA	National Oceanic and Atmospheric Administration
NPSH	net positive suction head
NSSS	nuclear steam supply system
nvt	neutron fluence, neutrons per square centimeter
PAB	primary auxiliary building
PLOCAP	post loss-of-coolant accident protection
PMF	probable maximum flood
PMH	probable maximum hurricane
ppm	parts per million
PSAR	preliminary safety analysis report
psf	pounds per square foot
psi	pounds per square inch

psia	pounds per square inch absolute
psig	pounds per square inch gauge
PWR	pressurized water reactor
QA	quality assurance
QC	quality control
RCPB	reactor coolant pressure boundary
R&D	research and development
rem	rad equivalent man
RWST	refueling water storage tank
sec/m^3	seconds per cubic meter
SI	safety injection
SSE	safe shutdown earthquake
U-235	uranium 235
UO_2	uranium dioxide
USAS	United States of America Standard
USGS	United States Geological Survey
v/o	volume percent
w/o	weight percent
X/Q	relative concentration

1.0 INTRODUCTION

1.1 General Background

This is the Atomic Energy Commission's (Commission) Safety Evaluation Report relating to the application of the Consolidated Edison Company of New York, Inc. (the applicant or Con-Ed) for a license to operate the Indian Point Nuclear Generating Unit No. 3 (Indian Point 3).

The applicant by application dated April 26, 1967, and as subsequently amended, requested a license to construct and operate a pressurized water reactor, to be known as Indian Point Nuclear Generating Unit No. 3 to be located in the town of Buchanan, New York about 24 miles north of New York City. The Commission reported the results of its construction permit review in its Safety Evaluation Report dated February 20, 1969. Following a public hearing before an Atomic Safety and Licensing Board in Montrose, New York on May 15 1969, the Director of Reactor Licensing issued provisional construction permit number CPPR-62 on August 13, 1969.

On December 4, 1970, the applicant filed, as Amendment No. 13, the Final Facility Description and Safety Analysis Report* required by Section 50.34(b) of Chapter 10 of the Code of Federal Regulations (10 CFR) as a prerequisite to obtaining an operating license.

The current application requests an operating license of 3025 megawatts thermal (MWt), equivalent to a net electrical output of

*Throughout this safety evaluation, the Final Facility Description and and Safety Analysis Report is referred to as an FSAR as in the Final Safety Analysis Report.

about 965 megawatts. This is the same power level requested in the initial application.

The radiological safety review with respect to a decision concerning issuance of an operating license for Indian Point 3 has been based on the applicant's Final Safety Analysis Report (Amendment 13) and subsequent Supplements one through 21 inclusive, all of which are available at the Atomic Energy Commission's Public Document Room at 1717 H Street, N. W., Washington, D. C. and at the Hendrick Hudson Free Library, 31 Albany Post Road, Montrose, New York. In the course of the review of the material submitted, numerous meetings were held with the applicant, the nuclear steam system supplier, Westinghouse Electric Corporation, and the applicant's architect-engineer, United Engineers and Constructors, to discuss the plant design, construction, and the proposed operation. As a consequence, additional information was requested which the applicant provided in certain of the supplements. A chronology of the principal actions relating to the processing of the application is attached as Appendix A to this Safety Evaluation Report.

This Safety Evaluation Report summarizes the results of the radiological safety review of Indian Point 3 performed by the Commission's Regulatory staff.

This Safety Evaluation Report also delineates the scope of the technical details considered in evaluating the radiological safety aspects of the proposed operation of the Indian Point 3 facility. In accordance with the provisions of 10 CFR Part 50, Appendix D of the Commission's regulations, a Draft and a Final Environmental Statement will set forth the considerations of the environmental impact of the proposed operation of the Indian Point 3 facility.

1.2 General Plant Description

The Indian Point 3 facility utilizes a nuclear steam supply system incorporating a pressurized water reactor and a four-loop reactor coolant system. The reactor core is composed of fuel rods made of slightly enriched uranium dioxide pellets enclosed in Zircaloy tubes with welded end plugs that are grouped and supported into assemblies. The mechanical control rods consist of clusters of stainless steel-clad silver-indium-cadmium alloy absorber rods that are inserted into Zircaloy guide tubes located within the fuel assemblies. The core fuel is loaded in three regions, each utilizing fuel of a different enrichment of U-235, with new fuel being introduced into the outer region, moved inward at successive refuelings, and removed from the inner region to spent fuel storage. Water will serve as both the moderator and the coolant, and will be circulated through the reactor vessel and core by four vertical, single stage, centrifugal pumps, one located in the cold leg of each loop.

The reactor and reactor coolant system will operate at a pressure of 2250 psia with a nominal reactor inlet temperature of 543°F and a nominal outlet temperature of 600.4°F. The reactor coolant water will be circulated through the four steam generators to produce saturated steam and then be returned back to the pumps to repeat the cycle. An electrically heated pressurizer connected to the hot leg piping of one of the loops will establish and maintain the reactor coolant pressure and provide a surge chamber and a water reserve to accommodate reactor coolant volume change during operation. The steam that is generated in the steam generators will be utilized to drive a four element tandem compound turbine and will be condensed in a radial flow single pass deaerating condenser. Cooling water drawn from the Hudson River will be pumped through the tubes of the condenser to remove the heat from, and thus condense, the steam after it has passed through the turbine. The condensate will then be pumped back to the steam generator to be heated for another cycle.

The reactor will be controlled by a coordinated combination of a soluble neutron absorber (boric acid) and mechanical control rods whose drive shafts penetrate the top head of the reactor vessel. The control system will allow the plant to accept step load changes of 10 percent and ramp load changes of 5 percent per minute over the range of 15 to 100 percent of full power under normal operating conditions. Plant protection systems that automatically initiate appropriate action whenever a monitored condition approaches pre-established limits are provided. These protection systems will act to shut down the reactor, close isolation valves, and initiate operation of the engineered safety features should any or all of these actions be required. Supervision and control of both the nuclear steam supply system and the steam and power conversion system will be accomplished from the control room.

The nuclear steam supply system is housed in a steel lined reinforced concrete cylindrical structure. The control building, the spent fuel pit, and the primary auxiliary building are all Category I* structures. The safety injection pumps, the containment spray pumps, the spray additive tank and boric acid tanks are among the equipment housed in the primary auxiliary building.

The plant is capable of being supplied with electrical power from two independent 138 kV feeders and two 13.8 kV underground feeders and is provided with independent and redundant onsite emergency power supplies capable of supplying power to shut down the plant safely or to operate the engineered safety features in the event of an accident and a loss of offsite power sources.

1.3 Comparison with Similar Facility Designs

Many features of the design of Indian Point 3 are similar to those we have evaluated and approved previously for other nuclear power plants now under construction or in operation. To the extent feasible and appropriate, we have made use of our previous evaluations of those features that were shown to be substantially the same as those previously considered. Where this has been done, the appropriate sections of this report identify the other facilities involved. Our

*Category I structures are defined in Section 3.2 of this Safety Evaluation Report.

Safety Evaluation Reports for these other facilities have been published and are available for public inspection at the Atomic Energy Commission's Public Document Room at 1717 H Street, N. W., Washington, D. C.

1.4 Identification of Agents and Contractors

The Westinghouse Electric Corporation (Westinghouse) is furnishing the nuclear steam supply system for Indian Point 3, including the first fuel loading, and is also furnishing the turbine generator set. For those items of the plant included within its scope of supply, Westinghouse has also acted as procurement agent. Westinghouse had contracted with United Engineers and Constructors as the architect engineer. Construction of the plant was performed by United Engineers until December 1969, when this function was assumed by WEDCO, a wholly owned subsidiary of Westinghouse.

Quirk, Lawler, and Matusky was the applicant's principal consultant for hydrological studies while Environmental Analysts, Inc. prepared population estimates for the applicant. The Research Division of New York University was the applicant's meteorological consultant.

1.5 Summary of Principal Review Matters

The evaluation performed by the staff included a review of the information submitted by the applicant particularly with regard to the following matters:

We evaluated the population density and use characteristics of the site environs, and the physical characteristics of the site, including seismology, meteorology, geology and hydrology to establish that these characteristics had been determined adequately and had been given appropriate consideration in the final design of the plant, and that the site characteristics are in accordance with the Commission's siting criteria (10 CFR Part 100), taking into consideration the design of the facility, including the engineered safety features provided.

We evaluated the design, fabrication, construction, and testing and performance characteristics of the plant structures, systems, and components important to safety to determine that they are in accord with the Commission's General Design Criteria, Quality Assurance Criteria, Regulatory Guides, and other appropriate rules, codes and standards, and that any departures from these criteria, codes, and standards have been identified and justified.

We evaluated the expected response of the facility to various anticipated operating transients and to a broad spectrum of accidents, and determined that the potential consequences of a few highly unlikely postulated accidents (design basis accidents) would exceed those of all other accidents considered. Conservative analyses were performed of these design basis accidents to determine that the calculated potential offsite doses that might result in the very unlikely event of their occurrence would not exceed the Commission's guidelines for site acceptability given in 10 CFR Part 100.

We evaluated the applicant's engineering and construction organizations, plans for the conduct of plant operations, including the proposed organization, staffing and training program, the plans for industrial security, and the plans for emergency actions to be taken in the unlikely event of an accident that might affect the general public, to determine that the applicant is technically qualified to safely operate the plant.

We evaluated the design of the systems provided for control of the radiological effluents from the plant to determine that these systems are capable of controlling the release of radioactive wastes from the facility within the limits of the Commission's regulations, and that the equipment provided is capable of being operated by the applicant in such a manner as to reduce radioactive releases to levels that are as low as practicable.

We evaluated the financial position of the applicant to determine that the applicant is financially qualified to operate Indian Point 3.

1.6 Facility Modifications Required as a Consequence of Regulatory Staff Review

As a consequence of the staff review, a number of design changes and emergency plan changes were made to Indian Point 3. These modifications are discussed in greater detail within the body of this Safety Evaluation Report. The principal changes which were made are as follows:

(1) The seismic instrumentation program has been augmented (see Section 3).

(2) The safety injection system has been redesigned to meet the single failure criterion (see Section 7).

(3) Interlocks have been placed on the residual heat removal system to prevent possible over-pressurization of this system (see Section 7).

(4) Design modifications to prevent premature initiation of the recirculation phase following a postulated loss-of-coolant accident (see Section 7).

(5) Design modifications to eliminate the need for automatic transfers between redundant d-c power sources (see Section 8).

(6) Modifications to the emergency diesel fuel oil transfer pumps (see Section 8).

(7) Provision of additional gaseous and particulate monitors to the radwaste area, the control room, and the fuel handling and storage area (see Section 12).

(8) Expanded emergency plans to include letters of agreement from the Coast Guard, medical support facilities, and the Penn Central Railroad (see Section 13).

(9) A more rapid method of alerting appropriate officials has been developed in the case of a radiological emergency (see Section 13).

(10) Numerous design changes have been required for protection against postulated high energy line breaks outside of the containment (see Sections 6 and 10).

2.0 SITE CHARACTERISTICS

2.1 Geography and Demography

The Indian Point facility is situated on a 239-acre tract of land located in Westchester County, New York on the east bank of the Hudson River. The three-unit nuclear facility is located approximately 2-1/2 miles southwest of Peekskill, New York and 24 miles north of the New York City boundary line.

The Indian Point nuclear facility is surrounded on all sides by high ground ranging in elevation from 600 to 1000 feet above sea level. Across the Hudson River, which varies in width between 4500 and 5000 feet in the vicinity of the plant site, the west bank is' flanked by steep heavily wooded slopes of the Dunderberg and West Mountains to the northwest (elevations 1086 feet and 1257 feet, respectively) and the Buckberg Mountains to the west-southwest (elevation 793 feet).

The closest cities with populations exceeding 25,000 are Newburgh, New York (1970 population of 26,219, a decrease of 15% since 1960), and White Plains, New York (1970 population of 50,220 a 0.5% decrease since 1960), both located approximately 17 miles from the Indian Point site. The area within five miles of the site has a population of 18,130 based on the 1970 census data. The projected population for the year 2010 is approximately 74,000 persons. The closest schools are located about one mile to the south and east of the site. Figures

2.2 and 2.3 show the 1970 population and predicted cumulative population data for the year 2010 relevant to the Indian Point nuclear facility.

At the present time the land surrounding the Indian Point site is residential with large areas devoted to parklands and a military reservation. A gypsum plant is adjacent to the southwest border of the Indian Point site. Northeast of the site, just within the 1100 meter low population zone radius, is a second industrial area on the shoreline of Lent's Cove. The closest commercial airport is at White Plains, New York, 17 miles south of the station. Minor seaplane activity occurs at Green's Cove, about 1.5 miles south of the plant.

The Hudson River in the area of the site is used for commercial ship and barge traffic and for pleasure boating. For recreation, there are several sections of the Palisades Interstate Park on the west bank, and fishermen's landings, parks and beaches are on the east bank of the Hudson River.

The Hudson River is not used for drinking water purposes below the plant site due to salt water intrusion in the tidal estuary. The nearest municipal intake of the Hudson River is that for the City of New York, which is located about 22 miles upriver from the Indian Point site, in the vicinity of Chelsea, New York. Four industrial plants within five miles of the site use water from the Hudson River for industrial purposes and one plant uses a well for its source of industrial water.

In a report prepared by Environmental Analysts, Inc. in June 1972, the population data submitted by the applicant was updated based on the 1970 census, and population projections were made through the year 2010. Based on this report there have been no significant demographic changes in the area of the site as described in our February 20, 1969 Safety Evaluation Report.

The minimum exclusion distance as provided by the applicant for Indian Point 3 is 350 meters from the centerline of the reactor and 330 meters from the outer surface of the containment building to the nearest property line (Figure 2.1). The outer edges of the cities of Newburgh and White Plains, New York, are the nearest boundaries of densely populated geographic centers containing more than 25,000 persons, and both are located approximately 17 miles from the plant site. However, based on projected populations, the outer boundary of the more densely populated section of the City of Peekskill was chosen by the applicant during the construction permit stage of review as the population center. The nearest boundary of Peekskill is 0.63 mile to the northeast; however, the nearest residential area of Peekskill is 0.85 mile to the east. The applicant has selected a low population zone having an outer boundary of 0.67 mile (1100 meters). On the basis that (1) the population within the proposed low population zone is small (approximately 50 people) and (2) the area of Peekskill in the area of the nuclear plant is of a general industrial nature, the staff at the time of the construction permit review concurred in the applicant's selection of the low population zone.

Based on the 10 CFR Part 100 definitions of the population center distance, the exclusion area and low population zone distances (for which adequate emergency plans have been developed), on our analysis of the onsite meteorological data from which dilution factors were calculated for various time periods (Section 2.3), and on the calculated potential radiological dose consequences of design basis accidents (Section 15.0), we have concluded that the exclusion area radius is acceptable from the standpoint of computed doses from all of the design basis accidents analyzed when the reactor is operated at the proposed power level of 3025 MWt.

2.2 Nearby Industrial, Transportion and Military Facilities

New York State Route 9, which passes through Peekskill and Buchanan, is located on the east bank of the Hudson River and Route 9W and the Palisades Interstate Parkway on the west bank of the Hudson River. A Penn Central railroad line passes within 0.85 mile of the Indian Point 3 containment structure on the east bank of the Hudson River; on the west bank, a line of the Penn Central Railroad passes approximately one mile from the Indian Point site. Two natural gas lines cross the Hudson River and pass about 620 feet from the Indian Point 3 containment structure. Based on previous staff reviews, failures of these gas lines will not impair the safe operation of Indian Point 3.

About 600 to 800 commercial barges and ships on the Hudson River pass by the Indian Point site each year. The cargoes consist of petroleum products, dry goods, and molasses. The applicant has indicated that no river traffic shipment of toxic materials or explosives currently pass the site. No new environmental hazards have been identified since the construction permit review of this plant.

The staff has reviewed the question of airport proximity to nuclear power plants in various other licensing cases. On the basis of these studies, we conclude that the Indian Point site is sufficiently far from an airport of significant size that the probability of a crash at the site is essentially that associated with general overflights and that the Indian Point facility need not be designed or operated with special provisions to protect the facility against the effects of an aircraft crash.

The military installations in the area include the New York State Military Reservation (Camp Smith) and the West Point Military Reservation. Camp Smith is about two miles and West Point is about six miles from Indian Point 3.

The closest industry to the Indian Point site is the Georgia Pacific gypsum plant located approximately 0.3 mile southwest of the Indian Point 3 containment structure. Oil, gas, gasoline, and molasses storage facilities are located just outside of the 1100 meter low population zone for this facility.

Because of the distance that separates these military and industrial facilities and because of experience gained in the operation of Indian Point 1 at the same site, we have concluded that these facilities will not affect the safe operation of Indian Point 3.

2.3 Meteorology

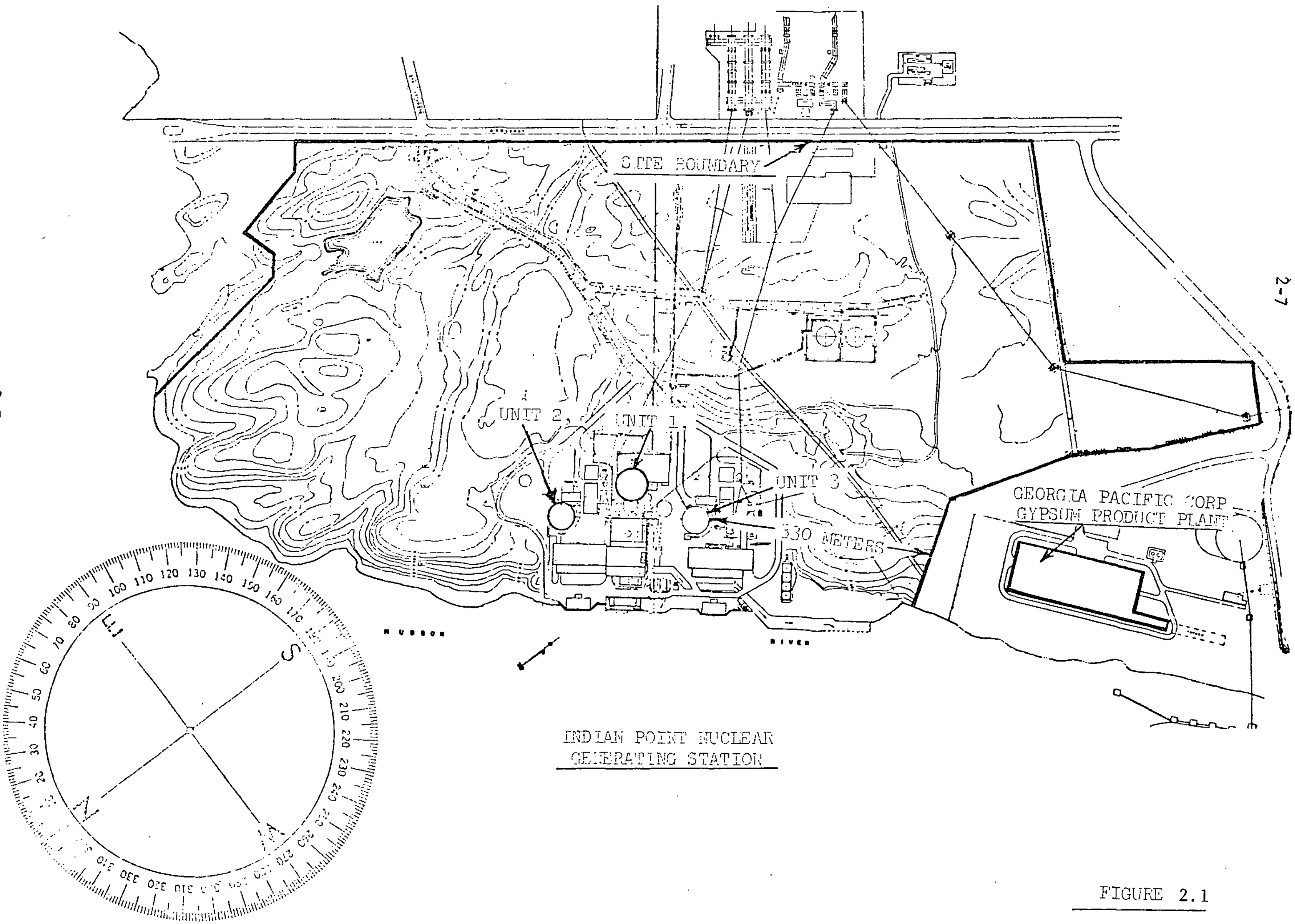
The plant is in a general climatic region which is primarily continental in character, but is subjected to some modification by marine air which can penetrate the site area. The general regional topography ranges from hilly to mountainous. Locally, the plant site lies along the Hudson River in a bowl surrounded on almost all sides by high ground ranging from 600 to 1000 feet above sea level. This topography decisively influences meteorological conditions in the valley in the following ways:

a. Orientation of the valley ridges channels the airflow.

b. The wind speeds in the valley tend to be lower than in open level terrain.

c. Differential heating of the hillsides and the plain at the mouth of the valley create local air circulation (e.g., diurnally-regulated up-and-down valley flow).

The measured prevailing winds show the influence of valley channelling. This channelling effect is not as pronounced during the winter months due to generally stronger westerly airflow aloft.



INDIAN POINT NUCLEAR
GENERATING STATION

FIGURE 2.1

2-7

CUMULATIVE POPULATION VS. DISTANCE 0-5 MILES

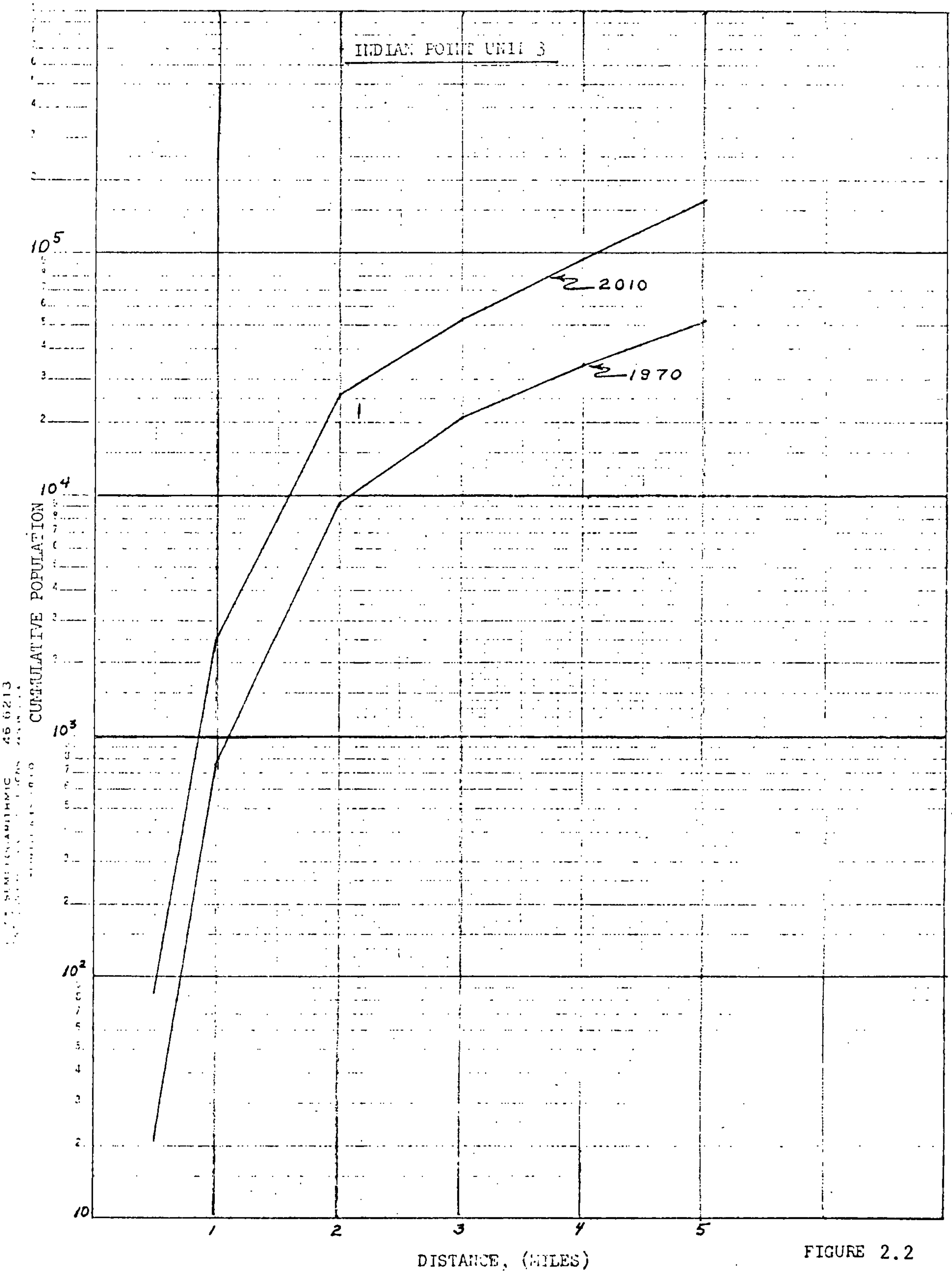


FIGURE 2.2

CUMULATIVE POPULATION VS. DISTANCE 0-50 MILES

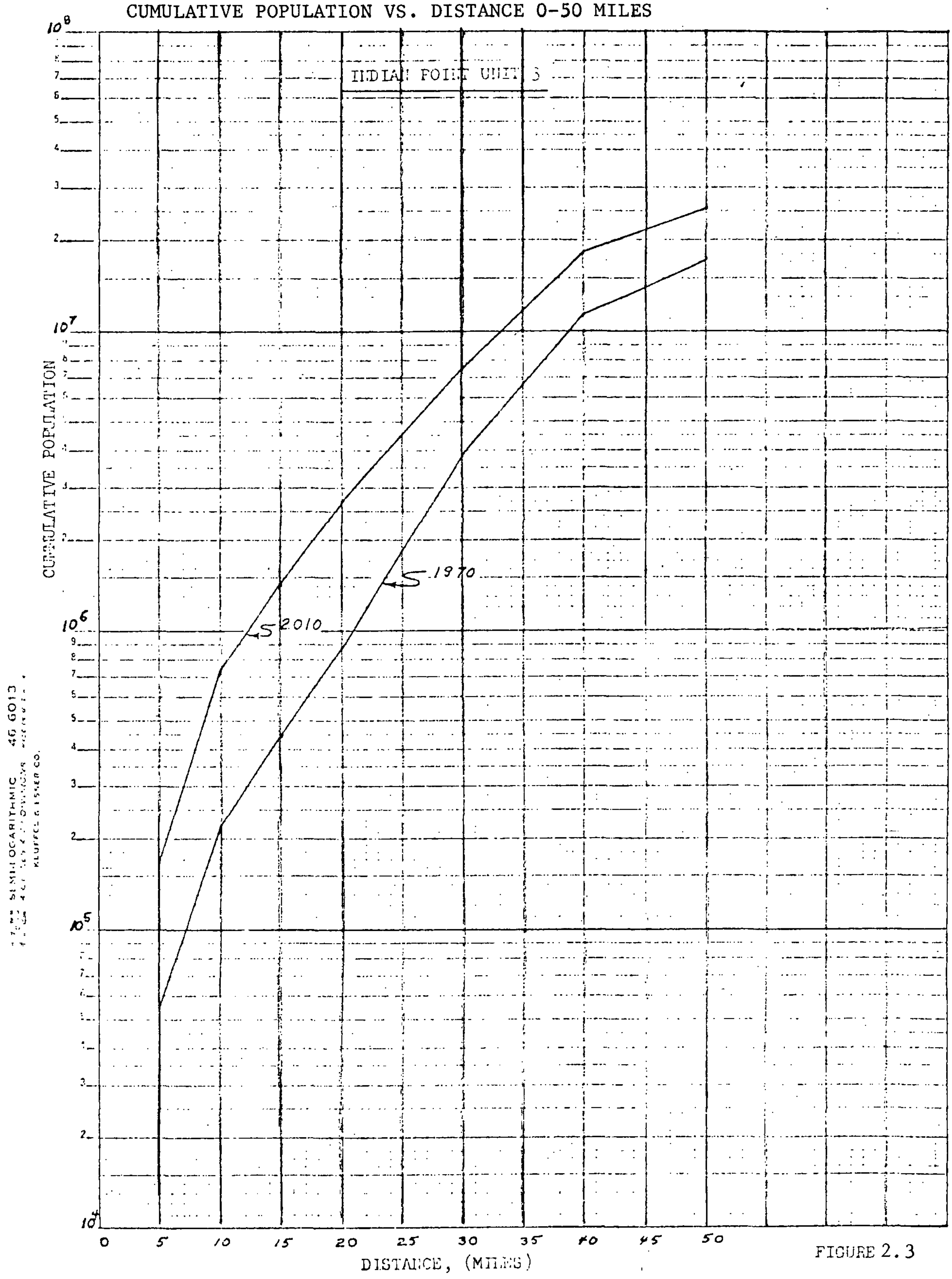


FIGURE 2.3

Several meteorological studies of atmospheric diffusion conditions have been based on onsite data. The initial onsite meteorological measurement program was conducted during the years 1956 and 1957. The program consisted of measurements of wind speed and temperature taken on a 300-foot tower. Data on the joint frequency distribution of wind direction and speed were taken at the 100-foot level and vertical temperature differences between the 150- and 7-foot levels were measured. These data were presented in exhibit L-5 for Indian Point 1 (Docket No. 50-3). The total joint frequency data recovery for this period is not now known because the data were presented as fractions of recovered data and the original records were not kept.

Another meteorological study was conducted during the years 1969 and 1970. This study was conducted primarily to describe the diurnal wind direction reversals in the Hudson River valley. Measurements of wind and temperature were made on a 100-foot tower in the same location as the now dismantled 300-foot tower and at other stations along the Hudson River located within five miles of the site. Data collected in this program were taken for the period November 26, 1969 - October 1, 1970, with recovery rate near 80%. Joint frequency distributions of wind direction and speed by vertical temperature difference class were not presented in this study. However, the applicant concluded that the annual average statistics of wind direction and speed, and of vertical temperature difference were substantially the same as the 1956-1957 data, thereby indicating that meteorological conditions are reasonably consistent from year to year. Diurnal valley flow wind reversals were found whenever the winds aloft were very light.

More recent data were acquired by the applicant during the years 1970 - 1972 utilizing the 100-foot tower. These data provided the basis for making a meteorological analysis of the site in accordance with current staff practices and verified the representativenes of of the 1956-57 data.

It is the Regulatory staff's practice to utilize for offsite dose calculations meteorological data that have been collected for at least one continuous year with a data recovery rate of at least 90%. Due to numerous equipment failures, the applicant's meteorological data recovery rate was often below 90% during the 1970-1972 years.

Consequently, for use in its accident analyses, a composite year of data was constructed by the staff where the recovery rate was 95%. This composite year consisted of January-July 1970, August 1971, September-October 1972, and November-December 1970.

Additional modifications of the applicant's data were made to have the data conform with present staff methods. The applicant recorded wind speeds and directions at the 100-foot elevation, while temperatures were measured at the 95-foot and 7-foot levels. The wind speed measurements were adjusted by the staff to represent wind speeds at a level of 33 feet (the height assumed for ground level release calculations) by use of a power law extrapolation. The temperature difference between the readings at the 95-foot and 7-foot levels were extrapolated to temperature differences simulating recording instruments at 150-foot and 30-foot levels. This new vertical temperature difference calculated by the staff utilized a logarithmic method to extrapolate the measured vertical temperature difference.

Additional data were submitted by the applicant in support of other meteorological models. In Supplement 14 of the FSAR, the applicant presented an analysis of diffusion conditions using the "split sigma model" to allow for greater wind meander, a procedure to allow for diffusion to the distance of the actual site boundary by direction instead of the minimum site boundary, a procedure to allow for the effect of averaging diffusion conditions over a two-hour period, and a turbulent building wake diffusion model developed from New York University wind tunnel tests. The application of any one of these four meteorological models would result in significant reductions in the calculated off-site doses. Although the staff felt that these meteorological models have some merit, they were not accepted at this time. Among the reasons for not accepting these proposed meteorological models was the concern that the instruments that recorded the basic data were not sufficiently accurate in the wind speed range of interest. Additional studies using more accurate instruments and conditions simulating winds below 4 mph may be acceptable to the staff at some future date.

We concluded that the applicant did not provide sufficient justification for the use of these meteorological models for evaluating the radiological consequences of accidents at this site; consequently we used our own, more conservative meteorological models.

Utilizing standard staff practices, an evaluation of the meteorological diffusion characteristics of the site was made for both accident analysis and routine release analysis purposes.

The evaluation of the calculated offsite doses resulting from radioactive releases due to postulated accidents requires calculations of the relative concentration, X/Q, for the first 30 days following an assumed accident. The impact of routine radioactive releases requires calculations of an annually averaged X/Q. These relative concentrations were then incorporated into dose analyses.

Accident dose analyses utilize calculated X/Q values which vary with time. The staff uses its most conservative assumptions when calculating the X/Q values for the first eight hours following an assumed accident. Additional credit is given for diffusion and spread of the gaseous plume for time periods beyond the first eight hours.

The calculated dose at the minimum exclusion radius (330 meters) at the end of the first two hours and the 30 day dose at the low population zone (1100 meters) must be within 10 CFR Part 100 limits.

In the evaluation of the diffusion of short term (0-2 hr) accidental releases from the plant, a ground level release was assumed with a building wake factor, cA, of 1000 square meters. The relative concentration, X/Q, using the composite year of data (1970-1972), which is exceeded 5% of the time was calculated to be 1.8×10^{-3} sec/m^3 at the minimum exclusion radius of 330 meters. This relative concentration is equivalent to a dispersion condition produced by Pasquill type F stability and a wind speed of 0.7 meters/second with the building wake effect being limited to a factor of three over the diffusion condition produced by a point source. A similar analysis of the 1956-57 data tends to confirm these results. Our meteorological consultant, the National Oceanic and Atmospheric Administration (NOAA), has calculated a similar X/Q value and the applicant estimates a value which is 40% lower (less conservative) than ours.

In addition to calculating the X/Q values utilized in the two-hour dose at the exclusion radius, the staff calculated X/Q values for the 30-day dose at the outer boundary of the low population zone (LPZ). Using the diffusion models presented in Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors", and the onsite meteorological data, the staff calculated X/Q values at the LPZ of 4.7×10^{-4} sec/m^3 for the 0-8 hour period, 1.4×10^{-4} sec/m^3 for the 8-24 hour period, 6.5×10^{-5} sec/m^3 for the 1-4 day period and 2.2×10^{-5} sec/m^3 for the 4-30 day period. The applicant has presented values which are in essential agreement with the staff's values for the first

24 hours but are a factor of two to three less for the time period from one to 30 days.

In our evaluation of the diffusion conditions associated with routine effluent release, the maximum annual average relative concentration, 2.6×10^{-5} sec/m^3, was calculated to the south-southwest of the plant at the site boundary (330 meters). Both the applicant and our meteorological consultant (NOAA) have presented values which are in essential agreement with ours.

As discussed in Section 11.0 of this Safety Evaluation Report, concerning effluent releases, the maximum annual average concentration at a location seven miles south of the plant is 2.4×10^{-7} sec/m^3.

We have concluded that the composite year of data presented in the FSAR provides a reasonable basis for estimating atmospheric diffusion conditions during accidental and routine gaseous effluent releases from the plant. It is not expected that subsequent data collection and analysis will change our estimates significantly because the data from the years 1956, 1957, and 1969 confirm the climatic representativeness of the data for the composite year.

2.4 Hydrology

The plant is located on the east bank of the Hudson River and is affected by ocean tides as modified by estuary effects between the site and the ocean. Runoff from precipitation-type floods, storm effects along the coastline, or a combination of these types of events can cause local high water levels. Such situations are common along estuaries such as the Hudson River. Similarly, low water levels are affected by tides, runoff, and cyclonic type storms such as hurricanes which can depress water levels by essentially blowing water downstream. Normal maximum tidal flows at the site, in both the upstream and downstream directions, vary between 250,000 and 300,000 cubic feet per second (cfs). Plant grade is about elevation 15.3 feet above mean sea level datum (MSL). The intake structure at elevation 15.0 ft. MSL is of the outdoor type with the safety-related service water pumps located landward of the circulating water pumps. Other safety-related facilities are more landward of the intake structure.

The Hudson River is used for water supply in the area, but only for industrial cooling purposes near the site. The river is used for public water supply some 30 miles upstream at Poughkeepsie, and may be used as a supplemental New York City source at Chelsea (22 miles upstream of the plant) during drought conditions. Ground water in the area is generally used for industrial and commercial purposes, with some limited residential usage on the west side of the river at Stony Point.

Floods from both runoff and hurricane-induced mechanisms have occurred in the area. The highest historical water level in the plant vicinity occurred in 1950 when a water level of 7.4 feet MSL was recorded about one-half mile downstream of the site.

The potential for site flooding from precipitation events, hurricanes, upstream dam failures, and from combinations thereof has been investigated by the applicant and evaluated by the staff. Water levels at the site for a probable maximum flood (PMF), a probable maximum hurricane (PMH) surge, coincident precipitation-type floods and hurricanes, and dam failures during various floods have been estimated. A PMF or a PMH is considered the upper limit of potential flooding that can reasonably be expected to occur at this particular site. The applicant's analyses of flooding events indicate that the worst such event reasonably possible would be the coincident occurrence of the runoff from a precipitation-type flood approximately half as severe as a PMF timed to occur with the worst conditions produced by a surge from a hurricane approximately half as severe as a PMH, and an arbitrarily assumed failure of a large upstream dam. This estimate of the water level at the site for such an event is elevation 15.0 feet MSL, and includes an allowance of 1.0 foot for coincident wave action. The individual occurrence of either a PMF or a PMH, however, were each estimated by the applicant to produce water levels at the site of 14.0 feet MSL and 14.5 feet MSL, respectively. Each estimate contains an allowance of 1.0 foot for coincident wind-generated wave action. Based upon a comparison of the applicant's estimates with similar determinations at other locations in the Northeastern U.S. and upon a review of the applicant's computational techniques, we concur with the applicant in the estimates of water levels at the site for these events, exclusive of the allowances provided for coincident wind-generated wave action.

We have independently estimated wind-generated wave action coincident with either a PMF, a PMH, or other reasonably extreme combinations of less severe events. Our analysis was based upon an analytical technique developed by the Corps of Engineers using a postulated 45 mile per hour overwater wind speed as the cause of such wave action. To assure that the plant will be safely shut down before wave action could cause a loss of function of service water pumps, the Technical Specifications require plant shutdown for water levels approaching 15 feet MSL and appropriate emergency procedures to protect service water pumps.

At the request of the staff, the applicant has analyzed the capability of local site drainage, including the roofs of safety-related structures, to store and/or pass the runoff from precipitation events as severe as could be produced by a local probable maximum storm. Although such facilities would undoubtedly overflow during severe rainstorms, the analysis has indicated that no loss of safety-related functions from such an event is anticipated.

The applicant has arbitrarily assumed the failure of upstream dams coincident with floods of a severity approximately half that of a PMF and has determined that water levels at the site would be somewhat less severe than would be produced by a PMF or PMH as discussed above. On the basis of our review, we agree.

The water levels which could be produced from tsunamis at the site, are considered to be substantially less than those which can be produced from a PMF, PMH, dam failures, or reasonable combinations of such flood producing mechanisms.

Ice-induced flooding at the site to levels approaching those estimated for a PMF, or PMH, dam failures, or reasonable combinations of such flood producing mechanisms, are not considered credible because of the adjacent extremely wide and relatively deep river channel.

The complete loss-of-cooling water at the Indian Point site is not considered credible because water can reach the site from both upstream and downstream sources.

The potential exists for minor flooding in the vicinity of the outdoor intake structure that could be produced by wave action, coincident with severe river flood levels. The staff has required the applicant to provide for such extreme conditions by instituting plant shutdown for water levels approaching elevation 15 feet MSL and to protect the service water pumps in such situations.

Low water levels are influenced by both droughts and tides. Extreme low water levels are caused primarily by severe wind storms, such as hurricanes, where storm winds tend to blow estuary water downstream. The safety-related effects of low water levels at the site are related to the ability of the service water pumps to provide a continuous water supply. The applicant has shown in the FSAR that the service water pumps, located in the outdoor intake structure at elevation 15.0 MSL, reach to elevation 18.5 feet below MSL. Mean low water, based on published U. S. Coast and Geodetic Survey records, is approximately elevation 1.5 feet below MSL. The Coastal Engineering Research Center's records of the 1932 and 1959 historical low water levels at New York City have been extrapolated to the site by use of U. S. Coast and Geodetic Survey tide difference inference techniques to indicate that low water levels at the site of approximately elevation 5.5 feet below MSL have been experienced. The 13-foot difference between the apparent historical low water level and the service water intake is considered by the staff to provide adequate assurance of a dependable safety-related water supply.

Ground water occurs in both unconsolidated surficial deposits, and the fractures and solution channels in the underlying bedrock. The surficial deposits range in thickness from a few feet in hills to several hundred feet in the valley flood plains. General surficial deposit ground water movement at riverbank locations is considered to be toward the river, except where high well pumping rates are employed (which are not expected at the site), or during relatively short periods of high river levels. Use of ground water within five miles of the site has been reported by the applicant in several categories; public water supply and commercial, industrial, and private use. The only public water supply use within five miles is at Stony Point across the river where water is drawn from shallow wells at a rate of about 550 gallons per minute. Most of the rest of the local wells take water from the deeper bedrock aquifer. Within two to three miles of Indian Point, almost all wells have been abandoned and connections have been made to public water supply systems. We have reviewed the potential for contamination of ground water sources from the plant and have concluded that such contamination is highly unlikely because of the direction of ground water movement and the very limited use of ground water in the plant vicinity.

2.5 Geology, Seismology, and Foundation Engineering

The staff has completed its review of the geology, seismology, and foundation engineering matters relating to Indian Point 3 and has concluded that the site foundation conditions are acceptable for the facility. This conclusion is based on reports from our consultants, the U. S. Geological Survey (USGS) and the National Oceanic and Atmospheric Administration (NOAA), formerly the U. S. Coast and Geodetic Survey. These reports are included as Appendices D and E, respectively, in the Safety Evaluation Report issued on February 20, 1969.

The U.S. Geological Survey stated that "There are no known active faults or other young geologic structures in the area that could be expected to localize earthquakes in the immediate vicinity of the site. Although several ancient faults occur in the area, none appears to have been tectonically active since glacial times, or for at least the past several hundred thousand years."

Likewise, in its evaluation of the seismological aspects of the site, the U. S. Coast and Geodetic Survey (now NOAA) stated that "based on the review of the seismic history of the site and the related geologic considerations, the Coast and Geodetic Survey believes that the applicant's proposal to use 0.10g for representing earthquake disturbances likely to occur within the lifetime of the facility to be adequate. The [Coast and Geodetic] Survey agrees with the applicant that 0.15g would provide adequate basis for designing protection against loss of function of components important to safety."

With regard to the ground which supports the structures at Indian Point 3, the applicant's geological consultant concluded that there were no cavernous conditions within the bedrock at the site. It based this conclusion on detailed studies of two nearby quarries and borings drilled at the site. The applicant reported that when excavations were made for Units 1, 2, and 3 cavernous conditions were not encountered. This conclusion is supported by a representative from the U.S. Geological Survey who visited the site and orally reported that there were no cavernous conditions.

The staff has also reviewed various aspects of the foundations for Units 1, 2 and 3. The applicant stated that "The Unit No. 1 structures are now at least 12 years old and there has never been any evidence of settlement cracking or other settlement related problems. The same can be said for recently completed Unit No. 2 structures." No evidence of settlement related problems has been observed with the Indian Point 3 structures during their construction.

Based on the performance of these foundations and the earlier reports of the USGS and NOAA, the foundation conditions at Indian Point are acceptable.

3.0 DESIGN CRITERIA - STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.1 Conformance with AEC General Design Criteria

The Indian Point 3 plant was designed and is being constructed on the basis of the proposed General Design Criteria, published July 11, 1967. Construction of the plant was partially completed and the Final Facility Description and Safety Analysis Report had been filed (filed on December 4, 1970) when the Commission published its revised General Design Criteria in February 1971 and the present version of the criteria in July 1971. As a result, we did not require the applicant to reanalyze the plant on the basis of the revised criteria. However, our technical review assessed the plant against the General Design Criteria now in effect and we have concluded that the plant design conforms to the intent of these newer criteria.

3.2 Classification of Structures, Components and Systems

The applicant has classified the seismic design of plant structures, components and systems into three principal categories. Class I* includes those structures, components and systems whose failure might cause or increase the severity of a loss-of-coolant accident, or result in an uncontrolled release of significant amounts of radioactivity, and those structures, components and systems essential to safe shutdown and isolation of the reactor. Class II includes those structures, components and systems that are important to reactor operation, but not essential

* In this Safety Evaluation Report, the staff utilizes the words Category I as equivalent to the applicant's seismic Class I notation.

to safe reactor shutdown and whose failure would not result in the release of significant amounts of radioactivity. Class III includes those structures, systems and components that are not directly related to reactor operation or containment. In addition, some structures have a mixed classification when they are basically a Class II or Class III designation, but contain components or systems of a Class I or Class II designation.

We find these classifications to be acceptable and we have concluded that the applicant placed all safety-related structures, systems, and components in their appropriate category.

3.3 Wind and Tornado Criteria

The applicant has considered the effects of tornado loads in the design of Category I structures. Tornado wind loading was taken as a 300 mph tangential wind traveling with a translational velocity of 60 mph. Also considered as a separate and combined loading condition is a 3 psi pressure drop external to the structure.

The wind loading and pressure drop parameters are consistent with the generally accepted criteria used for nuclear power plants. ASCE Paper No. 3269 was utilized to determine the loads resulting from these wind and tornado effects. We have concluded that in the design of the facility, the methods of converting wind and tornado velocities into forces on the structures are in accordance with the state-of-the-art and are acceptable.

3.4 Water Level (Flood) Design Criteria

The applicant established that the most severe flooding condition corresponds to a water elevation of 15 feet above mean sea level (MSL). This elevation is lower by three inches than the critical elevation at which water would start seeping into the lowest of the plant buildings.

As discussed in Section 2.4 of this report, the staff concluded that flooding levels under the most extreme conditions could reach a level of 15.0 feet MSL, exclusive of wind-generated wave action. Wind-generated wave action could raise the flooding level above plant grade in the vicinity of the service water pumps.

In the event of wind-generated wave action in conjunction with extreme flooding conditions, the plant will still be protected. In this unlikely event, the plant will be shut down in accordance with the Technical Specifications, and the service water pump area will be protected. Other areas, such as the diesel generator area, will not require additional protection from the wind-generated waves in that these waves rapidly dissipate once they strike land. Calculations have shown that the plant intake structure can bear a hydraulic load associated with 21 feet of water. Therefore, we have concluded that the intake structure can withstand the additional hydraulic load produced by wind-generated waves.

Consequently, the combination of the elevation of the plant structures, the load-bearing capability of the intake structure, and the Technical Specification requirements on plant operation and service water pump protection, result in acceptable conditions to protect the plant against flooding.

3.5 Missile Protection Criteria

Various structures at the Indian Point 3 site have been designed and constructed to withstand the effects of tornado generated missiles. Among these structures are the primary auxiliary building, the control room, the containment, the diesel generator building, the cable tunnels, and the waste hold-up tank pit.

The tornado generated missiles include a spectrum of possible items that could be dislodged during tornadic winds and become missiles. The missiles assumed by the applicant include two horizontal missiles: a four inch by twelve inch by twelve foot wooden plank traveling end-on at 300 mph and an automobile weighing two tons with a contact area of 20 ft^2 traveling not more than 25 feet off the ground at 50 mph, and two vertical missiles: a four inch by twelve inch by twelve foot wooden plank at 90 mph and a passenger car weighing two tons at 17 mph less than 25 feet above the ground. We find that the missile criteria utilized by the applicant are adequate on the basis that they have been used on previous plants.

The effects of missiles generated inside of the containment have also been considered. The reactor coolant system is protected by a three foot thick concrete shield wall which encloses the reactor coolant loop and the pressurizer. A two foot thick concrete operating floor provides additional protection against internally generated missiles. The effects of missiles generated by the fracture of control rod drive mechanisms have also been considered. A structure over the control rod drive mechanisms has been provided to block any such potential missiles. We have concluded that the measures taken to provide protection against internally generated missiles are acceptable.

3.6 Protection Against Dynamic Effects Associated with the Loss-of-Coolant Accident

The applicant has provided protection against pipe whip in accordance with the criteria in Regulatory Guide 1.46 "Protection Against Pipe Whip Inside Containment". The piping/support systems have been dynamically analyzed by the time-history method for each postulated break.

We conclude that the applicant has provided adequate pipe whip restraints to protect against postulated breaks, both longitudinal and circumferential at specified locations within the reactor coolant pressure boundary and in the main steam and feedwater systems.

3.7 Seismic Design

We and our consultant, Nathan M. Newmark Consulting Engineering Services, have reviewed and evaluated the seismic design input criteria employed by the applicant with reference to structures, systems and components. The seismic loads are based on horizontal ground accelerations of 0.10 g for the Operating Basis Earthquake and 0.15 g for the safe shutdown earthquake with vertical accelerations equal to two-thirds the horizontal ground accelerations. The consultant's report is attached as Appendix B.

The seismic design response spectra curves were presented in the application for a construction permit and found acceptable. The modified earthquake time histories used for component equipment design were adjusted in amplitude and frequency to envelope the response spectra specified for the site. We and our seismic design consultant conclude that the seismic input criteria proposed by the applicant provides an acceptable basis for seismic design.

The modal response multi-degree-of-freedom method and the normal mode-time history method are used for the analysis of all Category I structures, systems and components. The vibratory motions and the associated mathematical models account for the soil-structure interaction and the coupling of all coupled Category I structures and plant equipment. Governing response parameters have been combined by the square root of the sum of the squares to obtain the modal maximums when the modal response spectrum method is used. The absolute sum of responses is used for closely spaced frequencies. Horizontal and vertical floor spectra inputs used for design and test verification of structures, systems and components were generated by the normal mode-time history method. Torsional loads have been adequately accounted for in the seismic analysis of the Category I structures. Vertical ground accelerations were assumed to be 2/3 of the horizontal ground accelerations and the horizontal and vertical effects were combined simultaneously. Constant vertical load factors were employed only where analysis showed sufficient vertical amplifications in the seismic system being analyzed.

We and our consultant have reviewed the information provided by the applicant and find the seismic system and subsystem dynamic analysis methods and procedures used by the applicant acceptable.

As part of the review of Indian Point 3, the staff concluded that the applicant's seismic instrumentation program required augmentation. In response to the staff's requirement for additional seismic instrumentation the applicant has added three peak shock recorders on the containment base mat in a tri-axial-arrangement. The applicant also added recording accelerographs on one steam generator, one reactor coolant pump, and on the pressurizer. A plan for the utilization of any acquired seismic data will be developed before an operating license is issued.

We conclude that the type, number, location and utilization of strong motion accelerographs to record seismic events and to provide data on the frequency, amplitude and phase relationship of the seismic response of the containment structure correspond to the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes" and is acceptable.

3.8 Design of Category I Structures

The review and evaluation of the Category I structures included the structural foundations, the containment, the auxiliary building, the control room, the intake structure, and a portion of the pump-house. The structures were built from a composite of structural steel and reinforced concrete members. In general, the structures were designed as continuous systems. The various structural components that were integrated into the continuous structures consist of slabs, walls, beams, and columns.

The analyses were based on elastic analysis procedures with the design being executed using the working stress design method and the ultimate strength design method. The design method for reinforced concrete followed that of ACI 318-63, with the use of specific loading combinations applicable to nuclear power plant design conditions. For the structural steel the AISC Specifications were utilized.

The loading combinations used for the design of the structures included normal dead and live loads, accident loads, wind and tornado loads, the flood loads, the missile loads and the earthquake loads.

The applicant has specified and utilized numerous loading combinations for the normal loading conditions as well as for the severe loading conditions that include the accident, the tornado and/or the safe shutdown earthquake.

As a result of the review and evaluation of the criteria and the procedures related to the design and construction, we conclude that the Category I structures have been acceptably designed and constructed.

3.9 Mechanical Systems and Components

3.9.1 Dynamic System Analysis and Testing

The staff has reviewed the effects of dynamic loads on the Indian Point 3 reactor coolant pressure boundary and on the reactor internal structures.

Because of the similarities of the Indian Point 3 design to Indian Point 2, the applicant has designated Indian Point 2 as the prototype plant from which preoperational vibration test results are applicable in evaluating the design adequacy of the reactor internal structures of the Indian Point 3 plant. A vibrational test program was conducted at Indian Point 2 which included various operational flow transients up to a temperature and pressure of 530°F and 2200 psi, respectively. The response characteristics of vibratory strain, acceleration, and pressure signals were analyzed in terms of major frequency components for obtaining modal contributions and to define the dynamic behavior under flow induced excitations. These test results were reported in Topical Report WCAP-7879, "Four Loop PWR Internals Assurance and Test Program." The staff has reviewed this topical report and has concluded that the vibration testing program conducted at the

Indian Point 2 plant acceptably demonstrates the integrity of the Indian Point 2 reactor internals to withstand flow-induced vibrations under normal operating conditions. The staff also concluded that these tests on Indian Point 2 serve as a prototype for other four loop Westinghouse plants, similar in design to Indian Point 2, including Indian Point 3. Thus, only a confirmatory preoperational vibration test in accordance with Regulatory Guide 1.20 will be conducted on Indian Point 3. On the basis of the applicability of the Indian Point 2 tests and conformance to Regulatory Guide 1.20, we have concluded that the vibration test program is acceptable.

The reactor internal structures must also withstand the loadings associated with the simultaneous occurrence of a LOCA and a safe shutdown earthquake (SSE). The applicant has applied the results of Topical Report WCAP-7822 "Indian Point Unit No. 2, Reactor Internals Mechanical Analysis for Blowdown Excitation" to demonstrate the capabilities of the Indian Point 3 reactor internals. The staff has reviewed this topical report including the mathematical models, analytical procedures and methods, allowable stress criteria, and allowable deflection and stability criteria. The staff concludes that the topical report is acceptable and applicable to Indian Point 3. The staff also concludes that the Indian Point 3 reactor internals will withstand the simultaneous occurrence of LOCA and SSE loadings within design limits.

The consequences of vibration in other parts of the reactor coolant pressure boundary (RCPB) have been considered by the staff. In accordance with the provisions of USAS B31.1.0, a vibration operational test program will be performed during startup and initial operations. This test program will verify that the piping and piping restraints within the RCPB have been designed to withstand the dynamic effects of valve closures, pump trips and other anticipated events which could cause significant vibrations. These tests will simulate transients that are expected to be experienced during reactor operation and will demonstrate that the requirements of USAS B31.1.0 to minimize vibrations have been met.

3.9.2 Category I Components Outside of the Reactor Coolant Pressure Boundary

All safety-related systems, components, and equipment outside of the reactor coolant pressure boundary are Category I and are designed to sustain normal loads, anticipated transients and one half of the safe shutdown earthquake (SSE) within the appropriate code allowable stress limits. These same systems, components, and equipment are also designed to sustain the SSE within stress limits which are comparable to those associated with the emergency operating condition category of current component codes. We have concluded that these stress criteria provide an acceptable margin of safety for Category I systems and components outside of the RCPB which may be subjected to seismic loadings.

3.10 Seismic Qualification of Category I Instrumentation and Electrical Equipment

A seismic qualification program for all Category I instrumentation and electrical equipment was implemented to confirm that (1) in the event of a safe shutdown earthquake, this equipment will function properly during the earthquake and following the post-accident operation, and (2) the support structures for this equipment are adequately designed to withstand the seismic disturbance. The operability of the instrumentation and electrical equipment was ensured by testing. The design adequacy of the supports was assured by either analysis or testing. The results of these tests and analyses are described in Topical Report WCAP 7397-L "Seismic Testing of Electrical and Control Equipment". We have evaluated this report and conclude that it is acceptable and applicable to Indian Point 3.

Additional information on this subject is given in Section 7.8 of this Safety Evaluation Report.

4.0 REACTOR

4.1 Summary Description

The Indian Point 3 core is similar to that of the Zion Unit 1 core and consists of 193 fuel assemblies with 204 fuel rods per assembly. The active heat transfer surface area for each plant is approximately 52,000 ft.^{2} The proposed initial power level for the Indian Point 3 core is 3025 megawatts thermal (MWt) as compared to 3250 MWt for the Zion Unit 1 core. A comparison of the Indian Point 3 and the Zion Unit 1 thermal hydraulic, and core mechanical and nuclear design parameters is given in Table 4.4.

4.2 Mechanical Design of Reactor Vessel Internals

For normal design loads of mechanical, hydraulic and thermal origin, including anticipated plant transients and the operational basis earthquake, the reactor internals were designed to the stress limit criteria of Article 4 of the ASME Boiler and Pressure Vessel Code Section III, 1965 Edition.

The reactor internal components have been designed to withstand the loads calculated to result from a loss-of-coolant accident (LOCA), the safe shutdown earthquake (SSE) and the combination of these postulated events, utilizing the criteria described in Section 14.3.3 of the FSAR, and in Topical Report WCAP-7822, "Indian Point Unit No. 2 Reactor Internals Mechanical Analysis for Blowdown Excitation". These criteria are consistent with the comparable Code emergency and faulted operating condition category limits and the criteria which have been accepted for all recently licensed plants. Accordingly, we have concluded that these design criteria are acceptable.

4.3 Nuclear Design

The nuclear design of the Indian Point 3 reactor is essentially the same as that for Indian Point 2 and Zion Units 1 and 2 previously reviewed by the staff and found acceptable. The design power level and average linear power density for Indian Point 3 is intermediate to Indian Point 2 and Zion as shown in Table 4.3 below. Our conclusions concerning the adequacy of the nuclear design presented in the Safety Evaluations for the above cited four-loop Westinghouse designed reactors apply to Indian Point 3 in most areas. These include design bases, reactivity control provisions, reactivity coefficients, nuclear design methods, and the general concept of reliance on ex-core neutron detectors for power level and power distribution monitoring and safety system functions.

TABLE 4.3

Design Power Level and Average Linear Power Density

Plant	Power Level (MWt)	Average Linear Power Density (kW/ft)
Indian Point 2	2760	5.7
Indian Point 3	3025	6.2
Zion 1 and 2	3250	6.7

We have examined the effects of fuel densification on the core power monitoring requirements and have concluded that Indian Point 3's system of reliance on ex-core neutron detectors is acceptable. This conclusion is based on the fact that the largest total peaking factor, F_Q, expected during operation of this nuclear power plant is smaller than the peaking factor (and its associated power level) which meets emergency core cooling acceptance criteria. This comparison is discussed below.

Fuel densification has been observed in some Westinghouse manufactured fuel. Densified fuel can result in local power spikes, greater stored energy in the fuel, and a reduced heat transfer capability within the fuel. (See Section 6.5 of this Safety Evaluation for a more complete discussion). The effects of fuel densification on the operation of Indian Point 3 have been calculated. It was determined that the core can be operated with densified fuel at a power level of 3025 MWt, and an F_Q, of 2.56, and will meet the AEC's emergency core cooling acceptance criteria.

In a separate series of calculations, the total peaking factor, F_Q, with densified fuel, was calculated for numerous operating conditions. To be conservative, this group of calculated F_Q's was calculated for many extreme control rod configurations not expected under usual operating conditions. These calculated values of F_Q have been correlated with the percent of axial offset. Axial offset is defined as the percent of the difference between the power levels of the top and bottom halves of the core, divided by their sum. The correlation shows that $F_Q \leq 2.56$ when the axial offset is in the range of -14 to +12 percent. The axial offset of the Indian Point 3 core is determined from measurements made by the ex-core detectors. To account for ex-core detector uncertainty the Technical Specifications limit the measured axial offsets from -11 to +9 percent when the core is at the design power level.

On the basis that the largest F_Q value that satisfies emergency core cooling limits equals or exceeds any F_Q value allowed by the Technical Specifications, the use of ex-core monitors satisfies the core power monitoring requirements.

The reactor is protected against uncontrolled axial xenon oscillations. It is predicted to be stable against azimuthal oscillations. This will be verified by tests in Indian Point 2 and/or Zion 1.

We conclude that these measures and those described in the FSAR assure that F_Q limits will be maintained and allow safe operation of the reactor at the design power level of 3025 MWt.

In addition to the provisions required for power maldistribution detection, control, and protection, a limited number of fixed in-core neutron detectors have been included in the Indian Point 3 design. Such detectors have also been included in the Indian Point 2 and H. B. Robinson reactors. The fixed in-core detector system consists of eight flux thimbles located symmetrically (radially and axially) throughout the core. Each thimble will have four miniature detectors (small argon filled, highly enriched U^{235} fission chambers) with a sensitive length of about one inch and will be about 0.15 inch in diameter. Individual detectors have a design limit of about 3×10^{21} nvt.

The detectors will provide input to a computer. The readings for each detector will be time averaged for one minute, and the computer will compute the following:

a. Mean power level seen by each detector string.

b. Axial offset seen by each string.

c. Core mean power level.

d. Core mean axial offset.

e. Radial quadrant tilt factors for the eight quadrants which describe the tilted power distribution curve for each detector string.

The computer will print out an alarm message whenever:

a. Any of the 8 mean power levels exceeds its limit.

b. Any of the 8 radial tilting factors exceeds its limit.

c. Any of the 8 axial offsets exceeds its limit.

d. The core mean axial offset exceeds its limit.

There is no safety requirement for these detectors, but their existence and use will provide extra assurance that power distribution limits are maintained.

4.4 Thermal and Hydraulic Design

The core thermal and hydraulic design parameters for Indian Point 3 and Zion Unit 1 are presented in Table 4.4 to facilitate comparison of these two reactors. The design criteria for prevention of fuel damage are the same for both reactors. The first criterion is that the minimum local DNBR, calculated using the Westinghouse W-3 correlation*, be maintained greater than 1.3 for steady state and anticipated transient conditions. The second criterion is that fuel melting will not occur for all steady state and anticipated transients.

We have reviewed the methods of analyses and the results of core thermal hydraulic performance for a spectrum of limiting anticipated transients presented in the FSAR. These include Loss of Coolant Flow (FSAR Section 14.1.6), Loss of External Electrical Load (FSAR Section 14.1.8) and Excessive Load Increase (FSAR Section 14.1.11). For all of these anticipated transients, the minimum DNBR during the transients is well above 1.3 using appropriate assumptions regarding initial power distribution. Additional analyses of core performance during transients have been presented in WCAP-7306, "Reactor Protection System Diversity in Westinghouse Pressurized Water Reactors" which is applicable to Indian Point 3.

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*The Westinghouse W-3 correlation is used to predict the heat flux and location where departure from nucleate boiling is predicted to occur.

On the basis of the applicant's ability to calculate power distribution, ability to verify these calculations experimentally with incore instrumentation, the adequacy of the W-3 correlation, the results of analyses for both the steady state and transient cases of interest, and a suitable margin between the minimum calculated DNBR and 1.3, we have concluded that the reactor thermal and hydraulic design is acceptable.

TABLE 4.4

REACTOR DESIGN COMPARISON*

THERMAL AND HYDRAULIC DESIGN PARAMETERS	Indian Point 3	Zion Station
Performance Characteristics		
Reactor Core Heat Output, MWt	3025	3250
Reactor Core Heat Output, Btu/hr	10324×10^6	11090×10^6
System Pressure, PSIA	2250	2250
Minimum DNBR at Nominal Conditions	2.21	2.02
Coolant Flow		
Total Flow Rate, lb/hr	136.3×10^6	135.0×10^6
Average Velocity Along Fuel Rods, ft/sec	15.6	15.3
Average Mass Velocity, lb/hr-ft^2	2.54×10^6	2.52×10^6
Coolant Temperature, °F		
Nominal Inlet	542.6	530.2
Average in Core	573.0	564.8
Average in Vessel	571.5	563.2
Nominal Outlet of Hot Channel	633.5	631.7
Heat Transfer at 100% Power		
Active Heat Transfer, Surface Area, ft^2	52,200	52,200
Average Heat Flux, Btu/hr-ft^2	193,000	207,900
Maximum Heat Flux, Btu/hr-ft^2	539,000	579,600
Average Thermal Output, kW/ft	6.2	6.7
Maximum Clad Temperature, °F		
Clad Surface at Nominal Pressure	657	657
Clad Average at Rated Power	715	720
Fuel Central Temperature, °F		
Maximum at 100% Power	4100	4250

CORE MECHANICAL DESIGN PARAMETERS	Indian Point 3	Zion Station
Fuel Assemblies		
Design	RCC Canless 15 x 15	RCC Canless 15 x 15
Number of Fuel Assemblies	193	193
UO_2 Rods per Assembly	204	204

*As originally presented in the FSAR

TABLE 4.4

REACTOR DESIGN COMPARISON (Cont'd)

CORE MECHANICAL DESIGN PARAMETERS	Indian Point 3	Zion Station
Overall Dimensions, in.	8.426 x 8.426	8.426 x 8.426
Number of Grids per Assembly	7	7
Fuel Rods		
Number	39,372	39,372
Outside Diameter, in.	0.422	0.422
Clad Thickness, in.	0.0243	0.0243
Clad Material	Zircaloy-4	Zircaloy-4
Fuel Pellets		
Material	UO_2 Sintered	UO_2 Sintered
Length, in.	0.600	0.600
Fuel Enrichment, w/o U-235		
Region 1	2.25	2.25
Region 2	2.80	2.80
Region 3	3.30	3.30
Rod Cluster Control Assemblies		
Number of Clusters, Full/Part Length	53/8	53/8
Number of Control Rods per Cluster	20	20
NUCLEAR DESIGN PARAMETERS		
Hot Channel Factors		
Heat Flux		
Nuclear, F_Q^N	2.72	2.71
Engineering, F_Q^E	1.03	1.03
Total	2.80	2.79
Enthalpy Rise		
Nuclear, $F_{\Delta H}^N$	1.58	1.58
Engineering, $F_{\Delta H}^E$	1.01	1.01

5.0 REACTOR COOLANT SYSTEM

5.1 Summary Description

The reactor coolant system includes a reactor vessel and four coolant loops connected in parallel to the reactor vessel. Each loop contains a circulating pump and a steam generator. The pressurizer, the pressurizer relief tank connecting piping, and instrumentation necessary for operational control are also part of the reactor coolant system.

5.2 Integrity of Reactor Coolant Pressure Boundary

Components of the reactor coolant pressure boundary are Category I and are built to meet the requirements of the codes and standards specified in 10 CFR 50.55a, except that the pumps are designed to an equivalent acceptable standard. The stress limit criteria specified for the normal and upset operating condition categories of the applicable codes apply for normal loads, anticipated transients and the Operational Basis Earthquake. Under the loads calculated to result from the Design Basis Accident, the safe shutdown earthquake and the combination of these postulated events, the components of the reactor coolant pressure boundary are designed to the applicable emergency and faulted operating condition limits of the appropriate codes, or where explicit limits are not provided in the codes, to the criteria of Appendix A of the FSAR. The criteria of Appendix A, as modified by Supplement 12 of the FSAR, are consistent with comparable current code criteria. We have concluded that these criteria are acceptable for components of the reactor coolant pressure boundary.

Table 5.2 lists the Code requirements to which the reactor coolant system has been designed and fabricated.

To assure compliance with the safety and design criteria, ferritic materials of pressure retaining components of the reactor coolant pressure boundary must exhibit adequate fracture toughness properties under normal reactor operating conditions, system hydrostatic tests, and during transient conditions to which the system may be subjected. We have reviewed materials testing and the operating limitations proposed by the applicant and find them acceptable.

The applicant has stated in the FSAR, Amendment Nos. 23 and 24, Supplement Nos. 9 and 10, respectively, that acceptance testing for ferritic materials was performed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III (1971 Edition, including Addenda through Summer 1972). Dropweight NDT data have been obtained for the reactor vessel material.

In establishing the operating pressure and temperature limitations during heatup, cooldown, and inservice hydrostatic tests of the system, the applicant has followed the recommendations of Appendix G, "Protection Against Non-Ductile Failure," of the 1972 Summer Addenda of the ASME Code, Section III.

We have reviewed the specific heatup, cooldown, and hydrostatic test limitation curves applicable to Indian Point 3 and conclude they meet the current fracture toughness Regulatory staff requirements. These curves form the basis for the heatup and cooldown limits included in the Technical Specifications.

We conclude that the planned operation of the reactor coolant system in conformance with the Technical Specification limits will assure adequate margins of safety.

Stainless steel that has been sensitized has an increased susceptibility to stress corrosion cracking. The applicant has shown in FSAR, Appendix 4D, and Amendment Nos. 21 and 23, Supplement Nos. 7 and 9, respectively, that significant sensitization of all nonstabilized austenitic stainless steel within the reactor coolant pressure boundary was avoided through materials selection and control of welding and heat treating processes. The precautions included: (1) use of approved procedures for welding and verification of them by periodic quality control checks; (2) use of low heat input procedures during shop and field welding operations; (3) check of core structures by the Strauss test; (4) not allowing use of wrought furnace sensitized stainless steel, and (5) limiting interpass temperatures during welding to 350°F maximum. Where stainless steel safe ends were welded to the vessel, the weld preparation of both the safe end and the nozzle were built up with Inconel. We conclude that the steps taken to avoid sensitization of austenitic stainless steel during the fabrication period are acceptable.

Selected welds and weld heat-affected zones must be inspected periodically to assure continued integrity of the reactor coolant pressure boundary during the service lifetime of the plant. The applicant has stated in Amendment No. 21 that the inservice inspection program for the reactor coolant pressure boundary will comply with Section XI of the ASME Boiler and Pressure Vessel Code, "Rules for In-Service Inspection of Reactor Coolant Systems," 1970 Edition. Access for inservice inspection was provided in the design and arrangement of pressure-containing components. Section 4.2 of the Technical Specifications lists the inservice inspection requirements for Indian Point 3.

The facility was constructed to allow either external or internal inspection of the reactor vessel using a remotely operable inspection tool capable of performing inspections of vessel surfaces, and circumferential, longitudinal, and nozzle welds.

We conclude that the access provisions and planning for inservice inspection are acceptable. The provisions of the AEC Guideline, "Inservice Inspection Requirements for Nuclear Power Plants Constructed with Limited Accessibility for Inservice Inspection," (January 31, 1969) have been satisfied.

The applicant has provided, for inservice inspection, access to the Group B and C fluid systems, such as the engineered safety systems, reactor shutdown systems, cooling water systems, and the radioactive waste treatment systems outside the limits of the reactor coolant pressure boundary. The applicant stated in Amendment No. 22 that when ASME Section XI of the Boiler and Pressure Vessel Code is revised to include additional system requirements in the above areas, that these requirements will be evaluated for application to Indian Point 3. We conclude that the planning for an inservice inspection program for the Group B and C fluid systems is acceptable.

5.3 Reactor Vessel Material Surveillance Program

A material surveillance program is required to monitor changes in the fracture toughness properties of the reactor vessel beltline material induced by neutron radiation.

The applicant has shown in the FSAR, Amendment Nos. 21 and 23, Supplement Nos. 7 and 9, that the proposed materials surveillance program, although differing in minor details, is technically equivalent to the requirements of the Commission's Appendix H, 10 CFR Part 50, 50.55(a). The only significant difference is that to obtain the optimum relationship between the integrated neutron flux seen by the vessel wall and the capsules, the capsules will have to be

TABLE 5.2

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 are designed and fabricated are:

Component	Code Edition	Class	Applicable Addenda
Reactor Vessel	1965	A	Summer 1965 and Code Cases 1332, 1335, 1339, 1359
Rod Drive Mechanism	1965	A	Summer 1966
Rod Drive Mechanism (part-length)	1965	A	Summer 1967
Steam Generators - Tube side	1965	A	Summer 1966
- Shell side	1965	A	Summer 1966
Pressurizer	1965	A	Summer 1966
Pressurizer Relief Tank	1965	C	Summer 1966
Pressurizer Safety Valves	1965		Summer 1966
Reactor Coolant Pump Volute	- Designed per ASME III Article 4.		

In addition the reactor coolant pipe was designed to ANSI B31.1 - 1955.

rotated from one location to the other during the service life of the vessel. The program is acceptable with respect to the number of capsules, number and type of specimens, and retention of archive material. The proposed withdrawal and rotation schedule will provide adequate monitoring of radiation effects occurring in the vessel material.

We have concluded that the reactor vessel material surveillance program will adequately provide for monitoring neutron induced changes in the fracture toughness of the reactor vessel material and is acceptable.

5.4 Leakage Detection System

The leakage detection system provided for the reactor coolant pressure boundary includes diverse leak detection methods, has sufficient sensitivity to measure small leaks including such leakage from small through-wall flaws, and has suitable control room alarms and readouts. The major components of the system are the containment atmosphere particulate and gaseous radioactivity monitors, main air recirculation unit condensate coil collection and measurement system, and level indicators on the containment sump. Indirect indication of leakage can be obtained from the containment humidity, pressure and temperature indicators.

We have reviewed the design and sensitivity of the leakage detection systems and have concluded that the systems have the capability to detect leakage from small through-wall flaws in the reactor coolant pressure boundary and are acceptable.

5.5 Pump Flywheel Integrity

The loss of pump flywheel integrity, which could result in high energy missiles and excessive vibration of the reactor coolant pump assembly, has been minimized by the use of a suitable material, adequate design and inspection.

The design, fabrication, and preservice and inservice inspections for the pump flywheels presented in Amendment No. 21 are in general accord with AEC Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity." Therefore, we conclude that the design, fabrication, and inspection of the flywheels are acceptable.

5.6 Evaluation of the Integrity of the Reactor Vessel

During installation of the reactor vessel at the site, a hoist failed, and the vessel was dropped. A reinspection of the vessel was performed, which involved dimensional checks, visual examination, and nondestructive examination by magnetic particle, liquid penetrant, and ultrasonic methods. The results obtained from the nondestructive examinations subsequently served as a basis for assessment of possible damage to the vessel using stress analysis and fracture mechanics criteria.

A report prepared by Oak Ridge National Laboratory entitled, "Summary Report and Reinspection and Appraisal of the Indian Point

Unit No. 3 Reactor Pressure Vessel Subsequent to Hoist Failure on January 12, 1971," covering the above incident and the subsequent reinspection and evaluation was reviewed by the Regulatory staff. Our review of the report revealed that the nondestructive examination techniques which were used were equal to or better than those specified by the ASME Boiler and Pressure Vessel Code, Section III, and in fact permitted a more comprehensive examination than that originally performed using the Code specified methods. No rejectable defects were disclosed as a result of the above indicated inspection, even though additional discontinuities were shown to be present in excess of those originally reported.

Appendix "C" of the ORNL report, which is in two parts, contains an assessment of the effects of this incident based on stress analysis and fracture mechanics. This appendix has been reviewed and evaluated.

The procedure in the first part of this appendix is inappropriate due to assumptions made relating to the stress, the imposed stress intensity, and the toughness. In the second part the toughness value that was used agrees well with an estimated lower bound reference toughness from the ASME Code, Section III, Appendix G, 1972 Summer Addenda. We believe that the calculated maximum bending stress is realistic. A critical flaw depth of approximately four inches was calculated. Our independent calculations, performed according to the procedures of Welding Research Council Bulletin No. 175, PVRC Recommendations on Toughness Requirements for Ferritic Materials, August 1972, confirm the results of this calculation. Further, using conservative assumptions, we have estimated that a four-inch deep flaw, assumed to exist in the most deleterious location and orientation, would have grown less than 0.001 inch due to this incident.

We concur with the findings of the report that no rejectable defects were disclosed, and that any existing flaws would not have been significantly extended as a consequence of this incident. There was no mechanical damage to the reactor vessel and, therefore, its integrity was not impaired by the drop which resulted from the hoist failure. On this basis we conclude that the Indian Point 3 pressure vessel is acceptable for its intended service.

5.7 Loose Parts Monitor

Occasionally, miscellaneous items such as nuts, bolts, etc., have become loose parts within reactor coolant systems. In addition to causing operational inconvenience, such loose parts can damage other components within the system or be an indication of undue wear or vibration. For such reasons, the staff has encouraged applicants over the past several years to support programs designed to develop effective, on-line loose parts monitoring. For the past few years we have required each applicant for an operating license of a PWR plant to initiate a program, or to participate in an ongoing program, the objective of which is the development of a functional, loose parts monitoring system within a reasonable period of time. We will require this applicant to commit to a similar undertaking.

It is of interest to note that prototype loose parts monitoring systems have been developed and are presently in operation or being installed at several plants. None, however, are plants utilizing a Westinghouse nuclear steam supply system. We will be evaluating the experience gained with these systems as it becomes available to us.

6.0 ENGINEERED SAFETY FEATURES

6.1 General

The purpose of the various engineered safety features is to provide a complete and consistent means of assuring that the public will be protected from excessive exposure to radioactive materials should a major accident occur in the plant. In this chapter we discuss the reactor containment system, the emergency core cooling system, the auxiliary feedwater system, fuel densification, and the post loss-of-coolant accident protection system. Certain of these systems have functions for normal plant operations as well as serving as engineered safety features.

Systems and components designated as engineered safety features are designed to be capable of assuring safe shutdown of the reactor under the adverse conditions of the various postulated design basis accidents described in Section 15 of this report. They are designed, therefore, to Category I standards and they must function even with complete loss of offsite power. Components and systems are provided in sufficient redundancy so that a single failure of any component or system will not result in the loss of the capability to achieve safe shutdown of the reactor. The instrumentation systems and emergency power systems are designed for the same seismic and redundancy requirements as the systems they serve. These systems will be described in Sections 7 and 8 of this report, respectively.

6.2 Containment Systems

6.2.1 Containment Functional Design

The Indian Point 3 containment is a steel-lined, reinforced concrete structure with a net free volume of approximately 2,610,000 ft^3. The containment houses the reactor and primary coolant system, including the pressurizer and steam generators, and certain components of other engineered safety features provided for the facility. The containment is designed for an internal pressure of 47 psig and a temperature of 271°F.

We have evaluated the containment system in comparison to the Commission's General Design Criteria stated in Appendix A to 10 CFR Part 50 of the Commission's Regulations and, in particular, to Criteria 16 and 50. As a result of our evaluation, we have concluded that the calculated pressure and temperature conditions resulting from a design basis loss-of-coolant accident will not exceed the design conditions of the containment structure. The highest calculated containment pressure and temperature are about 44 psig and 268°F, respectively, which are calculated for the loss-of-coolant accident resulting from a postulated double-ended rupture of a pump suction pipe in the reactor coolant system.

The applicant has described the results and methods used to analyze the containment pressure response for a number of design basis loss-of-coolant accidents in FSAR Supplement 12. Break locations and sizes were varied to determine that the double-ended pipe rupture at the pump suction of the reactor coolant system results in the highest containment pressure. As discussed below, we have reviewed these analyses, and verified by our own analyses that the methods used by the applicant were acceptably conservative.

The applicant has analyzed the containment pressure response from postulated loss-of-coolant accidents in the following manner. Mass and energy release rates were calculated using the SATAN V, LOCTA and REFLOOD computer codes. TThese mass and energy addition rates were then used as inputs to the COCO computer program, which is used by the applicant to calculate the containment pressure response. The SATAN V computer code was used by the applicant to determine the mass and energy addition rates to the containment during the blowdown phase of the accident; i.e., the phase of the accident during which most of the energy contained in the reactor coolant system, including the primary coolant, metal, and core stored energy, is released to the containment. To obtain a conservatively high energy release rate to the containment during the blowdown phase, the applicant extended the time that the core would remain in nucleate boiling. The LOCTA computer program was used to calculate this energy release. The calculational approach used by the applicant assumes that more energy would be transferred to the containment for containment analyses than for emergency core cooling studies. This additional energy release from the core will increase the containment pressure. Both the SATAN V computer code and the LOCTA computer code have been accepted by the AEC for calculating energy release during a LOCA.

During the core reflood phase of the accident, mass and energy release rates were calculated by the applicant using the computer code REFLOOD. The analyses of the reflood phase of the accident are important with regard to pipe ruptures of the reactor coolant system cold legs, since the steam and entrained liquid carried out of the core for these break locations pass through the steam generators and represent an additional energy source. The steam and entrained water leaving the core and passing through the steam generators will be evaporated and/or superheated to the temperature of the steam generator secondary fluid.

Results of the FLECHT* experiments indicate that the carryout fraction of fluid leaving the core during reflood is about 80% of the incoming flow to the core. The rate of energy release to the containment during this phase is proportional to the flow rate into the core. The rupture of the cold leg at the pump suction results in the highest mass flow through the core, and thus through the steam generators. We have compared the mass and energy release to the containment during the reflood phase of the accident using our FLOOD computer code with that predicted by the applicant using the REFLOOD

*FLECHT - Full Length Emergency Cooling Heat Transfer.

computer code. The results of this comparison indicate equivalent predictions of energy release. Therefore, we have accepted the REFLOOD computer code as a realistic method of computing core reflood for this plant.

We have analyzed the containment pressure response for a double-ended rupture in the suction leg of the reactor coolant system using the CONTEMPT-LT computer code which includes the energy addition to the containment from the steam generators. In our analysis, we assumed the core is quenched at the 10-foot elevation, whereas the applicant assumed that entrainment continued until the quench front reached the 8-foot elevation. Consequently, in our analysis the energy release was greater and the containment pressure slightly higher. We calculated a peak containment pressure of about 44 psig as compared to 40 psig calculated by the applicant using the COCO computer code.

We conclude that the maximum containment pressure is conservatively calculated to be below the design pressure (47 psig) of the containment structure.

The applicant has analyzed the pressure response of the containment interior compartments, such as the reactor vessel cavity and steam generator compartments, to postulated loss-of-coolant accidents. The applicant calculates peak differential pressures of 600 psi in the reactor cavity and 6.4 psi in a steam generator compartment, and has designed these compartments accordingly. The reactor cavity is designed for a pressure of 1000 psi, and the steam generator compartments are designed for a pressure of 7 psi. We have performed similar calculations and our results are in agreement with the applicant's. We, therefore, conclude that the design pressures of the compartments are acceptable.

6.2.2 Containment Heat Removal Systems

The Containment Spray System (CSS) and the Containment Air Recirculation Cooling and Filtration System (CARCFS) are provided to reduce the containment pressure and remove fission products from the containment atmosphere following a loss-of-coolant accident. Any of the following combinations of equipment will provide adequate heat removal capability:

(1) Both spray trains of the CSS.

(2) All five fan-cooler units of the CARCFS.

(3) One spray train of the CSS and three fan-cooler units of the CARCFS.

The CSS, which consists of two separate spray trains of equal capacity, is designed as a Category I system. Missile protection of system components is provided by direct shielding and by physical separation of duplicate equipment. The containment sump screen assemblies, through which the containment spray flows prior to recirculation, are designed to prevent debris from entering the spray system which could clog the spray nozzles.

The CSS includes a system for injecting sodium hydroxide into the spray water to enhance iodine removal from the containment atmosphere if fission products are released from the core following an accident. The sodium hydroxide enters the spray water system through eductors. The motive fluid for the spray additive eductors is the borated water supplied from the discharge of the spray pumps. The spray additive tank contains enough sodium hydroxide to bring the entire post-accident containment water inventory to a pH of 8.3. Provision has been made for monitoring and adjusting the pH of the recirculating cooling water.

A high containment pressure signal will automatically actuate the CSS. The system pumps and valves can also be manually operated from the control room. The spray pumps initially take suction from the refueling water storage tank. When the water in the tank reaches a low level, a switchover from injection to recirculation is manually initiated. During the recirculation phase, spray water is supplied by redundant recirculation pumps located within the containment. These recirculation pumps take suction from the recirculation sump. Environmental qualification tests have been performed on the recirculation pump motors in simulated accident environments more severe than would be expected for postulated loss-of-coolant accidents. Backup recirculation capability is provided by the redundant residual heat removal pumps located outside the containment, which take suction from the containment sump. At the time the recirculation phase is initiated, sufficient water has been delivered to the containment to provide the required net positive suction head to the recirculation pumps. The residual heat exchangers cool the spray water during the recirculation phase.

The Containment Air Recirculation Cooling and Filtration System (CARCFS) is designed to remove heat from the containment to prevent the containment design pressure from being exceeded and remove fission products from the containment atmosphere if they are released following a loss-of-coolant accident. The CARCFS consists of five equal capacity air handling units. All components of the CARCFS, except the filter sections of the air handling units, are part of the Containment Ventilation System which removes heat from the containment under normal plant operating conditions. Under accident conditions, a portion of the air flow passes through the filter sections before being mixed with the main stream and cooled. Receipt of a safety injection signal will automatically place the CARCFS in operation. The system can also be manually operated from the control room.

The CARCFS is designed as a Category I system. The air handling units, the air flow distribution header, and the service water cooling piping are located outside the primary concrete shield for missile protection and at an elevation that precludes flooding under loss-of-coolant accident conditions. All system components are protected against the differential pressure that may occur during the rapid pressure rise in the containment following a loss-of-coolant accident. Environmental qualification tests simulating accident environments have been performed on the fan motors in the air handling units to assure that they will perform satisfactorily under post-accident conditions. The CARCFS equipment, including fans, cooling coils, damper doors, filters, and ducting, is accessible for inspection and maintenance during normal plant operation. The system is designed to permit functional testing of components periodically and after component maintenance.

We have reviewed the containment heat removal systems for conformance with General Design Criteria 38, 39 and 40, and have found them to be acceptable.

6.2.3 Containment Isolation Systems

The Containment Isolation System is designed to isolate the containment atmosphere from the outside environment under accident conditions. Double barrier protection, in the form of closed systems and/or isolation valves, is provided so that no single valve or piping failure can result in loss of containment integrity. Lines penetrating the containment, up to and including the second isolation barrier, are designed to the same seismic criteria as the containment and are considered to be extensions of containment. Isolation valves inside containment are protected against missiles which could be generated under loss-of-coolant accident conditions.

The automatic isolation valves are tripped closed by one of two containment isolation signals. The first signal is derived from the safety injection signal and closes most of the automatic isolation valves. These valves are in process lines that have no post-accident safety function or would not result in damage to equipment if isolated. The second isolation signal is derived from actuation of the containment spray system. The valves closed by this signal are in lines which provide cooling water and seal water to the reactor coolant pumps.

We have reviewed the isolation valve arrangements for conformance to General Design Criteria 54, 55, 56, and 57, and conclude that the design meets the intent of these criteria.

6.2.4 Combustible Gas Control Systems

Following a loss-of-coolant accident, hydrogen may accumulate inside the containment. The major sources of hydrogen generation include: (1) a chemical reaction between the fuel rod cladding and the steam in contact with the cladding, (2) corrosion of aluminum by the alkaline spray solution, and (3) radiolytic decomposition of the cooling water in the reactor core and the containment sumps. The generation of sufficient hydrogen could lead to potentially combustible mixtures in the containment.

The applicant's analysis of post-LOCA hydrogen generation, which is based on AEC Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," indicates that the hydrogen concentration will not reach the lower flammability limit of 4 v/o until 23 days after the accident. Our analysis of hydrogen generation in the containment confirms the applicant's results.

To preclude the accumulation of combustible gas mixtures following a LOCA; a hydrogen recombination system is provided. The Category I Hydrogen Recombination System consists of two redundant flame recombiner units. Either unit will be capable of maintaining the hydrogen concentration in the containment below the lower flammability limit. A separate control station will be provided for each recombiner unit. Provision has been made to functionally test the Hydrogen Recombination System during normal plant operation and the testing frequency is included in the Technical Specifications.

Hydrogen gas must be supplied to the flame recombiner system as fuel, and oxygen gas must eventually be supplied to the containment to replace the oxygen consumed in the recombination process. Since hydrogen and oxygen are not kept at the site in large quantities, bulk gas would have to be brought to the site. The applicant has stated that sufficient hydrogen and oxygen can be brought to the site in about five days following a loss-of-coolant accident. At this time, the hydrogen concentration in the containment would be about 2.6 v/o.

A sampling system has been provided to permit monitoring of the combustible gas concentrations in the containment atmosphere following a loss-of-coolant accident. Sample lines originate in each air handling unit of the Containment Air Recirculation Cooling and Filtration System. The CARCFS, with only three of the five air handling units operating, is capable of mixing the containment atmosphere. Therefore, hydrogen stratification should not occur in the containment and the samples taken will be representative of the containment atmosphere.

The applicant has also provided a backup purge system that is capable of maintaining the hydrogen concentration in the containment below 3 v/o. The purged containment air would be filtered and exhausted from the plant stack.

Based on our review of the systems provided for combustible gas control following a loss-of-coolant accident, we have concluded that the systems meet the recommendations of AEC Regulatory Guide 1.7 and are, therefore, acceptable.

6.2.5 Leakage Testing Program

Leakage testing of the reactor primary containment and associated components is intended to provide preservice and periodic verification of the leaktight integrity of the containment.

The applicant has stated in the FSAR in Section 5.1.7 that the primary reactor containment and its components have been designed so that periodic integrated leakage rate testing can be conducted at a test pressure corresponding to the calculated peak accident pressure. Penetrations, including personnel and equipment hatches, airlocks, and isolation valves, have been designed to provide individual leak testing at calculated peak accident pressure.

We have reviewed the provisions for leakage testing and conclude that the containment system will permit containment leakage rate testing in compliance with "Reactor Containment Leakage Testing for Water Cooled Power Reactors," 10 CFR 50, Appendix J, and is acceptable.

6.3 Emergency Core Cooling System (ECCS)

6.3.1 Design Bases

The basic design and layout of the emergency core cooling system for the Indian Point 3 plant are similar to those developed and approved for the Zion and Indian Point 2 plants. The design bases are to prevent fuel and cladding damage that would interfere with adequate emergency core cooling and to mitigate the amount of clad-water reaction for any break size in the primary coolant system up to a double ended rupture of the largest primary coolant line. These requirements are intended to be met even with the minimum effectiveness of the ECCS, that is, operation assumed without offsite power and with only two of the three onsite diesel generators operable.

6.3.2 System Design

The emergency core cooling system consists of a high-head safety injection system, a low-head safety injection system, and an accumulator injection system.

The three high-head safety injection pumps deliver water to two separate discharge headers. The flow from each header is then injected into each of the four cold legs of the reactor coolant system.

As shown in Figure 6.2-1 of the FSAR, the high-head safety pumps deliver borated water to one of these discharge headers. The boron injection tank is located on the discharge side of the high-head pumps to minimize the time to insert negative reactivity into the core. Should one of the three high-head pumps fail to operate, water would still be pumped through the boron injection tank and then on to one of the discharge headers. As discussed in Section 7 of this report, the system was modified as a consequence of the staff review so that it now meets our single failure criterion.

Four passively activated accumulators are provided to reflood the core during the loss-of-coolant accidents resulting from intermediate or large size breaks. The four accumulators discharge through the low head safety injection lines to the four cold legs of the primary system. During normal operation, the accumulators are isolated from the primary coolant system by two check valves in series. A normally open gate valve is also located in the lines between each accumulator and the cold leg piping. In order to assure that the gate valve will be open when operation of the accumulator is required, the design includes automatic valve opening on a Safety Injection signal. There is a valve position indication in the control room, and audible alarms sound when the valve is not fully open. Each cold leg is connected to one accumulator by a 10-inch line.

The boron injection tank is located on one of the high head SIS delivery lines, and is normally isolated via motor operated isolation valves. Appropriate safety injection system activation signals will place the boron injection tank on line for delivery, and the system design is such that two-of-three high head pumps could discharge through the boron injection tank.

Two residual heat removal pumps provide low-head safety injection emergency coolant flow which recovers the core following blowdown. These residual heat removal pumps take suction from the refueling water storage tank. Only one of these residual heat removal pumps is required to meet the design objectives of the low-head injection system, therefore, this system can tolerate a failure of an active component. By proper valve arrangements the low-head system can be directed to discharge to the core through two of the hot legs. However, premature injection through the hot legs is prevented by the Technical Specifications that require locking off of the power to the valves controlling injection flow through the hot legs.

At the end of the injection phase the emergency core cooling system is then aligned for the recirculation phase. Two modes of operation are possible during the recirculation phase. One mode of operation establishes a flow path that is completely internal to the containment, the other mode circulates sump water outside of the containment. The internal recirculation loop utilizes the recirculation pumps which draw water from the recirculation sump. This water is cooled in the residual heat exchangers and then pumped to the core and the containment sprays. The cycle is completed when the spray water falls to the containment floor and the ECCS water spills out of the break and then flows to the sump.

If the primary system break is small, the reactor coolant pressure at the end of the injection phase may be above the shut-off head of the recirculation pumps. Under these circumstances the external recirculation mode will be used. In this cooling mode, water is taken from containment sump by the residual heat removal pumps, cooled, and then injected into the core by way of the high head safety injection pumps.

Care has been taken to minimize possible radiation effects due to this external recirculation path. Discharges from pressure relieving devices are collected in closed systems and radioactive leakage from pumps, flanges, and seals in this external loop has been limited to 999 cubic centimeters per hour. The staff has calculated that the dose at the exclusion distance from this leakage is about 0.1 rem (Thyroid) during the first two hours following a LOCA.

6.3.3 Performance Evaluation

On June 29, 1971, the AEC issued an Interim Policy Statement[1/] containing interim acceptance criteria for the performance of emergency core cooling systems for light-water cooled nuclear power reactors. A public rule making hearing on the Interim Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors has been held.

In accordance with the Interim Policy Statement, the performance of the emergency core cooling system is judged to be acceptable if the course of the loss-of-coolant accident is limited as follows:

[1/] 36 Federal Register, 12247.

1. The calculated maximum fuel element cladding temperature does not exceed 2,300 °F.
2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of cladding in the reactor.
3. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling, and before the cladding is so embrittled as to fail during or after quenching.
4. The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

Indian Point 3 has been analyzed using the Westinghouse Evaluation Model specified in Appendix A, Part 3 of the Interim Policy Statement The results of the analyses of the ECCS performance capability are provided in Amendments 6, 9, and 19 to the FSAR.

The applicant presented the results of analyses of calculated peak clad temperatures for a spectrum of pipe break sizes up to and including the double-ended rupture of the largest coolant pipe. The calculated peak clad temperatures, assuming normal plant operation, at 102% of the ultimate power level of 3216 MWt are as follows:

Break Size and Type	Peak Clad Temperature (°F)
Double Ended Hot Leg (Guillotine)	1034
Double Ended Cold Leg (Guillotine)	2003
Double Ended Cold Leg (Split)	1995
0.6 Double Ended Cold Leg (Guillotine)	1604
0.6 Double Ended Cold Leg (Split)	1924
3.0 ft^2 Cold Leg (Split)	1664
0.5 ft^2 Cold Leg (Split)	1124

The results of the analyses indicated that for each of the assumed pipe breaks, the total core metal-water reaction is less than 1%. The maximum hot-spot metal water reaction is 2.3%, and the total core metal-water reaction is less than 0.1%. Therefore, no significant amount of cladding would become embrittled and the core geometry would be preserved. As a result, the core would remain amenable to cooling and the long-term removal of decay heat would be carried out effectively by the emergency core cooling system.

On the basis of our evaluation, we consider that the predicted functional performance of the Indian Point 3 ECCS for the full spectrum of break sizes is in accord with the Commission's Interim Policy Statement and satisfies the Interim Acceptance Criteria for Emergency Core Cooling Systems.

The above analyses do not include the effects of fuel densification. This topic is discussed in Section 6.5 of this report.

We have reviewed the applicant's analysis of the consequences of small breaks requiring the operation of the emergency core cooling system. The peak clad temperature associated with the spectrum of small breaks analyzed occurs at the 3.5 inch break size, and is only 1200°F. In view of the relatively low peak clad temperature for the worst case small break, we conclude that the information provided by the applicant provides reasonable assurance that the ECCS performance is adequate to accommodate small breaks.

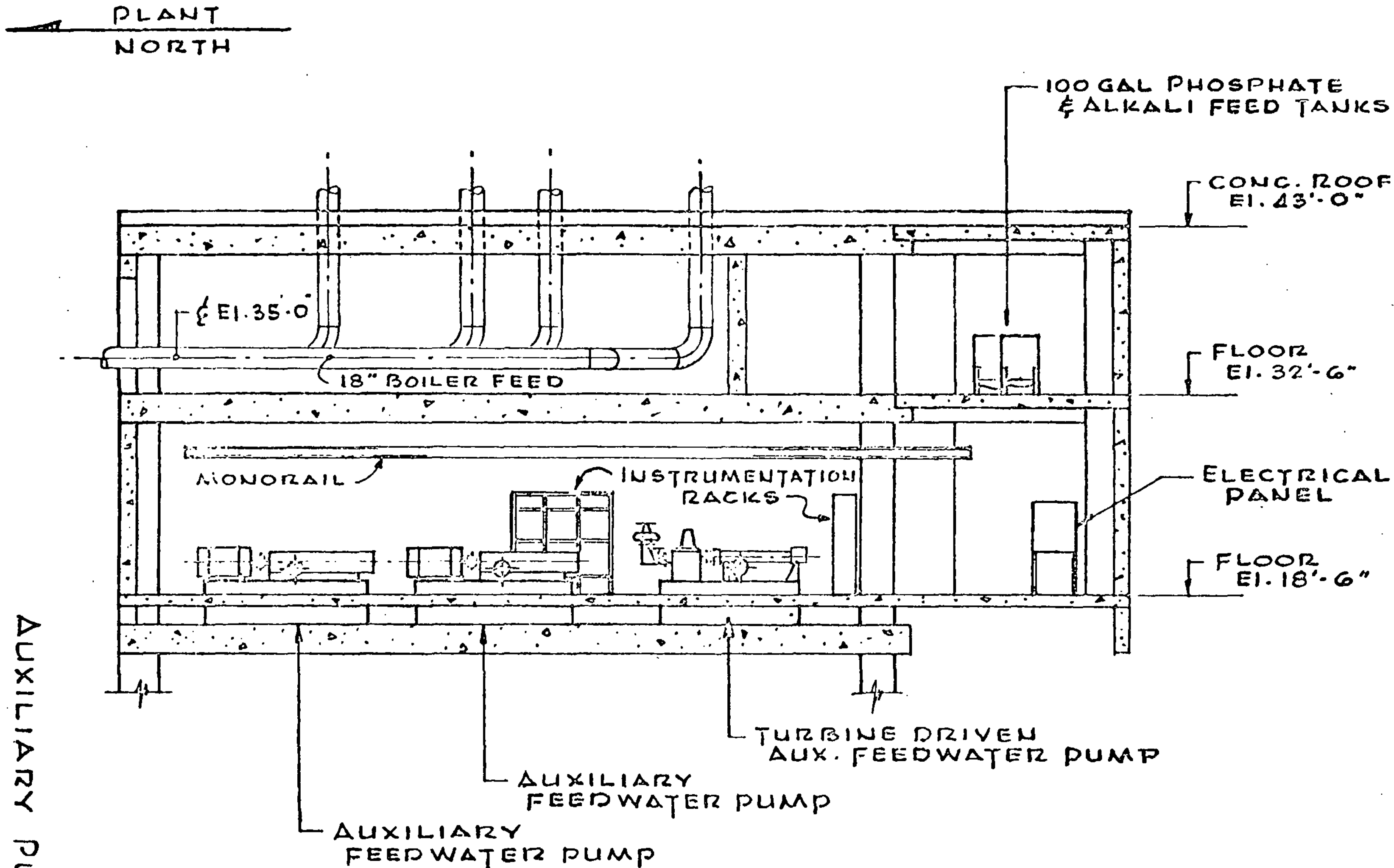
6.4 Auxiliary Feedwater System

The auxiliary feedwater system removes heat from the secondary system whenever there is a loss of normal feedwater. Normal feedwater can be lost by pipe breaks, pump failures, valve malfunctions, or loss of offsite power. The auxiliary feedwater system also influences the fuel cladding peak temperature following a small break in the primary coolant system.

Auxiliary feedwater is supplied by two motor-driven auxiliary feedwater pumps and one steam turbine-driven auxiliary pump. Each motor-driven pump will deliver 400 gpm (at a head of 3200 feet) and the steam turbine-driven pump will supply 800 gpm (at a head of 3300 feet).

These pumps draw their water from the condensate storage tank and have an alternate supply of water stored in a 1.5 million gallon tank. A third supply of auxiliary feedwater is a city water system that is piped into the auxiliary feedwater pump room. (See Section 9.2 of this report for a further description of the condensate storage facilities.) Electric power for the two motor-driven auxiliary feedwater pumps is automatically obtained from the diesel generators in the event of a loss of offsite power.

Several modifications have been made to this system in order to give it additional protection in the unlikely event of high energy line breaks outside of the containment. The auxiliary feedwater lines are directly connected into the feedwater system and experience the same pressure as the feedwater system. The staff had a concern that a break in an auxiliary feedwater line within the room that houses the motor-driven and the steam turbine-driven auxiliary feedwater pumps might result in back flow from the feedwater system and could possibly flood these three pumps. Because of this concern, the applicant put check valves in the piping that connects the discharge side of these pumps with the normal feedwater system. These check valves are located outside of the auxiliary feedwater pump room and prevent backflow from the feedwater system into the auxiliary feedwater system. (See Figure 6.4)



AUXILIARY PUMP ROOM

FIGURE 6.4

Another modification made by the applicant as a result of our review is additional protection of the electric motor-driven auxiliary feedwater pumps from a high temperature-high humidity environment. The staff postulated that a break in the steam supply to the steam turbine-driven auxiliary feedwater pump might result in temperature and humidity conditions in the pump room for which the electric motor-driven pumps were untested. These motor-driven pumps are "drip-proof," but their operability at elevated temperatures and in a steam environment has not been demonstrated. Consequently, the applicant has installed two redundant valves in the steam supply line to the auxiliary feedwater turbine-driven pump. These valves are outside of the room that houses the auxiliary feedwater pumps. Each valve is signaled to close automatically on high temperature in the pump room. Each valve has its own separate temperature sensor. There is control room indication of each valve's position, and an alarm will sound upon closure. Operation of these valves would limit the temperature and humidity rise in the pump room due to a break in the steam supply to the steam turbine auxiliary feedwater pump.

The applicant has examined the consequences of pipe ruptures in the vicinity of the auxiliary feedwater pumps which might cause flooding in the pump room. The applicant has modified the drainage capabilities of the pump room to prevent water levels from reaching a depth of 14 inches off the floor from such postulated breaks. At the 14-inch level, water would begin to touch the bottom of the electric motor-driven auxiliary feedwater pumps. The applicant has also installed pipe restraints on feedwater lines in the room above the auxiliary feedwater room. This was done to eliminate any concrete from the pump room roof falling onto the pumps as a result of whipping of the pipes after a postulated pipe rupture.

The staff has reviewed these modifications and found them acceptable.

Because of the important role of the auxiliary feedwater system following a loss of feedwater and also following small breaks in the primary coolant system, the staff has reviewed the design criteria that this system meets. The applicant supplied a list of these design criteria in Supplement 20 of the FSAR.

The principal design criteria of the auxiliary feedwater system are that (1) the distribution piping is Category I throughout, (2) the system can withstand a single failure and still meet its performance requirements, (3) the pumps are driven by diverse principles - two are electric driven pumps and one is steam driven, (4) the auxiliary feedwater pumps are automatically started by safety injection signals or by a plant trip concurrent with loss of offsite power, and (5) one electric driven pump has sufficient capacity to limit the steam generator water level from dropping below 10 feet above the steam generator tube sheet. One electric driven auxiliary feedwater pump has enough capacity to limit the primary coolant heat up after a loss of offsite power so that primary water is not expelled through the pressurizer relief valve.

In view of the design modifications that the applicant has made to limit the consequences of pipe whip, flooding, and temperature and pressure transients in the pump room, and in view of the design criteria that were in effect at the time of the construction permit, we have concluded the design of the auxiliary feedwater system, as modified, is acceptable.

We have also made an independent analysis of the auxiliary feedwater system's capability to remove decay heat following a loss of offsite power. Based on our analysis, one electric driven auxiliary feedwater pump has the capacity to meet the design criteria of maintaining at least ten feet of water above the steam generator tube sheet and preventing the primary coolant system from discharging liquid from the pressurizer relief valve after a loss of offsite power.

6.5 Fuel Densification

The fuel in current Westinghouse reactors is uranium oxide, UO_2, in the form of pellets. In the manufacturing process the UO_2 powder is compacted into pellets and sintered to form a ceramic-like solid. The as-manufactured pellets have densities less than the maximum theoretical density of void-free UO_2. The void volume is distributed in small voids or pores throughout the pellet.

Some Westinghouse fuel has experienced densification after irradiation. Densification occurs as a result of high temperature changes in the micro-structure of the oxide in the hotter central regions of the pellets and as a result of the disappearance or annihilation of small pores from the oxide matrix during irradiation.

Densification of fuel causes a decrease in the volume of the fuel pellet with corresponding changes in the pellet radius and length. There are three principal effects associated with fuel densification:

(1) A decrease in the pellet length will cause the linear heat generation rate to increase by an amount in direct proportion to the percentage decrease in pellet length.

(2) A decrease in the pellet length can lead to generation of axial gaps within the fuel column, resulting in increased local neutron flux and the generation of local power spikes.

(3) A decrease in the pellet radius increases the radial clearance between the fuel pellet and fuel rod cladding, causing a decrease in the gap thermal conductance and, consequently, in the capability to transfer heat across the radial gap. This decrease in heat transfer capability will cause the stored energy in the fuel pellet to increase. A decrease in radial gap conductance also will degrade the heat transfer capability of the fuel rod during various transient and accident conditions.

In summary, the effects of fuel densification cause the fuel rod to contain more stored energy, increase the linear heat generation rate of the pellet, decrease the heat transfer capability of the fuel rod and create the potential for a local power spike in any fuel rod.

To assess the safety implications of fuel densification, all of these effects were evaluated for the Indian Point 3 reactor under all modes of reactor operation.

Prior to initiating the staff review of the effect of densification on the Indian Point 3 fuel, the staff completed a detailed review of fuel densification effects in connection with Point Beach Unit 2 (Docket No. 50-301) which also has a Westinghouse nuclear steam supply system. As a result of that review, we concluded that Westinghouse analytical techniques conservatively predict the effects of fuel densification and are generally applicable to other Westinghouse designed plants. The bases for our conclusions stated below, including results of staff calculations, were presented in the additional testimony prepared for the Point Beach Unit 2 hearing. The applicant has used the methods developed by Westinghouse for Point Beach Unit 2 to evaluate effects of densification at Indian Point 3.

Using the previously approved methods, a determination was made of how rapidly the fuel densified, the clad creepdown, the time required for unsupported clad tubing to flatten (time-to-collapse), and the effects of fuel densification on gap conductance. These determinations are discussed below.

Examinations of density changes in irradiated fuel by Westinghouse have shown that, for exposure times of less than 14 hours of power operation, no temperature-dependent densification has occurred, but that after 2000 hours of reactor operation fuel densification has probably been completed.

The properties of the UO_2 pellets in reactor fuel assemblies are dependent on the many variables which exist in the manufacturing process. We considered how the manufacturing process could affect densification and concluded that we are unable to attribute densification to the control of one or more process parameters at this time. Therefore, until further irradiation data are accumulated, it will be assumed that all fuel will densify to an extent consistent with present observations.

Westinghouse examined the effects of initial density, peak power, burnup, fission rate and internal gas pressure on the densification process. The only clear conclusions that can be drawn at this time are that there is increased fuel column shrinkage with decreased initial density and the assumption should be made that axial shrinkage is greater than radial shrinkage.

Because of these unknowns the evaluation model specified by the staff conservatively requires the assumption of instantaneous densification.

Cladding creepdown is the term used to indicate the phenomenon which affects the geometry of the gap between the fuel pellets and the cladding.

The applicant's creep model (Westinghouse Report E-PA-475, "Clad Creep Model," Westinghouse Proprietary, October 1972) was normalized to match the measurements of fuel rods which had been subjected to reactor operating conditions. These fuel rods had physical characteristics similar to those of the Indian Point 3 prepressurized fuel rods, and the environmental conditions were similar to those expected for the Indian Point 3 plant. On this basis we conclude that the cladding creepdown calculation method utilized for Indian Point 3 is acceptable.

Time-to-collapse is the term used to indicate the time required for an unsupported clad tubing to become dimensionally unstable and flatten into the axial gap volume caused by the fuel pellet densification. The data on which the Westinghouse collapse model is based were for cladding which is similar to that used for Indian Point 3. Using the previously approved time-to-collapse model, the applicant calculates that there will be no collapsed rods during the first fuel cycle.

Gap conductance is a measure of the ability to transfer heat from the fuel pellet to the cladding. The effect of densification is to increase the radial gap between the fuel pellet and the cladding, thus decreasing the gap conductance and increasing the fuel pellet stored energy. The staff has established guidelines for calculating the gap conductance used in analyzing the behavior of the fuel for all modes of reactor operation. Westinghouse has followed these guidelines in developing an acceptable model for the prediction of the gap conductance. This model has been used for the Indian Point 3 plant, and includes the effects of initial diametral gap size, the amount of fill and fission gas (pressure and chemical composition) in the gap, the amount of densification, the surface roughness of the fuel and clad and their material properties, and, in the case of fuel-to-clad contact, the contact pressure.

In summary, the staff's review of the applicant's densification methods concludes that:

(1) The time to collapse method used by Westinghouse for the Indian Point 3 plant is acceptable.

(2) An acceptable calculational method has been used to describe the cladding creepdown effect that tends to increase gap conductance with lifetime.

(3) The Westinghouse calculations of gap conductance used in the performance analysis are acceptable.

Having demonstrated that the previously approved Westinghouse fuel densification models are applicable to Indian Point 3, the applicant then determined how fuel densification would affect the operation of Indian Point 3. A preliminary report filed by the applicant on April 2, 1973 and a final report to be filed, address the effects of fuel densification on the operation of Indian Point 3.

The effects of fuel densification on overpower transient limits, on the departure from nucleate boiling (DNB) limits, and the loss-of-coolant accident limits were presented by the applicant in its April 2, 1973 preliminary report. This fuel densification report utilized a total peaking factor, F_Q, of 2.56. Of the three limits examined by the applicant, the loss-of-coolant accident limit established the most restrictive linear heat generation rate. In order to remain within the 2300°F temperature limit required by the Interim Policy Statement the peak linear heat generation rate, with fuel densification, is 16.8 kW/ft. By comparison, the peak linear heat generation rate without fuel densification effects is listed in Table 3.2.2-1 of the FSARas 17.5 kW/ft.

Analyses of the effects of fuel densification on the loss-of-coolant accident limit presented in the preliminary report were based upon the double-ended rupture of a primary system cold leg. This particular break was selected because it had the highest fuel clad temperature of all break sizes analyzed and reported in the FSAR. (See Section 6.3 of this Safety Evaluation Report). It is assumed that when densification effects are considered for other sized breaks, this break will still result in a higher fuel clad temperature than any other sized break. The final submittal will examine other sized breaks to verify that the cold leg break is still limiting. The final report will also review the effects of fuel densification of the loss-of-flow transient, steam line rupture, control rod ejection and other accidents and transients. Analyses of similar Westinghouse nuclear plants indicate that these other transients and accidents will not result in peak linear heat generation rates lower than that set by the cold leg break. The staff will review the final report to verify this.

The Indian Point 3 reactor design parameters have been compared to the Zion plant and many of these parameters are listed in Table 4.4-1 of this report. This comparison is useful because the two plants are quite similar and the effects of fuel densification on the Zion plant has been reviewed by the staff. The plants are quite similar except that Indian Point 3 has a 7% lower rated power (3025 MWt vs 3250 MWt), has a higher initial fuel density and a higher initial fuel pressurization. The control rod patterns are different.

6.6 Post Loss-of-Coolant Accident Protection (PLOCAP)

The possibility of reactor vessel failure as a result of thermal shock caused by emergency core cooling action in the unlikely event of a LOCA during the later portions of plant life was discussed during the construction permit phase of our review. The injection of cold water into a hot reactor pressure vessel raises the possibility that a vessel embrittled by irradiation and having a small internal defect could fail. During the construction permit review the applicant committed to the development of an additional engineered safety feature, the post loss-of-coolant accident protection (PLOCAP) system, which would provide a means of covering and cooling the core in the event reactor vessel integrity is lost. A conceptual design of a PLOCAP system was submitted which subsequently was integrated into the existing ECCS in such a manner that capability of the ECCS to meet its design objectives would be maintained.

Recent analyses by the reactor vendor indicate that cold water injection toward the end of the vessel's service life might cause defects of the maximum allowable size to grow, but the vessel would not be expected to fail under these conditions.

Additional data needed to resolve the thermal shock problem are expected to be provided by the Oak Ridge National Laboratory Heavy Section Steel Technology (HSST) Program. Since the reactor vessel materials will not be significantly changed by irradiation during the initial five years of operation, no thermal shock problem will exist before the HSST program is completed and the final data analyzed.

Fracture toughness of the vessel material will be monitored by testing of the surveillance samples withdrawn from the reactor at specified intervals. The Indian Point 3 surveillance program is in compliance with the intent of Appendix H of 10 CFR Part 50.55 A. The design of Indian Point 3 has the capability of annealing the reactor vessel in place to permit partial recovery of fracture toughness properties.

The applicant has provided in the design of Indian Point 3 all equipment and structural requirements such as space to accommodate cavity flood tanks; the cavity sump and piping are in place as are the cavity flood pump cubicles and associated piping. Should it be deemed necessary, equipment to complete the PLOCAP system can be procured and installed without major revision of the station.

We have concluded that it is not necessary to provide an operational system such as PLOCAP at this time and that the provisions made in the design of Indian Point 3 for future installation of PLOCAP are acceptable.

7.0 INSTRUMENTATION AND CONTROL SYSTEMS

7.1 General

The instrumentation and control systems for Indian Point 3 have been evaluated against the Commission's General Design Criteria as published July 1971 and the Institute of Electrical and Electronics Engineers Standard, IEEE 279, "Criteria for Nuclear Power Plant Protection Systems," dated August 1968.

The evaluation of the Indian Point 3 plant was accomplished by comparing its design with that of the previously evaluated Indian Point 2 plant. In addition to the information in the FSAR various electrical diagrams were reviewed to determine that the final design conforms to the design criteria.

7.2 Reactor Trip System

The design of the reactor trip system is virtually identical to that of Indian Point 2. The basic design has been reviewed extensively in the past and we conclude that the design for Indian Point 3 is acceptable.

During our review we considered the adequacy of reactor protection for operation with less than four coolant loops in service. When operating with one of the coolant loops out of service, the reactor is normally automatically limited to 60% of rated power. However, by manual adjustment of several protection system setpoints, adequate reactor protection can be provided for operation up to 75% of rated power. We have concluded that this aspect of the design does not conform to the requirements of IEEE Std 279-1968. However, since the need for manual adjustments during reactor power operation is expected to arise infrequently and the Technical Specifications will require adjustment of overtemperature ΔT setpoints prior to increasing the power level limit, we have concluded that the design is acceptable for the Indian Point 3 plant.

7.3 Initiation and Control of Engineered Safety Feature Systems

The design of the protection systems for initiation and control of the operation of the engineered safety feature systems is functionally identical to the design for Indian Point 2. The basic design has been reviewed extensively in the past and we consider it to be acceptable. Therefore, our review of the Indian Point 3 design concentrated on those aspects of the design that differ from those of Indian Point 2.

We have reviewed the capability for testing the engineered safety feature circuits during reactor power operation. As a result of our review the design has been changed to permit more complete testing of the circuits during reactor operation. To prevent actuation of the associated engineered safety feature systems during the tests, operation of certain circuits is blocked. The continuity of the circuits that are not operational during the tests is verified using permanently installed equipment. Use of an ohmmeter is not necessary. Since automatic initiation of one train of engineered safety feature equipment is disabled during these tests, it is necessary to test the two logic trains one at a time. As a result of our review, separate annunciators have been installed on the main control board to provide unique identification of the logic train being tested. Manual initiation of safety injection is not blocked during these tests. We have concluded that this testing capability is acceptable.

We have reviewed the procedure and circuits used to change operation of the safety injection system from the injection phase to the recirculation phase following a loss-of-coolant accident. To facilitate the change in operating modes of the system, a series of eight switches are provided and these would be operated in a sequence depending on whether the high pressure injection pumps were needed in the recirculation phase. The original design was such that premature operation of certain recirculation switches could prevent operation of redundant safety injection system components. As a result of our review, the design was modified to prevent the loss of redundant functions due to the malpositioning of any single recirculation switch while there is a safety injection signal present. We have concluded that this approach is acceptable, but we have not completed confirmation of the necessary circuit changes. Prior to the issuance of the operating license, we will review the applicable schematic diagrams to verify that no single malpositioned recirculation switch will disable redundant functions when a safety injection signal is present.

We also requested that the applicant re-examine the adequacy of the information available to the reactor operator during the change-over to the recirculation phase. The original procedure required that the operator manipulate the recirculation switches in either of two sequences depending on the indicated flow in three out of four low pressure injection lines. With the original design of the power supplies for these flow instruments, a single failure could result in loss of two flow instruments. We informed the applicant of our requirement that there must be sufficient information available to the operator to complete correctly the change-over following a loss-of-coolant accident, even in the event of any single failure.

In Supplement 18 of the FSAR the applicant stated that flow indication from only two of the four low pressure injection lines was sufficient to meet this system's original design criterion. This original design criterion required that there be a measured flow of at least 600 gpm through the low pressure injection lines and this criterion is met with just two flow meters. Procedures have been modified to allow the operator to manipulate the recirculation switches based on just two flow meter readings. Based on these revised procedures and the original design criterion for the use of this system, no single power supply failure would result in insufficient information for the operator.

The original design of the Indian Point 3 safety injection system did not meet the single failure criterion. Safety injection pumps 31, 32 and 33, as shown in Figure 6.2-1 of the FSAR, were designed to provide flow down high pressure injection lines 16 and 56. The original design assigned pump 31 to line 56 and pump 33 to line 16. If either pump 31 or 33 failed to start, its pumping requirements were then to be accomplished by pump 32. This required automatic closure of valve 851A or 851B, depending on which pump failed.

This design did not meet the single failure criterion in that it relied upon a failed piece of equipment to generate a signal to initiate the operation of valve 851A or 851B.

In response to our requirement that the system be designed in accordance with the single failure criterion, the existing automatic control circuits were removed. An additional orifice was installed on line 56 to balance the flow distribution to both injection headers.

The modified system can tolerate the failure of any one of the three safety injection pumps and will still provide adequate flow down each high pressure injection line without requiring any automatic valve motion. We conclude that the modified system now meets the single failure criterion and is acceptable.

We reviewed the design to assure that all operating bypasses conform to the requirements of IEEE Std 279-1968. At our request, an additional bypass switch was installed to provide assurance that no single failure would result in a bypass of the low pressurizer pressure/low pressurizer level signal in both safety injection logic trains. On this basis, we conclude that the modified design is acceptable.

We conclude that the design of the protection systems for initiation and control of the engineered safety feature systems conforms to the requirements of the Commission's General Design Criteria and IEEE Std 279-1968 and is therefore acceptable.

7.4 Systems Required for Safe Shutdown

The instrumentation and control systems provided for safe shutdown have been reviewed, and on the basis that the design meets all applicable criteria we have concluded that their design is acceptable. The controls for the service water system were found acceptable, provided the essential header is isolated from the conventional header during reactor operation. The Technical Specifications require that this condition exist during reactor operation.

During a meeting on May 31, 1973, the applicant provided the staff with further information on the design criteria of the auxiliary feedwater system. Based on the applicant's statements, the auxiliary feedwater system meets the single failure criterion. This design criterion, as well as the other design criteria that apply to the auxiliary feedwater system, were documented in Supplement 21 to the FSAR. Based on these statements, we find the criteria for the instrumentation and control of the auxiliary feedwater system acceptable. The confirmation of the implementation of these design criteria will be done when the electrical schematics for this system are submitted by the applicant and prior to issuance of the operating license.

We have reviewed the instrumentation and controls provided outside the control room and determined that they are identical to those provided for Indian Point 2 and are acceptable.

7.5 Safety-Related Display Instrumentation

We have reviewed the instrumentation systems that provide information to enable the operator to perform required safety functions throughout all operating conditions of the plant and to monitor the course of accidents. We have concluded that the safety-related display instrumentation is acceptable.

7.6 RHR System Interlocks

During the review of this application, the staff took the position that additional protection of the low pressure Residual Heat Removal (RHR) System from possible over-pressurization was required. Motor operated valves 730 and 731 are used to isolate the suction line of the low pressure RHR system from the high pressure reactor coolant system. A letter was issued by the staff on May 2, 1973 to the applicant stating our requirements to automatically close RHR system valves 730 and 731 whenever the primary system pressure exceeded the RHR design pressure. The staff also required independent interlocks on these valves to prevent their opening whenever the primary system pressure exceeded the RHR system design pressure. Both the interlocks and the automatic closure of these valves were to be designed to meet the single failure criterion.

The applicant responded to the staff requirements in a letter dated May 25, 1973. The staff has reviewed the criteria proposed for the design modifications to be incorporated into the RHR system and finds them acceptable. Confirmation of the implementation of these criteria will be obtained when the electrical schematics for this system are submitted by the applicant prior to issuance of the operating license.

7.7 Control Systems Not Required for Safety

The applicant has stated that the functional design of the reactor control systems for Indian Point 3 is the same as that for Indian Point 2 with the exception of minor changes in equipment. We have reviewed the design and changes and have concluded that such equipment changes have not changed the functional design or degraded the safety of this plant and have concluded that these control systems are acceptable.

7.8 Seismic, Radiation, and Environmental Qualification

The seismic design criterion for the reactor protection system and engineered safety feature circuits requires that the equipment not lose its capability to perform the required safety functions during or following a safe shutdown earthquake. We have reviewed the type tests performed to demonstrate conformance with the seismic design criteria and have concluded that the seismic qualification program is acceptable.

The design criterion for safety-related equipment installed inside the containment structure is that the equipment shall be capable of functioning under the post-accident temperature, pressure, humidity and radiation conditions for the time periods required. We have reviewed the type tests performed to demonstrate conformance with these design criteria and have concluded that the environmental and radiation qualification program is acceptable.

7.9 Common Mode Failures and Anticipated Transients Without Scram

In connection with our review of potential common mode failures, we have considered the need for means of preventing common mode failures from negating protective functions and of possible design features to make tolerable the consequences of failure of scram during anticipated transients. This concern is applicable to all light water cooled power reactors.

This problem is being studied on a generic basis. If the probability of any of the events considered is determined to be sufficiently high to warrant consideration as a design basis for plants having a nuclear steam supply system similar to Indian Point 3, suitable design modifications to reduce the probabilities or to limit the consequences to acceptable levels may be necessary.

8.0 ELECTRIC POWER

8.1 General

The design of the safety-related electric power systems for Indian Point 3 is similar to that for Indian Point 2. Therefore, our review concentrated on those aspects of the design that have changed since our evaluation of Indian Point 2 and those aspects of the design affected by changes in regulatory requirements.

8.2 Offsite Power

Two 138 kilovolt (kV) circuits connect the Buchanan switchyard to the Millwood Substation which is connected to the Consolidated Edison, Niagara Mohawk, and Connecticut Light and Power transmission networks. Two additional 138 kV lines, using separate routes from the first two lines, connect the Buchanan switchyard to the Orange and Rockland system.

Two 138 kV circuits connect the Indian Point station and the Buchanan switchyard. These circuits carry the output power from Indian Point 1 and supply power to the station auxiliary transformers for Indian Point 2 and Indian Point 3. The normal source of power for startup of Indian Point 3 and the preferred source of power in the event of an accident is the station auxiliary transformer. A second source of offsite power is available to Indian Point 3 via two underground 13.8 kV circuits from the Buchanan switchyard. In addition to power from the transmission network, power is available from two gas turbine generators, one located in the Buchanan substation and one located on the Indian Point site, which can be connected to the 13.8 kV circuits.

We have concluded that the offsite power system provides two physically independent circuits that connect with the onsite power distribution system in accordance with General Design Criterion 17 and is acceptable.

8.3 Onsite Power

8.3.1 A-C Power Systems

The original design of the onsite emergency power supply for Unit No. 3 employed four 480 V buses energized upon loss of normal power by three diesel generators, two of which were required to furnish energy to engineered safety features. The applicant had proposed an automatic system of cross-connecting sources and loads. Both the ACRS and the AEC staff believed that the onsite power sources needed greater independence, at least to the extent that they could not be connected together with automatically operated devices.

Consequently design modifications were made so that the emergency a-c power is now supplied by three physically and electrically independent diesel generator sets. The redundant engineered safety feature and safe shutdown loads are arranged in three groups, each group powered from its assigned diesel generator in the event of loss of offsite power. Any two of the three load groups and their associated diesel-generator sets are adequate to mitigate the consequences of an accident. No manual or automatic interconnections or transfers are necessary. We have concluded that the design of the onsite a-c power system is in conformance with Regulatory Guide 1.6 "Independence Between Redundant Safety (Onsite) Power Sources and Between Their Distribution Systems" and Regulatory Guide 1.9 "Selection of Diesel Generator Set Capacity for Standby Power Supplies" and with IEEE Std. 308 and is acceptable.

8.3.2 D-C Power Systems

The applicant originally proposed the use of two d-c power systems and automatic transfer devices to supply power to the three engineered safety feature load groups. We concluded that such a design could unduly compromise the independence of redundant safety systems. To meet our requirements, the applicant modified the design to eliminate the need for automatic transfers between redundant power sources. This was accomplished by the addition of a third d-c power system.

We have concluded that the modified design of the d-c power system is compatible with the a-c power system, meets the regulatory positions of Regulatory Guide 1.6, and is acceptable.

As a result of the changes in the design of the onsite d-c power systems discussed above, the instrument power supplies will be changed. We have informed the applicant of our requirement that the power supplies for the protection system must be designed in accordance with IEEE Std 279-1968. Prior to issuance of the operating license, we will review the design changes to assure that the requirements of IEEE Std 279-1968 are met.

8.4 Separation and Identification of Redundant Protection and Emergency Power Systems

We have reviewed the means used to provide physical separation between redundant protection and emergency power systems.

The diesel generators and their local panels are located in three separate rooms of a Category I structure. Two batteries are located in separate battery rooms with no other equipment. The third battery (and its associated equipment), which was added to comply with our requirement, is located in the room with the diesel generator to which it supplies power. The applicant has examined the environmental conditions associated with this location and has found that operation of the battery and the diesel generator will not be adversely affected at this location. We have concluded that this location is acceptable.

The criteria used for the installation of cables and cable trays require a minimum of one foot between redundant circuits spaced either horizontally or vertically except that a minimum of three feet is required between redundant heavy power circuits spaced vertically. Where these distances are not provided, fire barriers are installed between redundant circuits. Two electric cable tunnels are provided between the control building and the containment penetration area, and separation is provided by locating redundant channels on opposite sides of the tunnels.

The identification methods used to distinguish between safety and non-safety equipment and between redundant channels of safety systems are color and numeric codes.

We have concluded that the identification and separation of redundant protection and emergency power systems is comparable to recently licensed operating plants and is acceptable on that basis.

8.5 Diesel Fuel Oil System

We reviewed the design of the power and control systems for the diesel fuel oil system and concluded that the design originally proposed by the applicant was unacceptable. Specifically, all three fuel oil transfer pumps were powered from non-safety buses, their power supplies would have been disconnected in the event of a loss of offsite power, and the control system was vulnerable to single failures. To meet our requirements, the system was modified so that the control system would meet the single failure criterion. Two fuel oil transfer pumps were powered from safety-related load centers that are automatically energized by the diesel generators. A new power supply for the third pump was added and designed in accordance with the requirements of IEEE Std 308. With the addition of this new power supply the diesel fuel transfer system can sustain a single failure and still supply an adequate amount of oil. (See Section 9.5 of this report.) On the basis that the power supply for the fuel transfer system meets the requirements of IEEE Std. 308 we conclude that the control and power systems are acceptable.

9.0 AUXILIARY SYSTEMS

9.1 Fuel Storage Systems and Handling Systems

The new fuel storage pool and the spent fuel storage pool are located in a Category I structure. The insertion and removal of fuel assemblies from the reactor vessel into the storage building is accomplished under borated water which serves as a transparent shield and cooling medium.

9.1.1 New Fuel Storage

New fuel assemblies are stored in a dry vault within the Fuel Storage Building which has capacity for one-third of a full core loading, with each fuel assembly located on a center-to-center spacing of 21 inches. Should the dry vault inadvertently be flooded with unborated water, the maximum k_{eff} for new fuel at this spacing will not exceed 0.90, a value well below criticality. Each new fuel assembly, for initial fueling and subsequent refueling, will move from the dry vault to the spent fuel storage pool, and then through a horizontal transfer tube into the refueling cavity within the reactor containment building, prior to insertion into the reactor.

We have reviewed the new fuel storage and handling facility and conclude that:

(1) Gravity drainage has been provided to handle inadvertent water flooding.

(2) That such flooding would not result in a critical assembly.

(3) The hatch covering the storage area has sufficiently strong lifting lugs.

(4) The in-place hatches can support all expected loads.

(5) The air-motor driven conveyor car which transfers new (and spent) fuel between the reactor cavity pool and the spent fuel storage pool has been used successfully in numerous operating reactors and can be expected to give reliable operation in this facility.

On this basis, we conclude that the new fuel storage vault is acceptable.

9.1.2 Spent Fuel Storage

The spent fuel storage pool is capable of accepting and storing one and one-third spent cores from the reactor. It is constructed of reinforced concrete and has a stainless steel liner. The spent fuel center-to-center spacing is designed to prevent the k_{eff} from exceeding 0.90 in unborated water. However, water in the fuel pool will be borated to the same concentration as the water in the refueling water storage tank which contains 1.4 weight percent boric acid.

There are no gravity drains in the spent fuel storage pool. Cooling water inlet and outlet connections are located such that failure of any pipeline will not completely drain the fuel pit and a minimum of seven feet of water would remain on top of the stored fuel elements. The control room operator receives a low level alarm upon loss of pool water and can initiate remedial action.

The spent fuel storage pool's capability to withstand tornado generated missiles (see Section 3.5 of this report) has been reviewed by the staff. If the design tornado missile of an automobile traveling 25 feet above the ground should strike the concrete Category I structure housing the spent fuel pool, it would not cause damage to the pool. If the design tornado missile consisting of a four inch by twelve inch by twelve foot wooden plank should strike the metal siding portion of the building surrounding the Category I spent fuel pool, it could penetrate the siding. However, it has been calculated that such a wooden missile could not sink through all of the 26 feet of water that covers the spent fuel to cause any significant damage to the stored fuel.

Based on the above we conclude that the spent fuel storage pool meets the tornado generated missile criteria and is acceptable.

9.1.3 Spent Fuel Pool Cooling and Cleanup System

The spent fuel pool cooling loop removes the residual heat from the fuel stored in the spent fuel pool, and consists of a pump, heat exchanger, filter, demineralizer, piping, valves and instrumentation. Approximately five percent of loop flow circulates through a demineralizer and filter, for water purification. The system is a non-redundant, non-seismic designed system; however, failure of this system will not compromise plant safety. The normal makeup water supply to the pool is from a non-seismic designed system which uses the Primary Makeup Water Storage Tank as the source.

We have reviewed the system configuration, piping, pump capacity, demineralizer capacity, and heat dissipation capability and find that the system features are comparable to other licensed nuclear power plants. In the event of a loss of pool water there are other available sources of make-up water nearby, such as the fire protection system, which can be hooked up in a timely manner.

We conclude that the spent fuel cooling and cleanup system is acceptable on the basis that there are alternate sources of water that can be used if the normal cooling system should fail.

9.1.4 Handling Systems

The major handling systems are located within the containment building and in the fuel storage building. A gantry type polar crane is used within the containment building for handling heavy loads such as shield plugs, the reactor vessel head, and the upper and lower vessel internals. Lighter loads, such as a fuel element and those loads requiring more sensitive positioning, are handled in the containment by a rectilinear bridge and trolley manipulator.

We have evaluated all phases of polar crane operation. Of particular concern was the inadvertent dropping of the shield plugs, head, and upper and lower vessel internals onto the reactor vessel. The applicant has provided results of calculations to verify that the shear stress of all supports and piping would not be exceeded if these heavy objects were dropped on them.

A rectilinear bridge and trolley manipulator, running on rails at the edge of the reactor cavity, is equipped with a long tube and pneumatic gripper which inserts and withdraws fuel assemblies from the core. The transfer system from the reactor cavity to the spent fuel storage pool moves each fuel assembly on a conveyor car mounted on tracks. The conveyor car is driven by an air motor through the transfer tube connecting the reactor cavity to the spent fuel storage pool.

Within the fuel storage building lighter loads are carried by a monorail hoist while heavier loads are handled by an overhead bridge crane.

The spent fuel pool bridge is a wheel-mounted walkway which carries an electric monorail hoist on an overhead structure. A handling tool suspended from the hoist moves the spent fuel with the tool length designed to limit maximum lift of spent fuel to a safe shielding depth. We have determined that the design uplift capacity of the hoist is less than the uplift strength of the fuel, and the spent fuel racks. If the spent fuel should become stuck in the fuel rack, the hoist lift capacity is insufficient to damage the fuel, or the racks.

The spent fuel building is equipped with an overhead bridge crane for movement of the spent fuel shipping cask. Spent fuel is moved by the monorail hoist from the spent fuel rack to the shipping cask. The loaded spent fuel shipping cask is moved from the end of the pool to the decontamination area and thence to a flatbed trailer by use of the overhead bridge crane. The spent fuel pool overhead bridge crane is equipped with mechanical stops to prevent crane movement over the spent fuel pool area.

An assumed fuel cask drop by the overhead bridge crane into the spent fuel cask loading area was analyzed for the worst drop condition. The worst drop condition in terms of pool structural damage is a drop in a perfectly vertical position starting from an elevation of five feet above the pool surface, or 43 feet above the pool bottom.

The cask velocity on striking the one-inch cask wear plate on the pool bottom is 40 ft/sec. This wear plate covers the 1/4 inch thick pit liner. Liner penetration would occur and the concrete beneath the liner would crack from this cask drop. Water would be expected to slowly flow through the punctured liner and fill the cracks in the concrete. The pit is 24 feet below surrounding grade and is founded on solid rock which limits the leakage rate through the punctured liner. The makeup water capacity is expected to meet any leakage which might occur. Since there is no other equipment on the pool bottom, damage by the dropped cask would be limited to the liner, the concrete below the liner, and the wear plate.

The applicant has provided guide rolls on the manipulator crane and trolley to prevent horizontal movement. Anti-rotation bars prevent each wheel from lifting from the rail.

Mechanical stops on the overhead bridge crane, which can only be removed by administrative control, assure that movement of the spent fuel cask by the fuel storage building crane is confined to certain areas, thereby avoiding travel over the spent fuel storage area.

As required in the Technical Specifications, test loads and functional checkouts of all of the cranes will be made throughout the life of the plant. In addition, the applicant has stated that the crane operator will be certified in accordance with Chapters 2 and 3, Operation, Overhead and Gantry Cranes, USAS B30.2.0 - 1967.

On the basis of our review of the various handling systems, we conclude that they are acceptable. This conclusion is based on the following:

(1) The right tool is assigned to the right job.

(2) Both mechanical stops and administrative procedures will prevent heavy masses from being carried over the spent fuel.

(3) Various mechanical devices have been installed to minimize the likelihood of the manipulator crane falling into the pool.

(4) Conservative analyses indicate that the consequences of dropping heavy objects within the containment and within the fuel storage building will not compromise safety.

At the present time Indian Point 3 has a 40 ton capacity, Category III, overhead crane. We have been advised by the applicant that it may purchase an overhead crane with an approximate load carrying capability of 70 tons and may also purchase a heavier fuel cask. Should a new crane be purchased, the staff will review the necessity of having the crane and its support structures built to Category I criteria.

9.2 Water Systems

9.2.1 Station Service Water System

The station service water system is a Category I design composed of two independent headers, whose pumps can be powered from the diesel generators. The two headers operate on a split system, one termed nuclear because it supplies the essential nuclear components, and the other termed conventional because it supplies less essential components. One of the three nuclear service water pumps and two of the three conventional service water pumps are operating during normal conditions. By manual valve operation, essential loads can all be carried by the nuclear header, or all can be transferred and carried by the conventional header.

The nuclear header loads are:

(1) The containment fan cooler units and motor coolers.

(2) The diesel generator water and lube oil jacket coolers.

(3) The instrument air compressor cooling system.

(4) The nuclear service water pump strainer blowdown.

(5) The turbine oil coolers.

(6) The generator hydrogen seal oil coolers.

(7) The boiler feed pump oil cooler.

(8) The radiation sample mixing nozzle.

The conventional header loads are the component cooling heat exchangers and the conventional service water pump strainer blowdown as well as other plant services. The component cooling heat exchangers and strainer blowdown services are considered less essential loads on the system only in the sense that cooling water to the component cooling heat exchangers is not required during the injection phase of a loss-of-coolant accident. Because of the heat capacity of the water in the component cooling system, the temperature rise rate of this system without the use of the component cooling heat exchanger is about 5°F per hour. Consequently the water temperature would only increase by a few degrees before the recirculation mode is initiated. We find this acceptable because the peak component cooling system water temperature would be significantly below any system temperature limits.

The service water pumps are located in a Category I designed intake structure and can take suction from any of four separate intakes, any one of which is capable of supplying the service water emergency requirements. A debris wall is provided, a coarse screen, and finally a fine traveling bank screen. For winter operation, warm water is circulated ahead of the coarse screen and electric heaters are provided to the driving head of the traveling screen to prevent icing of screen panels. Water is supplied from the Hudson River.

9.2.2 Cooling System for Reactor Auxiliaries

The component cooling system is a closed loop designed to:

(1) Remove residual and sensible heat from the reactor coolant system via the residual heat removal loop following a loss-of-coolant accident, and also during plant shutdown.

(2) Cool the letdown flow to the chemical and volume control system during power operation.

(3) Provide cooling to dissipate waste heat from various primary plant components.

The component cooling system is a Category I design. During normal operation, two component cooling pumps and one component cooling heat exchanger provide sufficient heat removal. A backup pump is provided which provides 50 percent flow capacity and a redundant heat exchanger provides a 100 percent backup service. All three pumps and both heat exchangers are utilized to remove residual and sensible heat during plant shutdown. In the event of failure of a pump or heat exchanger, safe shutdown is not affected, but the cooldown period is extended.

We conclude the system design is adequate for long-term accident cooling and is acceptable.

9.2.3 Condensate Storage Facilities

The single condensate water storage facility is a 600,000 gallon water tank built to Category I design. The tank is connected to a diffusing pipe inside the condenser shell for makeup purposes on low water level signal. An isolating signal will secure the storage tank from the condenser when the tank level reaches 360,000 gallons. This ensures a condensate reserve for 24 hours of operation of the auxiliary feedwater pumps in order to maintain hot shutdown conditions following a turbine trip at full power. The storage tank and piping system to the auxiliary feedwater pumps is a Category I design and similar in capacity to those used in other PWR type reactor plants. We conclude that the design of the condensate storage facility is acceptable.

9.3 Process Auxiliaries

9.3.1 Compressed Air System

Instrument air and station service compressed air operate as two separate systems. The capability does exist, however, for the service air system to back-up the instrument air system. The instrument air system is equipped with refrigerant dryers and dessicant dryers to maintain instrument quality conditions, and reduce the air dewpoint compatible with the lowest expected outdoor temperature. In the event of service air introduction into the instrument air system, the air passes through two liquid oil prefilters, and two oil vapor prefilters. Components essential to plant safety, which are serviced by the instrument air systems, are provided with back-up dry nitrogen cylinders to assure safe shutdown action of the component in the event of failure of the instrument air systems. The components having dry nitrogen cylinders are the auxiliary boiler feed pump control valve, the steam dump valves to atmosphere, the service water supply valves to the conventional plant, the containment building penetration and weld channel pressurization system.

We have concluded that the instrument and service air systems are acceptable on the basis that a backup system is provided in the event of failure of the instrument air system.

9.3.2 Process Sampling System

The process sampling system provides liquid samples for both chemical and radiochemical analyses. Basically, the sample lines originate from two sources. One source is inside the containment and consists of high temperature and high pressure lines that come from the pressurizer, the reactor coolant system, and the steam generator blowdown lines. The other source is outside containment, and consists of low temperature and low pressure lines which come from the auxiliary coolant system and the chemical and volume control system.

The sample lines inside containment are all isolated by manual valves and a second air-operated fail-closed valve. Only the sample line from the recirculating pump discharge is equipped with remote manual valves inside containment followed by two manual valves outside the containment. This provision enables sampling following a loss-of-coolant accident and loss of service air.

We conclude the system design is acceptable.

9.3.3 <u>Chemical, Volume Control, and Liquid Poison Systems</u>

The chemical and volume control system is designed to:

(1) Adjust the concentration of chemical neutron absorber for reactivity control.

(2) Maintain a proper water inventory in the reactor coolant system.

(3) Provide seal water for the reactor coolant pump shaft seals.

(4) Process coolant effluent for reuse of boric acid and reactor make-up water.

(5) Maintain a proper concentration of corrosion inhibiting chemicals in the coolant.

(6) Maintain coolant and corrosion activities within design levels.

The system is also used to fill and hydrotest the reactor coolant system.

The system consists of letdown coolers, flow controls, boron meter, purification demineralizer prefilter, purification demineralizer, purification filters, charging pumps, reactor coolant pump seal coolers, and a volume control tank. We have reviewed the system to assure that redundant components and alternate flow paths exist in order to permit equipment maintenance and assure operability.

We have verified that any charging pump and boric acid transfer pump can be operated from the onsite diesel generator power on loss of offsite a-c power. The system is capable of making the core subcritical with no rods inserted in less than sixteen minutes.

The chemical and volume control system is similar to systems used in previously licensed reactor plants of this type. We conclude the system design is acceptable.

9.3.4 Gross Failed Fuel Detection System

A Gross Failed Fuel Detector (GFFD) has been installed on the hot leg of one of the reactor coolant loops. This system is similar to those installed in other PWR's, including Indian Point 2, and is described in Section 11.2 of the FSAR.

The GFFD samples primary coolant activity from one of the hot legs. Whenever the coolant activity exceeds a preset value by 20,000 counts per minute, an alarm will go off alerting the operator to a significant increase in primary coolant activity. The set point value is determined from the frequent radiochemical analyses made of the primary coolant. The set point value will however always correspond to coolant activity levels below technical specification limits. Should the alarm sound, the operator can rapidly make another radiochemical analysis of the primary coolant. It is estimated that this analysis can be accomplished in one half an hour.

Experience with GFFD systems is limited at this time. Nonetheless the staff concludes that this system has the potential for detecting abrupt gross failures of a fuel element and meets the intent of the recommendations of the January 15, 1969 ACRS letter which called for a means for early detection of abrupt gross fuel failures.

9.4 Air Conditioning, Heating, Cooling, and Ventilation Systems

9.4.1 Control Room

The control room air conditioning, heating, and ventilation system is designed to maintain 75°F dry bulb temperature and 50% relative humidity, and permit cleanup of airborne particulate radioactivity after a LOCA. (See Section 15 of this report for control room doses after a LOCA.)

The control room air conditioning system is a Category I design capable of taking its power from the diesel generator bus.

We have evaluated the system to assure functional capability, especially during a loss-of-offsite power accompanying a loss-of-coolant accident. The system is similar to those of other previously licensed reactor plants of this type. We conclude that the system is acceptable.

9.4.2 Auxiliary Building and Radwaste Area

The primary auxiliary building ventilation system serves to circulate filtered air through various rooms of the building to remove equipment heat, and control the flow of radioactivity from low activity to potentially higher activity areas. Air is exhausted from each of the building compartments through ductwork designed to sweep the room as it travels to the room exhaust register. Air flows to the exhaust plenum and discharges through roughing filters, HEPA and charcoal filters before discharge to the plant vent.

The system is similar to those of prior reactor plants of this type. We conclude the system design is acceptable. (See Section 11 of this report for further information.)

9.4.3 Fuel Storage Building

The fuel storage building ventilation system is supplied from louvered ceiling tempering fan units which are interlocked with the exhaust fans for quick closure in the event high radiation levels are detected in the building. The exhaust system draws air from the pool surface and ceiling areas which exhaust to a plenum equipped with a roughing filter, HEPA and charcoal filter, before discharge to the plant vent. Within the plant vent is a 50,000 cubic feet per minute dilution fan actuated by a high radiation alarm. This fan exhausts from the auxiliary building, radwaste area, and fuel storage building.

The system is similar to those of prior plants of this type. We conclude the system design is acceptable. (See Section 11 of this report for further information.)

9.5 Other Auxiliary Systems

9.5.1 Fire Protection Systems

There are three basic fire fighting systems for Indian Point 3: water, carbon dioxide and foam-water. The water supply is from a 1,500,000 gallon onsite storage tank whose source is the Catskill water supply. The water system is an extension of the Indian Point 1 system for yard hydrant protection. Portions of the fire system within Indian Point 3 are designed to Category I criteria. These areas are the diesel generator building, electrical tunnel from the control building to the containment building and the primary auxiliary building. The fire protection system is designed to applicable portions of the Nuclear Energy Property Insurance Association and the National Underwriters Codes for Standards.

The water system is connected as a loop system to permit water flow in either direction. Hose reels are located in the turbine building, and temperature controlled deluge watersprays are located at the main transformers and the station auxiliary transformer.

The foam-water system is a separate electric thermostat actuated deluge system serving the hydrogen seal oil unit, boiler feed pump oil console, lube oil storage tank, and lube oil reservoir.

Portable carbon dioxide extinguishers serve the diesel generator rooms, backing up the water system, and also serve the primary auxiliary building, turbine hall, control building, fan house, electrical tunnel, fuel storage building, waste holdup tank pit, auxiliary feed pump building, containment building and electrical penetration tunnel.

We have concluded the fire protection system design is acceptable.

9.5.2 Communication Systems

The intra-plant communication systems are the page-party public address system and the Bell telephone system. The page-party system is powered from a motor control center which can be connected to the diesel generator bus. The system can be merged with the Indian Point 1 and Indian Point 2 systems. Page channels and party channels are controlled from the control room which has the capability to transmit page and party conversations through loudspeakers located throughout the plant and site. In the primary, or nuclear area, handset stations allow usage of two party-line channels for conversing with the control room.

We have concluded that the communications system is acceptable.

9.5.3 Diesel Generator Fuel Oil Storage and Transfer System

The three onsite diesel generator sets are separated, independent of function, and each has its own 7700 gallon underground fuel storage tank. Each tank is equipped with a vertical fuel oil transfer pump which, through one of two redundant headers, automatically and independently fills the day tank for the diesel it serves following a start signal from the day tank. Manual header valving allows any transfer pump to supply fuel to the day tanks of all three diesel generators. Each storage tank has an alarm level to alert the operator to refill the tank from an outside source. The entire transfer system is designed to seismic Category I requirements.

To comply with our requirements, the applicant has placed all three transfer pumps on emergency power supply buses, rather than only two pumps which had been proposed earlier. With respect to the latter arrangement, in the event of loss of off-site power, only two transfer pumps would have been connected to an essential power supply. With all three transfer pumps powered from essential buses, approximately 93 hours of diesel fuel is available. In the event of failure of a single transfer pump, up to 62 hours operation is available.

When the diesel fuel in the 7700 gallon storage tanks is exhausted additional supplies can be obtained both on site and immediately adjacent to the site. Two 30,000 gallon tanks on site and one 200,000 gallon tank at the Consolidated Edison Buchanan site store fuel oil that is compatible with the diesel generators. The Technical Specifications require that the oil stored in these tanks be compatible with the diesels and that at least seven days of fuel supply for Indian Point 3 be available. Since these large storage tanks are not directly piped into the 7700 gallon underground fuel storage tanks, provisions have been made to transfer the oil in the larger tanks to the underground tanks. The applicant has a contract with a local company to supply an oil truck, on a priority basis, to effect this transfer if necessary. Oil transfer hoses with the appropriate fittings are installed near the outlets of these large storage tanks to facilitate this transfer. Adequate space is available around the storage facilities to place an oil truck there while it is being filled.

Based on the above considerations, we have concluded that the diesel oil storage capacity needed for Indian Point 3 is acceptable.

9.5.4 Diesel Generator Cooling Water System

The service water supply to the diesel generator lube oil and jacket water coolers is shown in Figure 9.6-1 of the FSAR. Cooling water flow to the diesels is required when the plant is on emergency power. This cooling is normally accomplished through the nuclear service water system, with the conventional service water system acting as a backup. Should the ten inch line in the nuclear service water system break or should the ten inch valve in this line inadvertently close, then all three diesels would eventually be inadequately cooled. The applicant has estimated that in approximately one hour the diesels would overheat and fail unless adequate cooling was restored. The bases for this estimate will be incorporated into the FSAR and reviewed by the staff.

Diesel failure can be prevented by switching over from the nuclear service water system to the conventional service water system. The operator is alerted to inadequate diesel cooling by an alarm in the control room. This alarm sounds when the flowmeter in the common discharge line of these three diesels measures less than 1000 gpm. The operator would then manually valve off the appropriate sections of the nuclear service water system and valve on the backup conventional service water system. The valves that must be opened or closed to affect this switchover are readily accessible, near the control room, and of four to ten inches in diameter. Depending on the break location, between two and seven valves must be repositioned. The conventional service water system has adequate capacity to supply the water valved off from the nuclear service water system.

Final acceptance of this method of coping with this type of failure in the nuclear header depends upon justification of the one hour estimate during which the diesels can supply the necessary emergency power without degradation. We have informed the applicant that suitable modifications will be made to the service water system if this estimate cannot be supported. Resolution of this matter will be the subject of a supplement to this Safety Evaluation.

9.5.5 Diesel Generator Starting System

Each diesel generator is automatically started by two redundant air motors, each air motor served from a common storage tank and compressor system. The piping and electrical service is arranged so manual transfer between diesel units of starting air is possible. Each air storage tank has sufficient air for four starts. Since this is consistent with previously approved systems, we have concluded that the diesel generator starting system for Indian Point 3 is acceptable.

10.0 STEAM AND POWER CONVERSION SYSTEM

10.1 Summary Description

The steam and power conversion system is of conventional design, similar to the designs used in previously approved plants. The system will remove the heat energy from the reactor coolant in four steam generators and convert it to electrical energy in the turbine driven generator. The condenser will transfer unusable heat in the cycle to the condenser cooling water. Upon loss of full load, the system will dissipate the energy in the reactor coolant through by-pass valves to the condenser and through the power operated valves to the atmosphere.

Steam generated on the secondary side of the steam generators will sequentially pass through the double flow high pressure turbine, moisture separators and reheaters, three double flow low pressure turbines and to three single pass, divided water box type condensers.

The condensate and feedwater system will return the condensate to the steam generators after passing it through five stages of feed heating.

10.2 Turbine Generator

The turbine is a four element, tandem-compound, six-flow exhaust type, 1800 RPM unit. It has a warranted rating of 1,021,793 kWe gross and a generator rating of 1,125,600 kva. The generator is direct coupled and hydrogen cooled. The turbine is similar to turbines in previously approved plants.

10.3 High Energy Line Rupture Outside Containment

In December 1972, the applicant was asked by the staff to assess the consequences of postulated pipe failures outside of containment including failure of the main steam and feedwater lines. The applicant has conducted its assessment for Indian Point 3 utilizing criteria and guidelines provided by the staff. The basic criteria require that:

1. Protection be provided for equipment necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming a concurrent and unrelated single active failure of protection equipment, from all effects resulting from ruptures in pipes carrying high-energy fluid, up to and including a double-ended rupture of such pipes, where the temperature and pressure conditions of the fluid exceed 200°F and 275 psig. Breaks should be assumed to occur in those locations specified in the staff pipe whip criteria. The rupture effects on equipment to be considered include pipe whip, structural (including the effects of jet impingement) and environmental.
2. Protection be provided for equipment necessary to shut down the reactor and maintain it in a safe shutdown condition, assuming a concurrent and unrelated single active failure of protection

equipment, from the environmental and structural effects (including the effects of jet impingement) resulting from a single open crack at the most adverse location in pipes carrying high-energy fluid routed in the vicinity of this equipment, where the temperature and pressure conditions of the fluid exceed 200°F and 275 psig. The size of the cracks should be assumed to be 1/2 the pipe diameter in length and 1/2 the wall thickness in width.

The applicant responded to this by meeting with the staff and by submitting reports on May 14, 1973 and June 8, 1973, describing its findings and the resultant plant modifications.

It is convenient to divide the applicant's responses into two piping groups. The first group includes large pipes such as main steam lines, feedwater lines, and auxiliary feedwater lines. These large pipes often require massive pipe restraints to prevent significant damage to structures and nearby pipes and valves should these pipes fail. They are characterized by having high mass and energy effluxes, if broken, which could produce significant pressure and temperature increases within the structures that surround them. The second group of pipes includes smaller sized high-energy lines such as steam generator blowdown lines, letdown lines, charging lines, sample lines, auxiliary steam lines and nitrogen lines. Some of these lines do require pipe restraints to limit their motion in the event of a postulated break, but in general their greatest damage potential lies in affecting cable trays and electrical equipment.

With regard to the group of larger high-energy lines, the staff reviewed their piping layout. All the steam and feedwater lines run directly from the containment to the turbine building passing through only one significant intermediate enclosure. The control room, emergency diesel generators, and the primary auxiliary building which houses most of the engineered safety features are all separated from the steam feedwater and auxiliary feedwater lines by a considerable distance and would not be affected by any rupture of one of these larger lines.

Between the turbine building and the containment is the pipe bridge and the auxiliary feed pump (AFP) building. Pipe ruptures within the pipe bridge area will not prevent the safe shutdown of the plant. The AFP building is shown in Figures 10.1 and 10.2 and also in Figure 6.4. The lower two rooms of the AFP building are concrete enclosures with thick walls and roofs, while the upper portions of the AFP building are made of light weatherproofing material. Pressure transients in the lower concrete rooms result in peak pressures significantly below the pressure retaining capability of these rooms. Should a pipe rupture in the upper portion of the AFP building, the light weatherproofing material would be blown off at pressures well below the structural capabilities of this area. The staff has made its own independent analyses of these pressure transients and has concluded that no pressure transients within the AFP building due to high energy line breaks would result in overstressing any portion of this structure or prevent the safe shutdown of the plant.

The possible loss of essential equipment, jet impingement effects, and pipe whip effects were also reviewed. The staff has concluded that no high energy line break within the AFP building would cause the loss of equipment in a manner to prevent safe shutdown and that the effects of jet impingement are negligible.

Pipe whip effects were also reviewed using the break location criteria specified by the AEC December 1972 letter. Thirty-two different postulated break locations were identified. Pipe whip effects were first calculated using a very conservative static loading method. A dynamic loading analysis was then made on main steam line 24 at the inlet to the first elbow outside the containment. This location was selected as the one which most clearly shows the response of the restraint systems. The results of the more precise dynamic analysis method showed that the static analysis method was considerably more conservative. Both the static and dynamic methods showed that the Indian Point 3 pipe restraints were adequately designed to prevent pipe whip. The staff reviewed the dynamic analysis method used by the applicant and found it acceptable. The staff also determined that the applicant had properly applied the break location criteria given in the AEC's December 1972 letter.

Some modifications to the AFP building were made as a consequence of this review.

The applicant calculated that the pressure and temperature in the concrete room that houses the auxiliary feedwater pumps would be

0.9 psig and 213°F, respectively, if there were a break in the steam supply to the steam turbine auxiliary feedwater pump. Since the electric driven pumps have never been tested in this kind of an environment, precautions were taken to prevent breaks in this turbine supply line from possibly affecting the electric motor driven auxiliary feedwater pumps. Two independent sensors will initiate action which will automatically shut valves in the steam turbine supply line if a high temperature occurs in the pump room. The staff has concluded that this modification is acceptable. In order to prevent flooding of the pump room by a broken auxiliary feedwater line within the pump room, check valves were installed in all the auxiliary feedwater lines. These check valves are located outside of the pump room and prevent backflow from the main feedwater system should an auxiliary feedwater line fail. The staff accepts this modification.

Another modification was the installation of three foot-long steel beams (16WF71) under each of the feedwater lines in the upper concrete room. The purpose of these beams is to prevent any broken feedwater line from impacting on the roof of the pump room with possible concrete spalling below. Analysis indicates that the calculated shear in the concrete roof of the pump room would be below the allowable shear, should a broken feedwater line strike the three-foot steel beam. Installation of these steel sections therefore prevents spalling of concrete onto the pumps and is acceptable.

In May 1973, the applicant committed to placing pipe whip restraints on the main feed lines in the upper concrete room. The purpose of these restraints is to provide additional protection for the auxiliary feedwater lines that are routed through this room. The staff finds this acceptable.

In summary,

(1) Breaks in a steam line or feedwater line outside of the auxiliary feed pump (AFP) building will not prevent safe shutdown.

(2) Breaks in high energy lines within the AFP building will not cause the loss of essential equipment and will not over-pressurize any section of the AFP building.

(3) Jet impingement effects have been found to be negligible and pipe whip restraints are adequate to prevent one broken high energy line from rupturing another.

(4) Design modifications have been made to prevent (a) flooding in the pump room, (b) concrete spalling, (c) interactions between a failed steam supply of the turbine AFP and the electric driven auxiliary feed pumps, and (d) loss of the auxiliary feedwater lines due to pipe whip of a feedwater line.

The possible effects of pipe whip, impingement, or high pressure and temperatures resulting from a postulated failure of any of the smaller high energy lines were investigated for the Control Building, the Diesel Generator Building, the Fuel Storage Building, the Turbine Building, and the Primary Auxiliary Building (PAB). Only the PAB required design modifications. The Control Room has essentially no high energy lines and the Diesel Generator Building's only high energy lines are the starting air lines whose failure would not damage Class I equipment. The Fuel Storage Building contains some auxiliary steam lines whose failure would not result in damage to the spent fuel pit. Failures of steam or feedwater lines within the Turbine Building will not prevent safe shutdown and will not cause the destruction of this large, highly ventilated building. Approximately 75 feet separate the nearest high energy line in the Turbine Building and the Control Building. This distance eliminates any concern about pipe whip and jet effects were found to produce negligible loads on the Control Building.

Design modifications that have been required as a result of the review of the smaller high energy lines include the addition of pipe restraints to portions of the steam generator blowdown lines, shielding around cable trays to eliminate jet impingement effects and an alarm system to prevent overheating of the penetration area because of a ruptured letdown line, steam generator blowdown line,

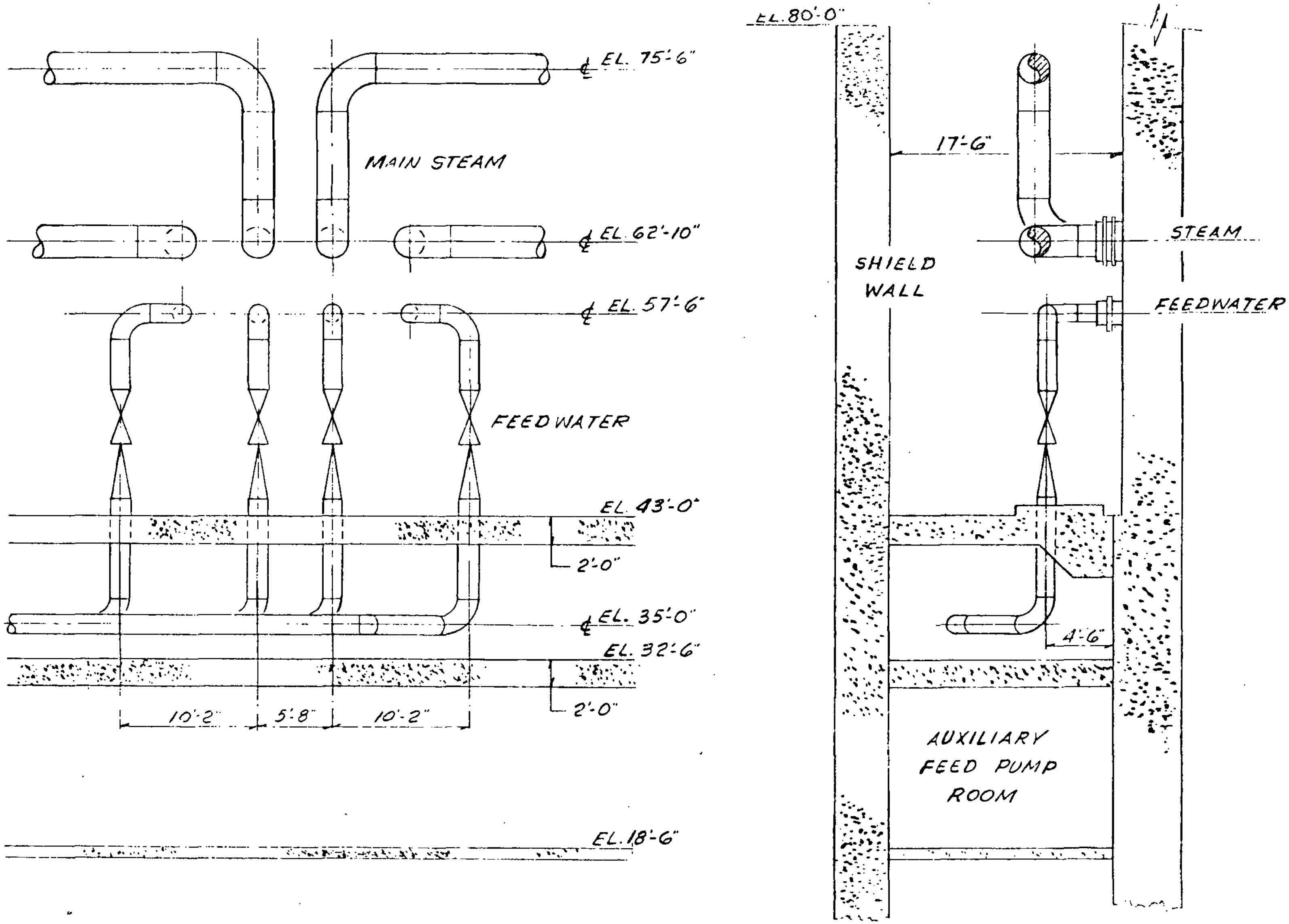
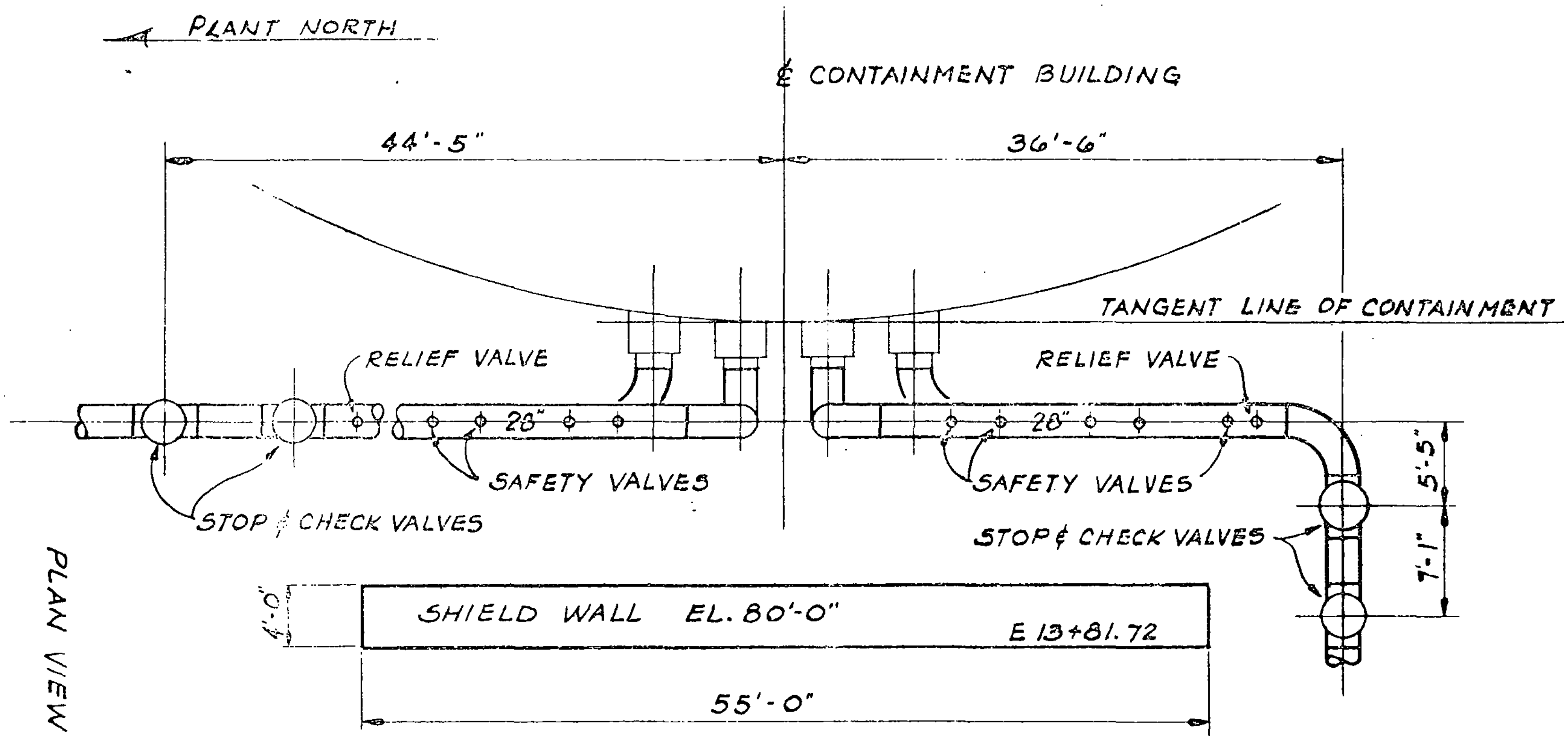


FIGURE 10.1
ELEVATION VIEW
SHIELD WALL REGION



PLAN VIEW OF SHIELD WALL AREA
FIGURE 10.2

sample line, or auxiliary steam line. The staff has reviewed these design modifications and finds them acceptable.

10.4 Other Features of Steam and Power Conversion System

Three divided box, single pass condensers will maintain turbine back pressure for all normal operating conditions including the period of time when the turbine steam bypass valves are in operation. The hot well capacity will provide a 5.5-minute holdup time for the condensate when operating at rated load.

Normal water level in the condenser hot well is maintained by the condensate makeup and surge systems. The makeup system connects the condenser to the Category I 600,000 gallon condensate storage tank. Automatic valves operate to maintain condenser water level. Should the amount of water within the condensate storage tank decrease to 360,000 gallons, the condensate storage tank will be automatically isolated. This 360,000 gallons of water ensures a 24-hour reserve for the auxiliary feedwater pumps to hold the plant at hot shutdown following a trip from full power.

We have concluded that the designs of the main condensers and condensate storage tank are acceptable.

One four-element, two-stage air ejector with separate inter-condenser and common aftercondenser is provided to withdraw non-condensible gases from and maintain a vacuum on each condenser. The ejectors use main steam, reduced in pressure by a regulating valve. In the event the air ejector exhaust radiation monitors reveal a high activity, the non-condensible exhaust gases will not be vented to the atmosphere but routed to the containment for ultimate passage through roughing filters, HEPA filters, and charcoal filters. Because of this feature, we have concluded that the main condenser evacuation system is acceptable.

To prevent overpressure in the steam generators on a turbine trip with reactor trip, without main steam safety valve operation, twelve turbine steam bypass valves open to discharge steam to the main condenser.

During startup, hot standby service and physics testing, the same steam bypass system can be manually actuated from the pressure controller to effect a simulated load on the reactor plant.

We have concluded that protection against system overpressure is acceptable.

The circulating water system is composed of six circulating water pumps, each providing 140,000 gpm. They are each in an individual pump well, thus tying a section of the condenser to an individual pump. Upon discharge from the condenser, the combined pump flow is directed to a canal.

The condensate and feedwater system supplies 13,823,282 pounds of feedwater per hour to four steam generators at a turbine load of

1022 MW(e). Three one-third capacity condensate pumps take suction from condenser hot wells.

Two one-half size feed pumps take suction from the condensate delivered from three stages of feed heating, and deliver it through one more stage of feed heating and feedwater regulating valves to the steam generators.

Each steam generator has two bottom blowdown connections for shell solids concentration control. Each blowdown line has a manual shutoff valve and two diaphragm operated trip valves. Blowdown discharges through a throttle valve to a flash tank where the water is cooled prior to discharge to the circulating water discharge canal., or through the liquid waste treatment system if radiation levels are high in the blowdown. (See Section 11 of this Safety Evaluation for further discussion.)

We have concluded that the condensate, feedwater, circulating water, and steam bypass atmospheric relief, and steam generator blowdown systems are acceptable.

11.0 Radioactive Waste Management

11.1 Design Objective and Criteria

The radioactive waste management systems for Indian Point 3 are designed to provide for the controlled handling and treatment of radioactive liquid, gaseous, and solid wastes. The applicant's design objective for these systems is to restrict the amount of radioactivity released from normal plant operation to unrestricted areas to within the limits set forth in 10 CFR Part 20.

The Technical Specifications issued as part of the operating license require the applicant to maintain and use existing plant equipment to achieve the lowest practicable releases of radioactive materials to the environment in accordance with the requirements of 10 CFR Part 20 and 10 CFR Part 50. The applicant is also required to maintain radiation exposures to in-plant personnel and the general public "as low as practicable" in conformance with the requirements of 10 CFR Part 20.

Our evaluation of the design and expected performance of the waste management systems for Indian Point 3 is based on the following design objectives:

Liquids

(1) Provision to treat liquid radioactive waste to control the expected releases of radioactive materials in liquid

effluents to the environment to less than 5 Ci/yr/unit, excluding tritium and dissolved noble gases.

(2) The calculated annual average radiation exposure to the whole body or any organ of an individual at or beyond the site boundary not to exceed 5 mrem.

(3) Concentration of radioactive materials in liquid effluents prior to dilution in the environment not to exceed the limits in 10 CFR Part 20, Appendix B, Table II, Column 2.

Gaseous

(1) Provisions to treat gaseous radioactive waste to limit the expected release of radioactive material in gaseous effluent from principal release points so that the annual average radiation exposure to the whole body or any organ of an individual at or beyond the site boundary not to exceed 5 mrem.

(2) Provisions to treat radioiodine released in gaseous effluent from principal release points so that the annual average thryoid dose to a child through the pasture-cow-milk pathway be less than 15 mrem. For Indian Point 3 the estimated thyroid dose is evaluated at the location of the nearest actual cow, approximately seven miles south of the site.

Solid

(1) Provisions to solidify all liquid waste from normal operation including anticipated operational occurrences prior to shipment to a licensed burial ground.

(2) Containers and method of packaging to meet the requirements of 10 CFR Part 71 and applicable Department of Transportation regulations.

The following sections present our evaluation of the liquid, gaseous, and solid waste treatment systems, the design codes and quality assurance criteria, and the radiation monitoring of process effluents and of in-plant areas. Our evaluation also considered radioactive effluent releases for combined operation of Indian Point 1, 2, and 3. Each unit is provided with separate waste treatment systems except for the steam generator blowdown and laundry treatment systems located at Indian Point 1 which are shared by Units 1, 2, and 3.

11.2 Liquid Waste

The liquid waste treatment system is divided into three main systems:

(1) The reactor coolant treatment system, which includes the chemical and volume control system and the boron recycle system.

(2) The liquid waste disposal system.

(3) The steam generator blowdown treatment system.

These systems serve only Indian Point 3. When the steam generator blowdown contains radioactivity above a predetermined value, it will be processed at Indian Point 1 along with the blowdown from Indian Point 1 and 2. The laundry and hot shower wastes are also processed at Indian Point 1. The collection rates and system capacities are presented in Table 11-1. The liquid effluents will be continously monitored before discharging through the circulating water duct to the Hudson River. If the radioactivity exceeds a predetermined value, the discharge will be automatically stopped by a valve on the discharge line.

11.2.1 Reactor Coolant Treatment System

The reactor coolant treatment system will collect and process deaerated liquids from shim bleed, equipment leaks and excess let-down flows. During normal operation the reactor coolant will be let down continuously and processed at a nominal rate of 75 gpm in the chemical and volume control systems (CVCS) to maintain coolant quality. This letdown stream will be processed through redundant deep-mixed-bed demineralizers to remove corrosion and fission products and returned to the reactor coolant system. Part of this stream, the shim bleed, will be processed through the boron recycle system. The excess letdown and the containment equipment leaks will also be processed through the boron recycle system. These streams will be collected in the reactor coolant drain tank and the CVCS holdup tank. They will be batch processed through redundant cation demineralization, gas stripping, and evaporation equipment. The evaporator condensate will be processed through an anion demineralizer to principally remove iodine and routed to one of two monitor tanks for sampling and analysis. Condensate will either be sent to the primary water tank for reuse in the reactor or released to the environment. The condensate can also be processed in the liquid waste disposal system. The boric acid concentrate from the evaporator will be filtered and then collected in the concentrate holding tank for sampling and analysis. The concentrate will either be sent to the boric acid tanks for reuse, or sent to the solid waste system for offsite disposal.

In our evaluation we estimated that approximately 15,000 gallons per day of shim bleed, excess letdown and equipment leaks will be collected. These wastes will be processed through the boric acid demineralizers and evaporators and we estimate a release of 0.035 Ci/yr of radioactivity, excluding tritium and dissolved gases. The applicant did not estimate the radioactivity released from this source. The processing capacity will be 43,000 gallons per day when using both evaporators. Our estimate assumed one-day holdup for decay and 10% release of the processed effluent to the environment. The liquid effluent will be continously monitored during its release to the environment.

11.2.2 Liquid Waste Disposal System

The liquid waste disposal system will collect and batch process aerated radioactive liquid wastes from equipment, floor and chemical drains. The system equipment includes collection and monitoring tanks, a filter, and a two-gpm evaporator. These wastes will be collected in the waste holdup and chemical drain tanks, then filtered, and evaporated. The evaporator condensate will be collected in one of two monitor tanks, sampled and analyzed. The condensate that meets specification will be returned to the reactor water storage tank for reuse or discharged to the Hudson River. Condensate not meeting the required quality will be recycled to the waste holdup tank for further treatment. The evaporator concentrate and spent filters will be sent to the solid waste system.

In our evaluation we estimated that approximately 140 gallons per day of equipment drain effluent and 330 gallons per day of floor and chemical drain effluents will be processed by the two-gpm waste evaporator. We assumed one-day holdup for decay, 10% release from equipment drain effluents, and 100% release of the condensate from the floor and chemical drain effluents. Our calculations showed that approximately 2 Ci/yr of radioactivity, excluding tritium and dissolved gases, will be released. The applicant estimated that approximately 2 Ci/yr of radioactivity, excluding tritium, would be released from this system.

11.2.3 Steam Generator Blowdown

The secondary coolant will be blown down from the steam generator at 10 gpm to maintain chemical purity. As shown in Figure 11-1, the blowdown from the Indian Point 3 steam generators can be directed to the treatment system installed at Indian Point 1 or can be directed to the steam generator blowdown flash tank installed at Indian Point 3. The steam generator blowdown flash tank at Indian Point 3 is intended to process only low level activity wastes. Wastes discharged from this tank would enter the environment without treatment. A continous beta-gamma monitor will measure the radioactivity of the secondary coolant that enters the blowdown flash tank. When the radioactivity in the secondary coolant exceeds a predetermined value, the monitor will activate an alarm and automatically close isolation valves on the blowdown and sampling lines. The blowdown stream from Indian Point 3 will then be routed manually to the Indian Point 1 blowdown treatment system. A composite sample of the liquid releases from the blowdown flash tank will be taken daily and analyzed for isotopic composition. The blowdown treatment system at Indian Point 1 is designed to handle blowdown simultaneously from all three units. This treatment system at Indian Point 1 consists of redundant filters and deep-mixed-bed demineralizers, with a total capacity of 132 gpm. Blowdown from Indian Point 3 can be diverted to the Indian Point 1 treatment system independent of any power generation at Indian Point 1. The effluent from the demineralizer will be discharged to the Hudson River. If the radioactivity in the demineralizer effluent exceeds a preset value, it will activate an alarm requiring appropriate action. Composite samples of the demineralizer effluent are taken daily and analyzed for isotopic composition.

Based on our evaluation, approximately 10 gpm blowdown from Indian Point 3 will be processed in the Indian Point 1 treatment system, resulting in an estimated release of 1.7 Ci/yr of radioactive material, excluding tritium and dissolved gases. With a 50-gpm blowdown rate, the applicant estimated the release rate to be 7.5 Ci/yr. The 132 gpm capacity of the Indian Point 1 system will therefore be adequate to process 50 gpm blowdown rates from Indian Point 1, 2 and 3. We conclude that this system has adequate capability and is acceptable.

11.2.4 Liquid Waste Summary

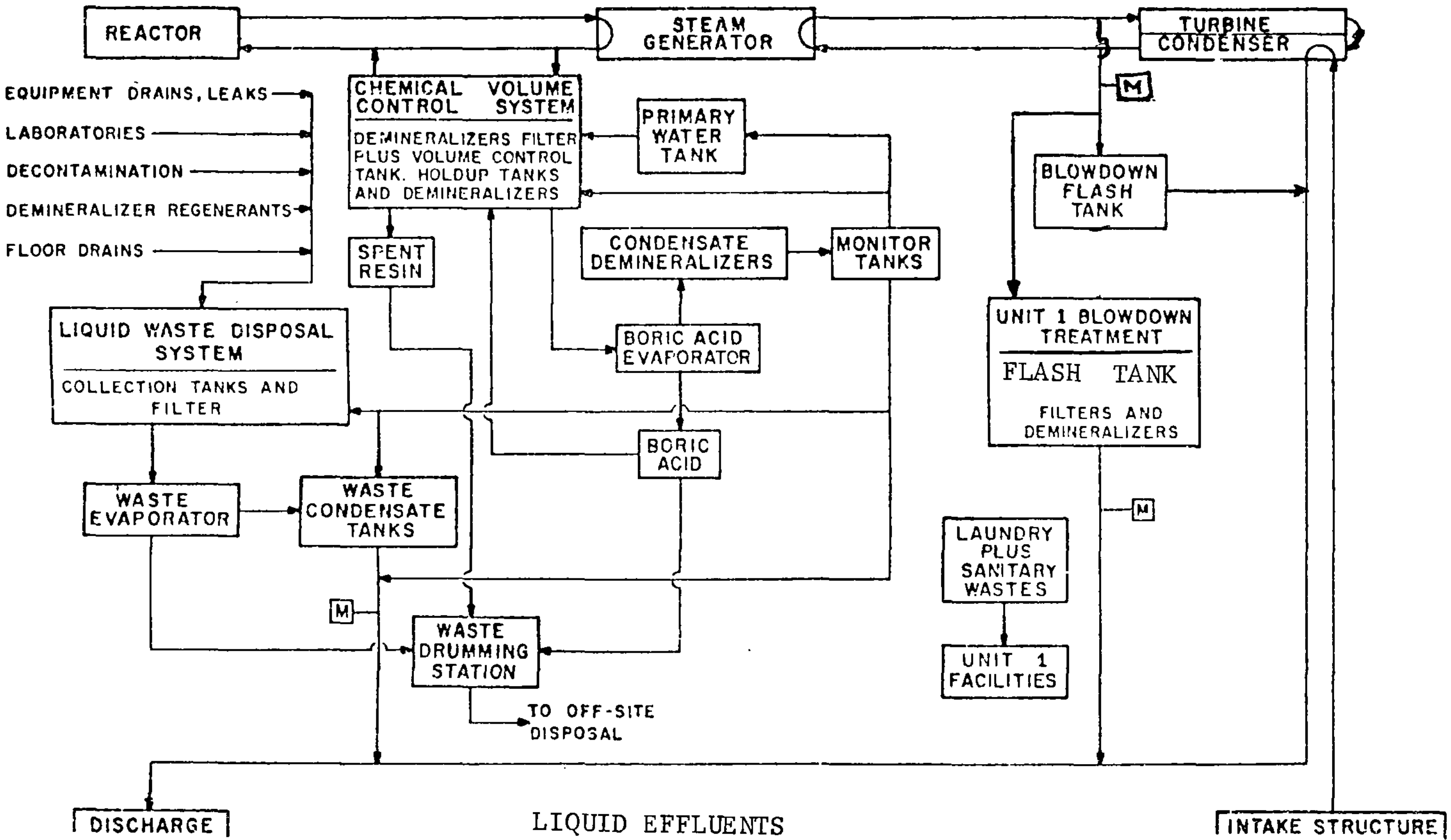
The total radioactivity in the liquid effluent released from Indian Point 3 to the environment was estimated by the applicant to be 9.6 Ci/yr, excluding tritium and 610 Ci/yr of tritium. Based on our evaluation, we calculate an annual release of radioactive material in the liquid effluent will be approximately 3.8 Ci/yr excluding tritium and 350 Ci/yr of tritium.

In our evaluation we calculated the radiation doses to an individual in unrestricted areas from the aquatic food chain and swimming would be less than 5 mrem to the whole body and less than 5 mrem to the thyroid. Our radiation dose calculations considered the combined operations of Units 1, 2 and 3 at the Indian Point site. All radioactive liquid wastes will be continously monitored before discharge. Assuming a circulating water flow rate at Indian Point 1 of 320,000 gpm and at Indian Point 2 and 3 of 870,000 gpm the radioactivity concentration released to the Hudson River will be less than 1% of the limits specified in 10 CFR Part 20.

The liquid waste treatment system has been designed to collect, process, and store waste from operation with the equivalent of 1 percent fission product inventory releases from failed fuel rods to the primary coolant. We have concluded that the liquid waste treatment system will be capable of producing liquid effluents which we consider as low as practicable and therefore is acceptable.

11.3 Gaseous Wastes

The gaseous wastes treatment system for Indian Point 3 include the waste gas processing, the condenser air ejector and the steam generator blowdown vent systems along with the containment purge, and the fuel storage, turbine and auxiliary building ventilation systems. These systems for Indian Point 3 are independent of Indian Point 1 and 2, except for the steam generator



LIQUID EFFLUENTS

FIGURE 11-1

blowdown system. Steam generator blowdown containing radioactivity above a predetermined level will be processed at Indian Point 1. The gaseous releases from all systems will be monitored except ventilation air released from the turbine building. The gases released from the waste gas processing system, the containment purge, the condenser air ejector and the auxiliary building ventilation will be discharged through the plant vent. Ventilation air from the turbine building will be discharged from the turbine building roof.

11.3.1 Waste Gas Processing

The waste gas processing system will collect and treat radioactive gases from the reactor coolant treatment system. These sources include the shim bleed gas stripper, holdup tank cover gases, equipment vents, and gases generated from sampling. The primary source of radioactivity is from degassing the shim bleed in the boron recovery system.

The gas processing system includes redundant compressors and four 525 ft^3 and six 40 ft^3 storage tanks. The waste gases will be pumped to one of the four storage tanks and recycled to the CVCS holdup tanks to provide cover gas during emptying operations. A second tank will be available as backup. When 110 psig pressure is reached in the inservice tank, the feed will be automatically switched to the backup tank. Prior to cold shutdown of the reactor, the reactor coolant will be degassed and the gas will be distributed among the six 40 ft^3 storage tanks.

Some hydrogen is also present in the gas released from the CVCS system. To prevent hydrogen-oxygen explosions, the process equipment vent system operates at positive pressure so as to minimize inleakage of air. In addition, no air or aerated liquids will be present in equipment that vents to this system. The storage tank gas will be automatically sampled and analyzed for hydrogen and oxygen. An alarm will alert the operator when the hydrogen concentration exceeds 2%.

The waste gas storage tanks have sufficient capacity to holdup gases for 45 days for radioactivity decay. Before release to the environemnt, the gas will be sampled and analyzed. During discharge at a controlled rate through the plant vent, the gas will be continuously monitored. Radioactivity releases above a predetermined value will automatically close a valve on the discharge line. Based on a holdup time of 45 days, the applicant estimated releases of 2000 Ci/yr of noble gases. Based on our evaluation assuming 45 days holdup, we calculate an average annual release rate of 1500 Ci/yr of noble gases.

11.3.2 Containment Purge

The containment purge system will process radioactive gases that build up in the containment atmosphere as a result of leaks from the primary system. In our evaluation we considered that the containment atmosphere will be purged four times per year. The equipment used for containment purging includes prefilters, HEPA filters and charcoal adsorbers. The filters and the exhaust fan will be shared with the primary auxiliary building ventilation system. Before purging, we assumed the air in the containment will be recirculated for 16 hours through an internal cleanup system consisting of HEPA filters and charcoal adsorbers. The containment air will then be purged through the HEPA filters, charcoal adsorbers and released through the plant vent. The applicant has estimated the radioactivity released from four purges per year to be 88 Ci/yr of noble gases and 0.00014 Ci/yr of iodine-131.

Based on our evaluation, assuming four purges/yr and 16 hours internal recirculation before purging, we calculate a release of 88 Ci/yr of noble gases and 0.026 Ci/yr of iodine-131. This shared system is acceptable since, during normal operations the exhaust fans provide a negative pressure in the exhaust plenum. This will prevent the cross flow between the containment and the primary auxiliary building. If the exhaust fan fails, the associated supply fan will automatically be shut down to prevent cross ventilation flow between these buildings.

11.3.3 Condenser Air Ejector

Gaseous radioactivity, along with noncondensable gases in the secondary coolant, will be removed from the turbine condenser by the air ejectors. Leakage in the steam generator from the primary to the secondary system will result in some radioactivity in the secondary system coolant. The gases from the condenser will pass through the steam jet ejectors, will be monitored, and then be released through the plant vent. The applicant has calculated that the activity released from the condenser air ejector will be 1300 Ci/yr of noble gases and 0.065 Ci/yr of iodine-131.

Based on our evaluation the radioactivity release will be 580 Ci/yr of noble gases and 0.13 Ci/yr of iodine-131 from this source.

11.3.4 Steam Generator Blowdown

At Indian Point 3 the steam generator blowdown will go to Indian Point 3 flash tank at a rate of 10 gpm. From the flash tank the steam vapor will be released without monitoring from a rooftop vent. When the radioactivity in the secondary coolant is above a predetermined value the blowdown will be automatically stopped and manually diverted to the blowdown flash tank at Indian Point 1. The blowdown system at Indian Point 1 will also receive the blowdowns from Indian Point 1 and 2. The vent from Indian Point 1 flash tank will be vented to the Indian Point 1 turbine condenser. The radioactivity released from Indian Point 1 condenser will be monitored and discharged through the Indian Point 1 stack. When the Indian Point 1 condenser is shut down the vapor will be released from the Indian Point 1 flash tank through an unmonitored rooftop vent. The applicant considered 6 weeks/year for this direct release and estimated a release of 0.13 Ci/yr of iodine-131 from this source.

Based on the past operating experience of Indian Point 1, we estimated that the blowdown vapor from Indian Point 3 would be directly released to the atmosphere for approximately 17 weeks per year. We calculated that this would release 0.16 Ci/yr of iodine-131. The applicant has been advised that capability for continuous monitoring of the blowdown effluent is required prior to initial startup of Indian Point 3.

11.3.5 Primary Auxiliary Building Ventilation

The atmosphere in the primary auxiliary building will contain radioactivity from equipment leaks. The ventilation system for this building will include pre-filters, HEPA filters and charcoal adsorbers. The filters and exhaust systems will be shared with the containment purge system. The ventilation system is designed to flow from clean to potentially more contaminated areas. The applicant estimated that the radioactivity released will be approximately 1300 Ci/yr of noble gases and less than 0.001 Ci/yr of iodine-131. Based on our evaluation we estimate 580 Ci/yr of noble gases and 0.05 Ci/yr of iodine-131.

11.3.6 Turbine Building Ventilation

Steam leaks from the secondary coolant system will release some radioactivity into the turbine building atmosphere. This will be discharged without monitoring to the environment through 11 roof-mounted exhaust fans. The applicant has estimated that the radioactivity released from this source will be 0.01 Ci/yr of iodine-131. Based on our evaluation, we estimate a release of approximately 0.04 Ci/yr of iodine-131.

11.3.7 Fuel Storage Building Ventilation

The Fuel Storage Building Ventilation System will include HEPA filters and charcoal adsorbers. Normally exhaust air will be processed through HEPA filters and discharged through the monitored plant vent. However, when the radioactivity is above a predetermined value, the ventilation exhaust air will be automatically diverted through the charcoal adsorbers prior to being released.

The applicant did not estimate the radioactivity release through the ventilation system under normal conditions. In our evaluation we determined that the radioactivity released from this building under normal conditions will be negligible. An analysis of radioactivity releases due to a fuel handling accident is given in Section 15 of this report.

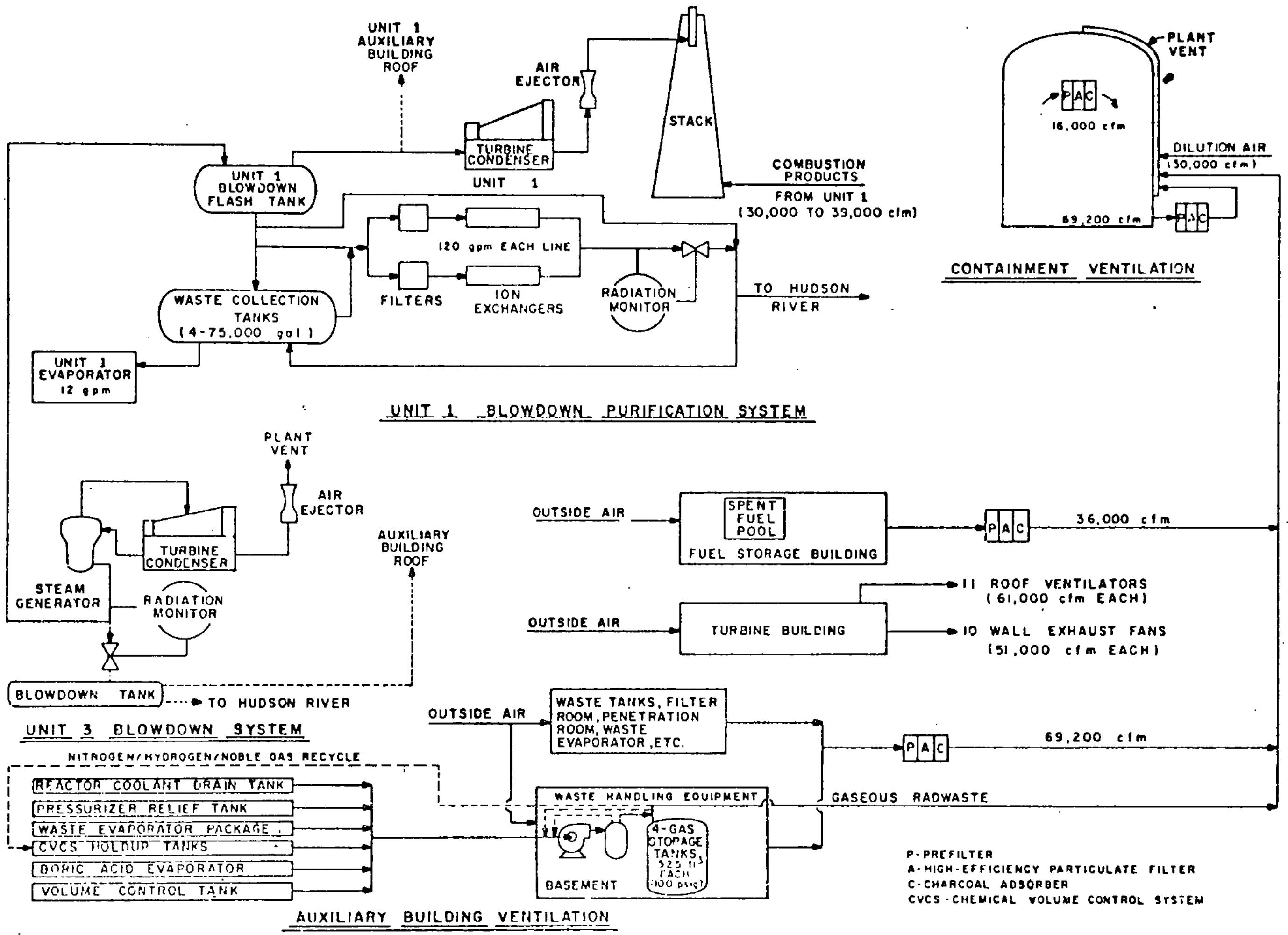
11.3.8 Gaseous Waste Summary

The applicant has estimated the radioactivity in the gaseous effluents released from Indian Point 3 will be 5500 Ci/yr of noble gases and 0.16 Ci/yr of iodine-131. For the combined operation of Indian Point 1, 2 and 3, the applicant's estimated releases are 11,000 Ci/yr of noble gases and 0.32 Ci/yr of iodine-131. The applicant also estimated that radiation doses to an individual at or beyond the site boundary from the combined operation will be 2.4 mrem/yr to the whole body and 1.4 mrem/yr to the thyroid from inhalation.

Based on our evaluation of the gaseous waste systems, we calculated that the radioactivity released from Indian Point 3 during normal operation will be 2700 Ci/yr of noble gases and 0.41 Ci/yr of iodine-131. For the combined operation of Indian Point 1, 2, and 3 we calculated the radioactivity release will be 6600 Ci/yr of noble gases and 0.88 Ci/yr of iodine-131. Based on the combined operation of Indian Point 1, 2 and 3 we calculated the annual average radiation doses at the site boundary will be less than 5 mrem to the whole body and less than 5 mrem to the thyroid from inhalation. We calculated that the radiation dose to a child's thyroid will be less than 15 mrem per year based on the grass-cow-milk pathway for radioiodine for the nearest actual cow, seven miles south of the site. The dose calculations were based on a maximum annual average relative concentration of 2.4×10^{-7} sec/m^3 for Indian Point 2 and 3 and 5.1×10^{-8} sec/m^3 for Indian Point 1. Subject to the installation of continuous monitoring capability prior to initial startup we conclude that the release of radioiodine in gaseous effluents are as low as practicable and are acceptable.

11.4 Solid Wastes

The sources of solid radioactive wastes will include spent demineralizer resins, evaporator concentrates, filters, and miscellaneous items such as contaminated clothing, gloves, shoe covers, glassware and paper. The solid waste disposal system is designed to package all solid wastes in 55-gallon drums. A facility will be provided for loading spent resin and evaporator concentrates. A hydraulic baler will be used for the miscellaneous wastes. The filled drums will be stored in a shielded area of the drumming room. The spent demineralizer resins after approximately six months storage will be slurried into shielded filter units within 55-gallon drums. The filtrate will be returned to the waste holdup tank. The evaporator concentrates will be pumped into 55-gallon drums containing vermiculite and cement for solidification. The miscellaneous solid waste, clothing, paper and glassware will be compressed in 55-gallon drums. The appli-



GASEOUS EFFLUENTS

FIGURE 11-2

cant has estimated that approximately 150 drums of spent resins and evaporator concentrate waste will be packaged and shipped each year. Based on the experience of operating reactors, we estimate that approximately 1000 drums containing 4900 Ci of radioactivity will be shipped from Indian Point 3 per year. All solid wastes will be packaged and shipped to a licensed burial ground in accordance with AEC and Department of Transportation regulations. We conclude that the solid waste system will have adequate capacity and is acceptable.

11.5 Design

The radioactive waste treatment systems will be designed and fabricated in accordance with acceptable codes and standards. The reactor coolant drain tank, waste filter, the spent resin storage tank, and the gas decay tanks will be designed to ASME III, Class C. The piping code will be USAS-B31.1. The equipment will be located in a Category I (seismic) structure. We have concluded that the equipment and piping designs of the radwaste systems are acceptable.

11.6 Process and Area Radiation Monitoring Systems

The process radiation monitoring systems will be designed to provide information regarding radioactivity levels in effluents released to the environment.

The liquid effluents in the discharge line from the waste condensate tanks will be monitored continuously. The monitor will automatically terminate the discharge if the radioactivity concentration exceeds predetermined values. A similar system will monitor the secondary coolant activity in the steam generators that will automatically stop the blowdown when the activity exceeds a predetermined value. The component cooling loop of the auxiliary coolant system and the essential service water system will be monitored for any primary coolant leakage into these systems. The circulating water discharge will be continuously sampled and analyzed.

The gaseous effluent in the plant vent will be continuously monitored for gross radioactivity, particulates, and radioiodine. The plant vent provides the discharge path for the gas decay tanks, the containment purge, the condenser air ejector and the ventilation systems for the primary auxiliary building and the fuel storage building. Radiation levels above a predetermined value will automatically stop the discharge from the gas decay tanks and activate the auxiliary dilution air supply to the plant vent. A similar monitoring system will serve the containment to control the purge and entry operations. Radiation levels in the containment above a predetermined value will automatically stop the purge.

A continuous monitor will measure the gross radioactivity in the effluent from the turbine condenser air ejector. Radioactivity in the gas decay tanks will be measured during the filling operation, and will alarm when the inventory limit is reached.

The air exhausted from the 11 roof-mounted exhaust fans on the turbine building will not be monitored for radioactivity since the building is not tight and therefore gases are exhausted from many places.

The area radiation monitoring system will be designed to provide information on radiation fields in the various areas of the plant for personnel protection. Monitor locations will include the control room containment, in-core instrumentation area, spent fuel building, sampling room changing pump room, and drumming station. If a radiation level rises above a predetermined value, an alarm will be sounded locally and in the control room.

Monitoring systems will detect, indicate, annunciate and/or record the levels of radioactivity to verify compliance with existing regulations to keep radiation levels within the plant and in unrestricted areas as low as practicable.

11.7 Personnel Protection

The personnel protection programs will be established to maintain exposure to plant personnel to levels as low as practicable. These programs include radiation shielding, area access control, area and personnel monitoring and protective clothing. The applicant's design objective for radiation shielding for normal operation is to maintain whole body dose rates for all controlled access areas of the plant to less than 1.25 rem per calendar year, assuming continuous occupancy and the equivalent of 1 percent fission product inventory releases from failed fuel rods into the primary coolant. The plant will be zoned into six radiation areas for personnel occupancy control. These range from continuous access at less than 0.1 mrem/hr maximum radiation to controlled access at greater than 15 mrem/hr.

Personnel monitoring equipment will be provided for all personnel at the plant. Records showing radiation exposures of all personnel at the plant will be maintained by the applicant. The records will contain at least a monthly tabulation of readings from beta-gamma-neutron film badges or their equivalent. Protective clothing and respiratory protective equipment will be available for the protection of personnel, when required.

We conclude that the personnel protection systems satisfy the requirements of existing regulations as pertains to exposure of individuals to radiation, and are acceptable.

11.8 Radiological Environmental Monitoring

A radiological environmental monitoring program has been in effect at the Indian Point site since 1958. Consequently, more than fifteen years of baseline data will be available prior to Indian Point 3 start up which can be used to predict and evaluate the potential effects of plant operation.

The Indian Point 3 monitoring program includes sampling of airborne particulates and radioiodines, lake and well water, drinking water, Hudson River water, Hudson River bottom sediments, soil, aquatic and land vegetation, milk, and Hudson River fish. The program also includes gamma spectroscopy of drinking water, Hudson River water and lake water. Tritium analysis is performed on drinking water. Airborne particulates are sampled at 21 stations which are located generally within 3 miles of the plant. In addition, direct measurements of gamma background are made annually at selected areas within a 5 mile radius of the plant. Thermoluminescent dosimeters are also located at specified offsite locations as well as a number of points on the site perimeter, for the purpose of measuring ambient radiation levels. The program conforms with Regulatory Guide 1.41 for measuring and reporting radioactivity in the environs of nuclear power plants and is acceptable.

11.9 Conclusions

We have concluded that the Indian Point 1 steam generator flash tank vent monitoring equipment will not satisfy the guidelines of Regulatory Guide 1.21 and General Design Criterion 64 and is not acceptable. The applicant has been informed that a monitoring system will be required to measure direct releases from the Indian Point 1 blowdown flash tank.

Subject to installing the above flash tank vent monitoring system prior to initial startup, we have concluded that the radioactive waste management systems will satisfy the as low as practicable guidelines of 10 CFR Parts 20 and 50, that the system is designed in accordance with acceptable codes and standards, and that the area monitoring system is similar to other monitoring systems previously accepted.

TABLE 11-1

INDIAN POINT NUCLEAR GENERATING UNIT 3
COLLECTION RATE FOR AND CAPACITY OF RADIOACTIVE
LIQUID WASTE TREATMENT SYSTEM

	Collection		Process	Monitor Tank
	Rate (gpd)	Capacity (gal)	Capacity (gpd)	Capacity (gal)
Reactor Coolant Treatment	15,000	200,000	43,000	20,000
Liquid Waste Process	470	26,000	2,900	2,000
Steam Generator Blowdown Treatment	43,000 to 220,000	----	400,000	----

12.0 RADIATION PROTECTION

12.1 Shielding

The radiation shielding is designed and the expected personnel occupancy factors are such as to allow plant operation at the maximum calculated power levels with 1.0% fuel defects without exceeding radiation doses permitted by 10 CFR Part 20 for both occupational and non-occupational personnel. The shielding for the Indian Point 3 plant is similar to other pressurized water reactors, from which considerable operating data have been obtained. On the basis of our comparison of the Indian Point 3 shielding design with that of other such plants, we conclude that the shielding is adequate to protect the health and safety of the public and operating personnel.

12.2 Ventilation

The Indian Point 3 station ventilation system is designed to provide a suitable environment for operations personnel. The primary Auxiliary Building Ventilation System allows control of flow direction of airborne radioactivity from low activity areas to higher activity areas in accordance with recommended practice. Also, the Control Room Air Conditioning, Heating, and Ventilation System is designed to permit removal of airborne particulate radioactivity from the air entering the air conditioned control room. The Ventilation system is designed to vent all compartments potentially containing airborne radioactivity to the outside.

The gaseous and particulate radioactivity monitoring system is designed to provide radiation detection equipment to provide adequate information and warning to assure that personnel exposures do not exceed 10 CFR 20 limits and to meet the intent of 10 CFR 50, Appendix A, Criterion 64 on monitoring radioactivity releases. The functions of the system are to warn operating personnel of any radiation health hazard that might develop and to give early warning of a plant malfunction which might lead to an airborne inhalation hazard.

During the review of the Indian Point 3 design the staff noted that airborne gaseous and particulate radioactivity were monitored continuously only in the plant vent, the containment system and the air ejector off-gas system. These fixed monitoring stations function primarily to monitor effluent releases and plant processes and are not effective in assuring in-plant control of personnel exposures. In-plant monitoring for radioactivity in air at Indian Point 3 was to be principally performed by portable gas and particulate monitors.

The staff felt that this system did not meet the intent of Section C.3K of Regulatory Guide 8.8 "Information Relevent to Maintaining Occupational Radiation Exposures As Low As Practicable - (Nuclear Reactors)."

In June, 1973 the staff issued a letter to the applicant requiring that fixed gaseous and particulate monitors with remote read-out provisions be installed in the radwaste area, the control room, and in the fuel handling and storage area. In a letter dated June 25, 1973 to Mr. D. B. Vassallo of the AEC, Mr. William J. Cahill, Jr. of Consolidated Edison stated that these monitors will be installed at the locations required by the AEC.

The addition of these fixed monitors, coupled with the Health Physics procedures on the use of the portable air and gas monitors, resulted in an adequate air monitoring program for plant personnel.

12.3 Health Physics Program

Radiation protection operating experience gained at Indian Point 1 and Indian Point 2 will be used to benefit the planned radiation safety program of Indian Point 3. The personnel monitoring program, the protective equipment that will be supplied to operations and maintenance personnel, and the portable radiation monitoring equipment and laboratory equipment available for day-to-day use are designed to assure that occupational exposures are maintained within the established guidelines of 10 CFR 20. The administrative controls and procedures, as well as the organization and staffing for carrying them out, are appropriate for implementing the rules and regulations set forth in 10 CFR 20. As a result of these factors, we conclude that the Health Physics program is acceptable.

13.0 CONDUCT OF OPERATIONS

13.1 Plant Organization and Staff Qualifications

The Indian Point Station staff, for Units 1, 2, and 3, will consist of approximately 380 full-time employees. The station is under the onsite supervision of the Manager, Nuclear Power Generation Department who reports to the Assistant Vice President, Power Generation Department, who in turn reports to the Executive Vice President, Central Operations. The Manager of the Nuclear Power Generation Department has the general responsibility for administering all phases of operation, training, and maintenance of the facility. The Station Manager for Operation and Maintenance and the Manager for Nuclear Services report to the Manager of the Nuclear Generation Department.

Approximately 275 people are under the direction of the Station Manager for Operation and Maintenance. About 130 of these people are assigned to the plant engineer and the remaining people are distributed among three Chief Engineers. Each Chief Engineer is responsible for administering all phases of Operation for one of the nuclear generation units. Also reporting to the Station Manager for Operation and Maintenance are five General Watch Foremen, each licensed as a Senior Reactor Operator for one of the units, who are responsible for facility operation on a shift-to-shift basis.

The Chief Engineer for Unit 3 has a staff of approximately 40 people, including an Operational Engineer who is responsible for day-to-day operation of Unit 3. Reporting to the Operational Engineer is a Watch Foreman, who has a Senior Reactor Operator license, a Senior Reactor Operator and Reactor Operator, both of whom are licensed as Reactor Operators, and three Nuclear Plant Operators. In addition, a Health Physics Technician is assigned to each shift as a shared function for all three units.

The Manager, Nuclear Services, is responsible for providing the staff services of training, technical engineering, and radiation safety. Three Directors report to the Manager, Nuclear Services. They are the Director of Nuclear Training, the Director of Technical Engineering, and the Director of Radiation Safety. These three Directors have staffs of 12, 40, and 30 persons, respectively.

The applicant has conducted a training program to train shift supervisory and control room personnel to operate Unit 3. A major feature of the training program provides that obtaining a license for Unit 2 be a prerequisite for Unit 3 licensed operating personnel for the initial plant staff. This will be followed by a three-month familiarization program to learn the differences between Unit 2 and Unit 3.

The key non-shift supervisory personnel and technical staff are currently performing their respective job functions for Units 1 and 2. Their job responsibilities are being expanded to include Unit 3.

The qualifications of key supervisory personnel with regard to educational background, experience and technical specialties have been reviewed except as noted below and are in general conformance with those defined in ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel." Personnel have not as yet been assigned to the positions of the Reactor Operator and Watch Foremen. The AEC staff will review the qualifications of the Watch Foremen as they become available to confirm that the intent of ANSI N18.1 has been met.

Technical support for the plant staff is available from the home office Departments of Mechanical Engineering, General Engineering, Electrical Engineering, Civil Engineering and the Office of Environmental Affairs. Additional technical support during the startup test program will be provided by WEDCO, a wholly owned subsidiary of Westinghouse Electric Corporation.

We have concluded that the organizational structure, the training, and qualifications of the staff for Indian Point are is adequate to provide an acceptable operating staff and technical support for the safe operation of the facility.

13.2 Emergency Planning

The applicant has established an organization for coping with emergencies. The plan includes written agreements, liaison and communications with appropriate local, State and Federal agencies that have responsibilities for coping with emergencies. The applicant has defined categories of incidents, including criteria for determining when protective measures should be considered and for the notification of offsite support groups. Arrangements have been made by the applicant to provide for medical support in the event of a radiological incident or other emergencies. Provisions for periodic training for both plant personnel and offsite emergency organizations have been included in the Emergency Plan. Most elements of this plan are currently in effect for Units 1 and 2.

Numerous improvements to the Consolidated Edison's Emergency Plan were brought about by the AEC staff during its review of Unit 3. These improvements include:

(1) A more rapid method of estimating offsite doses in case of an emergency. Emergency alerts can now be issued for information available to the operator in the control room rather than waiting for the results of an offsite survey.

(2) The Emergency Plan has been augmented to include notification of the Penn Central Railroad in case of an emergency.

(3) Additional letters of agreement from medical support facilities and the Coast Guard have been acquired. These groups could potentially participate in case of an emergency.

We have reviewed the augmented Emergency Plan and conclude that it meets the criteria of Appendix E of 10 CFR 50, and that adequate arrangements have been made to cope with the possible consequences of the accidents at the site, and that there is reasonable assurance that such arrangements will be satisfactorily implemented in the unlikely event that they are needed.

13.3 Safety Review and Audit

The safety review and audit function for Indian Point 3 will be conducted by the Nuclear Facilities Safety Committee. This committee was established in 1962 and has been performing that function since then for Unit 1 and subsequently for Unit 2. The Nuclear Facilities Safety Committee is advisory to the Executive Vice President, and the President and Chairman of the Board and provides corporate management with a review and audit capability to verify that organizational checks and balances are functioning to assure continued safe operation and design adequacy of the plant. In a letter dated April 12, 1973, from William J. Cahill, Jr., Vice President of Consolidated Edison of New York, to Mr. R. C. DeYoung, Directorate of Licensing, U. S. Atomic Energy Commission, the applicant has assured the AEC staff that the Nuclear Facilities Safety Committee will function in accordance with the requirements of ANSI N18.7 "Standard for Administrative Control for Nuclear Power Plants," Sections 3.0 through 4.4.

Details of responsibility and authority of the review and audit functions are given in Section 6 of the Technical Specifications.

We conclude that the applicant's plans for the Safety Review and Audit functions are acceptable.

13.4 Plant Procedures

Plant operations are to be performed in accordance with written and approved operating and emergency procedures. Areas covered include normal startup, operation and shutdown, abnormal conditions and emergencies, refueling, maintenance, surveillance, testing, and radiation control. All procedures and changes thereto will be reviewed prior to implementation by the applicant. Safety-related procedures will be given a thorough review by the Unit 3 Operating Staff. These procedures then require the approval of the Unit 3 Chief Engineer after review and comment by the Nuclear Facility Safety Committee.

We conclude that the provisions for preparation, review, approval, and use of written procedures are satisfactory.

13.5 Industrial Security

The applicant has submitted a description of its Industrial Security Plan for protection of the Indian Point Nuclear Power Station Unit 3 from industrial sabotage. The information was submitted as proprietary information and is withheld from public disclosure pursuant to Section 2.790 of the Commission's regulations. We have reviewed the program and conclude that adequate security provisions have been made for Indian Point Unit 3, and that it meets the intent and principles of Safety Guide No. 17.

14.0 INITIAL TESTS AND OPERATION

The initial startup, including preoperational checkout of equipment, functional and system tests, fuel loading, initial criticality and power operation will be performed by the regular plant staff. Technical assistance will be provided by WEDCO and Westinghouse. The WEDCO and Westinghouse personnel will assist in writing procedures, interpreting test results and any problems that may arise during the testing program.

We have reviewed the applicant's preoperational and startup testing program and conclude that it is in general accord with the AEC publications "Guide for the Planning of Preoperational Testing Programs" and "Guide for the Planning of Initial Startup Programs." The program will provide an adequate basis to confirm the safe operation of the plant and is therefore acceptable.

15.0 ACCIDENT ANALYSES

15.1 General

The applicant has analyzed reactor performance for normal steady-state plant operation and for anticipated operational transients on the basis of the initial core power level of 3025 megawatts thermal (MWt).

The postulated design basis accidents analyzed for offsite radiological consequences by the applicant are the same as those analyzed for previously licensed PWR plants, including a steam line break accident, a steam generator tube-rupture accident, a loss-of-coolant accident, a fuel-handling accident, and a rupture of a radioactive gas-storage tank in the gaseous radioactive waste treatment system.

On the basis of our experience with the evaluation of the steam-line break, the steam generator tube rupture, and radioactive gas-storage tank rupture accidents for PWR plants of similar design, we have concluded that the consequences of these accidents can be controlled by limiting the permissible primary and secondary coolant system radioactivity concentrations and the permissible inventory of radioactivity in a gas storage tank so that potential offsite doses are small. We will include limits in the Technical Specifications on primary and secondary coolant radioactivity concentrations and on the radioactivity in a gas storage tank such that the potential two-hour doses at the exclusion radius that we calculate for these accidents are well below the 10 CFR Part 100 guideline values.

15.2 Iodine Removal Equipment

15.2.1 Spray

An internal recirculation containment spray system is provided to remove heat from the containment atmosphere and to remove iodine which may be present in the containment following a loss-of-coolant accident. Initially, the two containment spray pumps take suction on the refueling water storage tank and deliver water to spray nozzles inside containment. Each pump has a design capacity of 2600 gpm. Concentrated sodium hydroxide solution is added at the suction of the spray pumps in quantities sufficient to maintain a pH of at least 9.3 in the water in the containment spray. Sodium hydroxide in the containment spray water will scavenge elemental radioiodine from the containment atmosphere. When the refueling water storage tank is exhausted, a portion of the recirculation flow provided for continued core cooling is diverted to the containment spray headers.

To calculate the total iodine removal constant for the proposed system, we made conservative assumptions regarding liquid film mass resistance and drop coalescence. Consistent with the conclusions of WASH-1233,* we assumed that 4% of the iodine in the

*WASH-1233 "Review of Organic Iodine Formation Under Accident Conditions in Water-Cooled Reactors" Published by the AEC, October, 1972.

containment atmosphere is in the form of organic iodides and 5% in a particulate form. Experiments have shown that sodium hydroxide spray solutions are not efficient in the removal of organic iodides; therefore, we assumed no reduction of the organic iodides by the containment spray.

We calculated an elemental iodine removal constant of 9.85 hr^{-1}. A two-hour reduction factor for the iodine accident dose at the exclusion area boundary of 5.2 and a thirty-day reduction factor for the iodine accident dose at the outer boundary of the low population zone of 8.8 was calculated as a result of iodine removal by the chemical additive sprays. Table 15.3 of this report lists removal rates and reduction limits for each form of iodine and the dose reduction factors due to the use of the sprays and filters.

15.2.2 Charcoal Filters

The air handling system (1) will remove heat from the containment in the post-accident environment and (2) will reduce the iodine concentration in the containment atmosphere by the use of charcoal filters. Five air handling units are provided. In each unit, a fan draws air through a moisture separator, cooling coils, roughing filters and high efficiency particulate air (HEPA) filters at a flow rate of approximately 24,000 cfm under post-accident conditions. Charcoal filters are located at the fan discharge header. They are isolated by butterfly valves. Under accident conditions, these valves are automatically opened by the high containment pressure signal and a flow rate of 8,000 cfm is diverted through these filters. Three of the five air handling units will operate even if normal offsite power is lost. This was assumed in our analyses. Under this circumstance, approximately 150% of the free volume of the containment is processed through the charcoal filters each hour.

Research performed to date using impregnated charcoals of various manufacturers indicates that at 100% relative humidity the removal efficiency decreases to about 70% for methyl iodide and to about 99% for elemental and particulate iodine. The staff assumes a value of 30% for methyl iodide and 90% for elemental and particulate iodine for the purposes of site and engineered safety feature evaluation. Together, the spray and filters reduce the overall two-hour iodine accident dose at the exclusion area boundary by a factor of 6.4 and the thirty-day overall iodine accident dose at the outer boundary of the low population zone by a factor of 20.

15.3 Radiological Consequences of Postulated Accidents

The postulated design basis accidents analyzed by the applicant and by us for offsite radiological consequences are the same as those analyzed for previously licensed PWR plants of similar design. The offsite doses calculated by the staff for these accidents are presented in Table 15.1 and the assumptions used are listed in Table 15.2 of this report. All doses are within 10 CFR Part 100 guideline values.

15.4 Control Room Doses

The applicant has met the requirements of General Design Criterion 19 of Appendix A to 10 CFR Part 50 by use of adequate concrete shielding around the control room and by filtering inlet air to the control room during emergencies. Under emergency conditions the air in the control room is recirculated and filtered through redundant 2,000 cfm clean up trains which consist of HEPA filters and two inch-deep charcoal beds. About 200 cfm of make-up air is added upsteam of the filter trains to assure control room pressurization. The units are automatically activated upon accident or high radiation signals.

The staff has calculated the potential radiation doses to control room personnel following a LOCA. The resulting doses are within the requirements set by General Design Criterion 19.

TABLE 15.1

POTENTIAL OFFSITE DOSES CALCULATED BY

STAFF FOR DESIGN BASIS ACCIDENTS AT 3025 MWT OPERATION*

ACCIDENT	EXCLUSION BOUNDARY TWO HOUR (330 METERS)		LOW POPULATION ZONE COURSE OF ACCIDENT (1100 METERS)	
	Thyroid (Rem)	Whole Body (Rem)	Thyroid (Rem)	Whole Body (Rem)
Loss of Coolant**	288	22	119	15
Refueling	67	8	19	2
Gas Decay Tank*** Rupture	Negligible	8	Negligible	2

-* Our calculated potential doses to control room personnel following a LOCA are within the guidelines of Criterion 19.

* The 2 Hour site boundary dose using the stretch power level of 3216 Mwt is 302 rem thyroid.

*** The Technical Specifications for Unit #3 will be set to reduce the inventory of noble gases stored in a single gas decay tank so that any single failure such as lifting and sticking of a pressure relief valve will not produce a whole body dose in excess of 0.5 rem at the site boundary.

TABLE 15.2

ASSUMPTIONS USED BY AEC REGULATORY STAFF IN CALCULATIONS OF OFFSITE DOSES FROM DESIGN BASIS ACCIDENTS

Loss-of-Coolant Accident Assumptions

Power Level	3025 Mwt
Operating Time	3 Years
Primary Containment Leak Rage	0.1%/day $\leq$24 Hours
	0.05%/day >24 Hours
Initial Iodine Form Distribution	91% Elemental
	4% Organic
	5% Particulate
Spray Filter Data:	
Filter Flow Rate	24,000 cfm
Filter Efficiencies	
Organic Iodine	30%
Particulate Iodine	90%
Elemental Iodine	90%
Primary Containment Volume	2.61×10^6 ft^3
Spray Fall Height	118 feet
Spray Flow Rate	2500 gpm
Elemental Mass Transfer Velocity	4.74 cm/sec
Spray Drop Diameter	1500 μ
Spray Terminal Velocity	480 cm/sec
Factor of Conservatism	1.11

X/Q Data, sec/m^3

Exclusion Boundary (330 meters)	
0-2 Hours (Equivalent to Pasquill "F", $\bar{\mu}$ = 0.7 m/sec)	1.8×10^{-3}
Low Population Zone Boundary (1100 meters)	
0-8 Hours (Equivalent to Pasquill"F", $\bar{\mu}$ = 0.7 m/sec)	4.7×10^{-4}
8-24 Hours	1.4×10^{-4}
24-96 Hours	6.5×10^{-5}
96-720 Hours	2.2×10^{-5}

TABLE 15.2 (Cont'd)

Refueling Accident Assumptions

1. Rupture of 204 fuel rods (one assembly).

2. All gap activity in the rods, assumed to be 10% of the noble gases and 10% of the iodine (with a peaking factor of 1.7), is released.

3. The accident occurs 100 hours after shutdown.

4. 99% of the iodine is retained in the pool water.

5. Iodine filter efficiencies of 70% and 90% for organic and elemental forms respectively.

6. On-site data used to determine X/Q values for ground release meteorology, and dose conversion factor.

Gas Decay Tank Rupture Assumptions

1. Gas decay tank contains all the primary coolant loop inventory of noble gases resulting from operation with 1% failed fuel (100,000 curies equivalent of Xe^{133}).

2. X/Q values based on on-site meteorological data.

TABLE 15.3

REMOVAL RATES AND REDUCTION LIMITS FOR EACH FORM OF IODINE

	Iodine Removal Rates, $Hrs.^{-1}$		
Time Period, Hours	Elemental	Particulate	Organic
0-0.448	10.3	0.897	0.149
0.448-5.13	0.447	0.897	0.149
5.13-10.28	0.447	0.447	0.149
10.28-10.75	0.447	0	0.149
10.75-46.36	0	0	0.149
16.36-720	0	0	0
		Reduction Limits	
Sprays	100	100	1
Filters	10,000	1,000	1,000

DOSE REDUCTION FACTORS DUE TO USE OF SPRAYS + INTERNAL FILTERS

Time	Thyroid	Whole Body
0 - 2 Hours	6.4	1.4
0 - 30 Days	20	1.5

16.0 TECHNICAL SPECIFICATIONS

The Technical Specifications in a license define certain features, characteristics, and conditions governing operation of a facility that cannot be changed without prior approval of the AEC. We reviewed the proposed Technical Specifications in detail and have held a number of meetings with the applicant to discuss their contents. Modifications to the proposed Technical Specifications submitted by the applicant were made to describe more clearly the allowed conditions for plant operation. The finally approved Technical Specifications will be made part of the operating license. Included will be sections covering safety limits and limiting safety system settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls. On the basis of our review, we conclude that normal plant operation within the limits of the Technical Specifications will not result in potential offsite exposures in excess of the 10 CFR Part 20 limits. Furthermore, the limiting conditions for operation and surveillance requirements will assure that necessary engineered safety features will be available in the event of malfunctions within the plant.

17.0 QUALITY ASSURANCE

17.1 General

The applicant has a turnkey contract with Westinghouse to provide the Nuclear Steam Supply System. Originally, United Engineers and Constructors (UE&C) had served as the Architect-Engineer. In 1969, the responsibility for the construction of the plant was assumed by WEDCO, a wholly owned subsidiary of Westinghouse. Each of the organizations has a quality assurance program. The applicant, in support of its overall responsibility for the quality assurance program, has retained the U. S. Testing Company (USTC) as its quality assurance agent. USTC's duties include audit of test procedures and physical certifications for compliance with accepted standards. As a part of this audit, USTC visits the various manufacturing facilities on behalf of the applicant; reports of these visits are forwarded to both the applicant and Westinghouse.

Our review was based on the information presented in Appendix B of the FSAR and reports from the AEC's Directorate of Regulatory Operations. The Directorate of Regulatory Operations has performed detailed inspections of work in progress both at the reactor site and at vendor shops.

In November 1969 members of the AEC staff inspected the applicant's offices, USTC, and UE&C. This inspection compared the applicant's quality assurance program to 18 criteria, which, in July 1970, became

Appendix B to 10 CFR Part 50. It was found that the applicant's quality assurance program was in general accord with the then developing Appendix B to 10 CFR Part 50.

During the construction phase other inspections have been held by the Directorate of Regulatory Operations. Any deficiencies uncovered by these inspections have been communicated to the applicant. The Directorate of Regulatory Operations will assure satisfactory resolution of all deficiencies prior to the issuance of an operating license.

In addition to reviewing the applicant's QA program for the construction of this facility, we reviewed the applicant's ability to comply with the requirements of Appendix B to 10 CFR Part 50 for the operational phase of Indian Point 3. This review was based on Appendix B to the FSAR, supplemented by information in Supplements 10 and 15 to the FSAR, and letters of commitment from the applicant, dated April 12, 1973 and June 28, 1973. This review is discussed below.

17.2 Organization and Program

Responsibility and authority to define and direct the QA Program is assigned by the applicant to its Vice-President for Quality Assurance and Reliability (QA&R) who reports directly to the Executive Vice-President, Central Operations. Reporting to the Vice-President of QA&R are a Director of Quality Assurance and a Director of Quality Standards and Reliability. On the staff of the Director of Quality

Assurance are a QA Manager for Engineering, a QA Manager for Operations, and QA Project Engineers, including one for the Indian Point 3 facility.

QA&R's responsibilities include review of specifications, design drawings, and modification, maintenance, and repair procedures for adequacy of QA provisions and verification of conformance to the quality assurance procedures. The Director of Quality Standards and Reliability is staffed with consultants having backgrounds in metallurgy, welding, non-destructive examination, reliability, quality systems, electrical engineering, and mechanical engineering.

The responsibility for operating and maintaining Indian Point 3 is assigned to the Vice-President of Power Supply who is on the same organizational level as the Vice-President for QA&R. An onsite Station QA Engineer reports to the Power Supply organization and is responsible for the effective implementation of onsite QA and Quality Control (QC) functions. When technical support is required or necessary, he has direct access to the centralized QA organization under the Vice-President of QA&R. The Station QA Engineer is independent of the Station Manager for Operation and Maintenance in that both persons are on the same organizational level. The Station QA Engineer and his staff perform quality control inspections, in-service inspection, receipt inspection, and control the Station Central Files.

Indian Point 3 does not have an onsite review committee but has, in addition to the Station QA Engineer and headquarters QA&R staff, a

Nuclear Facilities Safety Committee (NFSC) responsible for advising the Executive Vice-President of Central Operations on safety aspects of the applicant's nuclear power facilities.

Based on our review of the applicant's organizational arrangements for the QA Program for Operations we conclude that adequate control, independence, authority, and management involvement are provided and that the QA organization is acceptable for the operational phase.

As part of our review, we requested the applicant to indicate its compliance with the provisions of AEC Regulatory Guide 1.33 "Quality Assurance Program Requirements (Operation)". The applicant had already committed to Appendix A of Regulatory Guide 1.33 and to ANSI N 45.2, but had not committed to ANS 3.2, draft 8. (now ANSI N 18.7) which is also part of AEC Regulatory Guide 1.33. In a letter dated April 12, 1973, the applicant stated its intent to implement both the requirements and recommendations of Section 4.0 of ANS 3.2, to evaluate the remaining sections of the standard, and to respond to the staff on these remaining sections by July 1, 1973.

In a letter dated June 28, 1973, from William Cahill, the applicant committed to the remaining sections of ANS 3.2 draft No. 8 with one minor, acceptable exception.

17.3 Audits

The Nuclear Facility Safety Committee (NFSC) will provide an independent review and audit of operations. This will include audits of the adequacy and implementation of all procedures used in the operation, maintenance, and environmental monitoring of each of its nuclear power plants not less than once a year. QA&R will audit compliance with this program and shall be responsible for assuring that necessary corrective actions are implemented. QA&R will also monitor maintenance, modification, and repair activities, principally through the inspection efforts of the onsite QA Engineer. QA&R prepares and distributes a monthly report which identifies significant conditions adverse to quality, corrective actions taken, and reports these to appropriate levels of management.

Based on our review of the Indian Point 3 audit program and the applicant's commitment to implement both the recommendations and the requirements of Section 4 of ANS 3.2, draft 8, we conclude that these audits will provide acceptable management attention to quality and safety related activities during the operational phase and will meet the requirements of Appendix B to 10 CFR 50.

17.4 Conclusions

We conclude that the QA Program for Indian Point 3 described in the FSAR, as amended, complies with the requirements of Appendix B to 10 CFR Part 50 and is acceptable for the operational phase of this facility.

18.0 THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

The application for an operating license for the Indian Point Nuclear Generating Unit No. 3 is being reviewed by the ACRS. We intend to issue a Supplement to this Safety Evaluation after the Committee's report to the Commission relative to its review is available. The Supplement will append a copy of the Committee's report and will address the significant comments made by the Committee, and will also describe steps taken by the staff to resolve any issues raised as a result of the Committee's review. The Supplement will also describe the resolution of those issues raised by the staff review that are not completely resolved at this time.

19.0 COMMON DEFENSE AND SECURITY

The application reflects that the activities to be conducted will be within the jurisdiction of the United States and that all of the directors and principal officers of the applicant are United States citizens. The applicant is not owned, dominated, or controlled by an alien, a foreign corporation or a foreign government. The activities to be conducted do not involve any restricted data, but the applicant has agreed to safeguard any such data that might become involved in accordance with the requirements of 10 CFR Part 50. The applicant will rely upon obtaining fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material for military purposes is involved. For these reasons and in the absence of any information to the contrary, we have found that the activities to be performed will not be inimical to the common defense and security.

20.0 FINANCIAL QUALIFICATIONS

The Commission's regulations which relate to financial data and information required to establish financial qualifications for an applicant for a facility operating license are 10 CFR 50 Part 33(f) and 10 CFR 50, Appendix C. We have reviewed the financial information presented in the application and have concluded that the applicant is financially qualified to operate Indian Point 3. A detailed discussion of the basis for our conclusion is presented in Appendix D of this report.

21.0 FINANCIAL PROTECTION AND INDEMNITY REQUIREMENTS

Pursuant to the financial protection and indemnification provisions of the Atomic Energy Act of 1954, as amended (Section 170 and related sections of the Commission's regulations), the Commission has issued regulations in 10 CFR Part 140. These regulations set forth the Commission's requirements with regard to proof of financial protection by, and indemnification of, licensees for facilities such as power reactors under 10 CFR Part 50.

21.1 Preoperational Storage of Nuclear Fuel

The Commission's regulations in 10 CFR Part 140 require that each holder of a construction permit under 10 CFR Part 50, who is also to be the holder of a license under 10 CFR Part 70 authorizing the ownership and possession for storage only of special nuclear material at the reactor construction site for future use as fuel in the reactor (after issuance of an operating license under 10 CFR Part 50), shall, during the interim storage period prior to licensed operation, have and maintain financial protection in the amount of $1,000,000 and execute an indemnity agreement with the Commission. Proof of financial protection is to be furnished prior to, and the indemnity agreement executed as of, the effective date of the 10 CFR Part 70 license. Payment of an annual indemnity fee is required.

The applicant is, with respect to Indian Point 3, subject to the foregoing requirements.

The applicant has furnished to the Commission proof of financial protection in the amount of $95,000,000 in the form of Nuclear Energy Liability Insurance Association policy (Nuclear Energy Liability Policy, facility form) No. NF-100 and a Mutual Atomic Energy Liability Underwriters policy (Nuclear Energy Liability Policy, facility form) No. MF-29, to cover operations of Indian Point Units 1 and 2.

Further, the applicant executed Indemnity Agreement No. B-19 with the Commission as of January 12, 1962. At such time as a pertinent license is issued for preoperational fuel storage for Indian Point 3, the Indemnity Agreement will be amended to cover that preoperational fuel storage. The applicant will be required to pay the annual indemnity fee applicable to preoperational fuel storage in addition to the indemnity fees it is presently paying.

21.2 Operating License

Under the Commission's regulations, 10 CFR Part 140, a license authorizing the operation of a reactor may not be issued until proof of financial protection in the amount required for such operation has been furnished, and an indemnity agreement covering such operation (as distinguished from preoperational fuel storage only) has been executed. The amount of financial protection which must be maintained for reactors which have a rated capacity of 100,000 electrical kilowatts or more is the maximum amount available from private sources, i.e., the combined capacity of the two nuclear liability insurance pools, which amount is currently \$95 million. Accordingly, no license authorizing operation of Indian Point 3 will be issued until proof of financial protection in the requisite amount has been received and the requisite indemnity agreement amended.

We expect that, in accordance with the usual procedure, the nuclear liability insurance pools will provide, several days in advance of anticipated issuance of the operating license document, evidence in writing, on behalf of the applicant, that the present coverage has been appropriately amended to include the new facility, to meet the requirements of the Commission's regulations for reactor operation. The amount of financial protection required for a reactor having the rated capacity of this facility would be \$95 million. The applicant is currently provided with this amount of financial protection in connection with Indian Point 1 and 2. The applicant will be required to pay an annual fee for operating license indemnity as provided in our regulations, at the rate of \$30 per each thousand kilowatts of thermal capacity authorized in its operating license.

On the basis of the above considerations, we conclude that the presently applicable requirements of 10 CFR Part 140 have been satisfied and that, prior to issuance of the operating license, the applicant will be required to comply with the provisions of 10 CFR Part 140 applicable to operating licensees, including those as to proof of financial protection in the requisite amount and as to execution of an appropriate indemnity agreement with the Commission.

22.0 CONCLUSIONS

Based on our evaluation of the application as set forth above, it is our position that upon favorable resolution of the outstanding matters described in Section 6.5, Section 9.2 and Section 11, we will be able to conclude that:

(1) The application for facility license filed by the applicant dated April 26, 1967, as amended (Amendments 1 through 31 of the original application and Amendments 1 and 2 of the Amended and Substituted Application) complies with the requirements of the Atomic Energy Act of 1954, as amended (Act), and the Commission's regulations set forth in 10 CFR Chapter 1.

(2) The construction of the Indian Point Nuclear Generating Unit No. 3 (the facility) has proceeded and there is reasonable assurance that it will be complete, in conformity with Provisional Construction Permit No. CPPR-62, the application as amended, the provisions of the Act, and the rules and regulations of the Commission.

(3) The facility will operate in conformity with the application as amended, the provisions of the Act, and the rules and regulations of the Commission.

(4) There is reasonable assurance (i) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR Part 1.

(5) The applicant is technically and financially qualified to engaged in the activities authorized by an operating license in accordance with the regulations of the Commission set forth in 10 CFR Part 1.

(6) The issuance of an operating license for the facility will not be inimical to the common defense and security or to the health and safety of the public.

Prior to final consideration of the matter of the issuance of an operating license to the applicant for the Indian Point 3, the unit must be completed in conformity with the construction permit, the application, the Act, and the rules and regulations of the Commission. Such completeness of construction as is required for safe operation at the authorized power level must be verified by the Commission's Directorate of Regulatory Operations prior to issuance of a license. Further, before an operating license is issued, the applicant will be required to satisfy the applicable provisions of 10 CFR Part 140.

Appendix A

Chronology of Radiological Review

CHRONOLOGY

REGULATORY RADIOLOGICAL REVIEW OF INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

December 4, 1970	Submittal of the Final Facility Description and Safety Analysis Report (Amendment No. 13 to the Application for Licenses)
February 4, 1971	Initial meeting with applicant, Westinghouse Electric Corporation, and United Engineers and Constructors to discuss projected construction schedule.
March 9, 1971	Meeting with applicant to discuss electrical design of Indian Point 3
May 18-19, 1971	Meeting with applicant to review instrumentation and control system drawings
June 30, 1971	Submittal of Amendment No. 14 and report, "Effect of Tornado Missiles on Stored Spent Fuel"
July 8, 1971	Submittal of Amendment No. 15 (Supplement No. 1 to Final Facility Description and Safety Analysis Report (FFDSAR)), consisting of revised and additional pages
August 2, 1971	Letter to applicant requesting additional information on emergency core cooling systems
August 13, 1971	Letter to applicant requesting additional information
November 10, 1971	Letter to applicant requesting additional information
February 10, 1972	Letter to applicant concerning implementation of an inservice inspection program for Indian Point Unit 3
February 16, 1972	Letter to applicant advising of revised review schedule and date for ACRS meeting
February 23, 1972	Letter from applicant advising review schedule in AEC letter of February 16 corresponds with construction schedule

April 3, 1972	Submittal of Amendment No. 16 (Supplement No. 2), consisting of responses to request of August 13, 1971, and revised pages
April 11, 1972	Letter to applicant transmitting draft criteria regarding industrial security
April 27, 1972	Letter to applicant advising that a public document room has been established in the vicinity of the plant
May 5, 1972	Submittal of Amendment No. 17 (Supplement No. 3), consisting of additional responses to request of August 13, 1971, and revised pages
June 5, 1972	Letter to applicant requesting additional financial information
June 19, 1972	Letter to applicant summarizing basis for AEC decision to delay review of Indian Point 3
June 30, 1972	Submittal of Amendment No. 18 (Supplement No. 4), consisting of additional responses to request of August 13, 1971, and revised pages
July 12, 1972	Letter to applicant requesting additional information
July 19, 1972	Letter from applicant transmitting a petition requesting extension of completion date of Indian Point 3
July 30, 1972	Submittal of Amendment No. 19 (Supplement No. 5), consisting of additional responses to request of August 13, 1971, partial response to request of November 10, 1971, and revised pages
August 1, 1972	Submittal of Amendment No. 20 (Supplement No. 6), consisting of response to request of August 2, 1971
August 1, 1972	Submittal of Amendment No. 21 (Supplement No. 7), consisting of additional responses to requests of August 13, 1971, and November 10, 1971
August 14, 1972	Issuance of Order extending completion date

August 25, 1972	Submittal of Amendment No. 22 (Supplement No. 8), consisting of response to request of July 12, 1972, and corrections to Supplement No. 7
September 26, 1972	Letter to applicant requesting analysis of results of failure of non-Category I (seismic) equipment
October 13, 1972	Meeting with applicant concerning Indian Point 3 emergency plans and conduct of operations
October 19, 1972	Meeting with applicant to discuss quality assurance
October 19, 1972	Issuance of Notice of Consideration of Issuance of Facility License and Notice of Opportunity for Hearing
October 30, 1972	Letter from applicant concerning request of September 26, 1972
November 6, 1972	Letter to applicant requesting additional information
November 10, 1972	Meeting with applicant concerning electrical drawings
November 20, 1972	Letter to applicant requesting analysis of consequences of fuel densification
December 19, 1972	Letter to applicant requesting analysis of consequences of postulated pipe failures outside containment
January 9, 1973	Letter from applicant concerning request of November 20, 1972
January 12, 1973	Submittal of Amendment No. 23 (Supplement No. 9), consisting in part of revised proposed technical specifications
January 12, 1973	Letter to applicant requesting updated financial information
January 16, 1973	Letter from applicant transmitting report, "Summary Report of Reinspection and Appraisal of the Indian Point Unit No. 3 Reactor Pressure Vessel Subsequent to Hoist Failure on January 12, 1971"

January 19, 1973	Submittal of Amendment No. 24 (Supplement No. 10), consisting of partial response to request of November 6, 1972, and Industrial Security Plan
January 22, 1973	Letter to applicant requesting additional information
January 23, 1973	Letter from applicant responding to request of September 26, 1972
January 24, 1973	Letter to applicant transmitting errata sheet for letter of December 19, 1972
January 31, 1973	Submittal of Amendment No. 25 (Supplement No. 11), consisting of partial response to request of November 6, 1972, and revised pages
February 6, 1973	Meeting with applicant to discuss high energy fluid lines
February 12, 1973	Letter from N. M. Newmark transmitting comments on the structural adequacy of Indian Point 3
February 16, 1973	Submittal of Amendment No. 26 (Supplement No. 12), consisting of partial response to request of November 6, 1972
February 20, 1973	Submittal of Amendment No. 27, consisting of financial information
February 23, 1973	Meeting with applicant to discuss electrical modifications required by AEC letter of January 22, 1973
February 28, 1973	Notice of Hearing
March 2, 1973	Submittal of Amendment No. 28 (Supplement No. 13), consisting of operating staff resumes, meteorological data, and response to request of January 22, 1973
March 5, 1973	Meeting with applicant concerning site meteorology
March 16, 1973	Submittal of Amendment No. 29 (Supplement No. 14), consisting of additional meteorology data, and revised pages

March 22, 1973	Letter to applicant requesting information on mechanical and instrumentation, control, and electrical systems
March 27, 1973	Submittal of Amendment No. 30 (Supplement No. 15), consisting of revised and additional pages and information regarding quality assurance program for post-construction phase
March 30, 1973	Letter to applicant requesting information on the quality assurance program for Indian Point 3
March 30, 1973	Letter to applicant requesting information relating to a control design deficiency
April 2, 1973	Letter from applicant in response to request of January 22, 1973
April 3, 1973	Letter from applicant transmitting proprietary and nonproprietary reports on fuel densification
April 9, 1973	Submittal of Amendment No. 31 (Supplement No. 16), consisting of revised pages and the Industrial Security Plan
April 9, 1973	Letter from applicant in response to request of March 22, 1973
April 16, 1973	Submittal of Revised and Substituted Application for Licenses to Atomic Safety and Licensing Board, et al
April 23, 1973	Meeting with applicant concerning effluent treatment
April 27, 1973	Submittal of Amendment No. 1 (Supplement No. 17), consisting of emergency plans and revised pages
May 2, 1973	Letter to applicant regarding low pressure piping
May 4, 1974	Letter to applicant advising that Industrial Security Plan transmitting April 9 will be withheld from public disclosure and returning January 19 version of plan
May 8, 1973	Submittal of Amendment No. 2 (Supplement No. 18), consisting of corrected pages

May 14, 1973	Letter from applicant transmitting report, "Analysis of High Energy Lines," in response to request of January 24, 1973
May 18, 1973	Letter from applicant transmitting to staff the Amended and Substituted Application to FSAR, dated April 13, 1973
May 21, 1973	Letter to applicant concerning current schedule for review of application
May 25, 1973	Letter from applicant in response to request of May 2, 1973
May 31, 1973	Meeting with applicant to discuss emergency core cooling systems
June 6, 1973	Submittal of Amendment No. 3 (Supplement No. 19), consisting of revised pages for FSAR and Security Plan
June 8, 1973	Letter from applicant transmitting report, "Dynamic Analysis of a Postulated Main or Feedwater Line Pipe Break Outside Containment"
June 11, 1973	Letter to applicant regarding airborne gaseous and particulate monitoring system
June 20, 1973	Letter from applicant in response to AEC letter of May 21, 1973, transmitting proposed revised schedule
June 27, 1973	Letter from applicant in response to AEC letter of March 30, 1973
June 28, 1973	Letter from applicant transmitting nonproprietary electrical drawings
June 28, 1973	Letter from applicant in response to AEC letter of March 30, 1973
July 5, 1973	Letter to applicant requesting review of the refueling water storage tank system design
July 6, 1973	Submittal of Amendment No. 4 (Supplement No. 20), consisting of corrected pages for FSAR

July 11, 1973	Meeting with ACRS Subcommittee
July 19, 1973	Letter to applicant stating that proprietary reports on fuel densification will be withheld from public disclosure
July 24, 1973	Letter from applicant in response to AEC letter of March 30, 1973, concerning control design deficiency
August 17, 1973	Letter from applicant transmitting proprietary and nonproprietary versions of report on fuel densification
August 24, 1973	Submittal of Amendment No. 5 (Supplement No. 21) consisting of corrected pages
August 31, 1973	Letter from applicant in response to request of July 5, 1973, requesting extension of time for submittal of information
September 7, 1973	Letter from applicant providing supplemental information to its April 2, 1973 and May 2, 1973 letters
September 14, 1973	Letter to applicant concerning the startup test program

Appendix B

Report on the Structural Adequacy of the

Indian Point Nuclear Generating Unit No. 3

by W. J. Hall and N. M. Newmark

NATHAN M. NEWMARK
CONSULTING ENGINEERING SERVICES

1114 CIVIL ENGINEERING BUILDING
URBANA, ILLINOIS 61801

12 February 1973

Mr. R. R. Maccary
Assistant Director for Engineering
Office of Technical Review
Directorate of Licensing
U.S. Atomic Energy Commission
Washington, D.C. 20545

Re: Contract No. AT(49-5)-2667
<u>Commentary</u>
<u>Final Report</u>
Indian Point Nuclear Generating Unit No. 3
Consolidated Edison Company of New York, Inc.
AEC Docket No. 50-286

Dear Mr. Maccary:

Dr. N. M. Newmark and I have reviewed the Final Safety Analysis Report for the Indian Point Nuclear Generating Unit No. 3 and are transmitting herewith 8 signed copies of our Commentary and Final Report.

Since we have previously visited the Indian Point Nuclear 2 unit which is constructed along the same lines as Indian Point No. 3, it probably will not be necessary for us to visit this facility but we will await instructions from your personnel in this regard.

Sincerely yours,

W J Hall

W. J. Hall

pg

Enclosure

cc: N. M. Newmark

APPENDIX B

NATHAN M. NEWMARK
CONSULTING ENGINEERING SERVICES

1114 CIVIL ENGINEERING BUILDING
URBANA, ILLINOIS 61801

12 February 1973

COMMENTARY
ON
STRUCTURAL ADEQUACY
OF THE
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

AEC Docket No. 50-286

by W. J. Hall and N. M. Newmark

1. Introduction

This report is based on information presented in the Indian Point Nuclear Generating Unit No. 3 FSAR and the Supplements thereto (Ref. 1) and on discussions with personnel of the AEC Directorate of Licensing. Specific items are singled out for discussion herein, and no attempt is made to review the basis of the seismic design criteria as reported in our PSAR review for this plant (Ref. 2) or in our related FSAR review for Indian Point Nuclear Generating Unit No. 2 (Ref. 3).

2. Foundations

The major facility structures for Indian Point Nuclear Generating Unit No. 3 are described as being founded directly on competent bedrock, and on the basis of the information available to us the foundation conditions appear acceptable for the seismic hazards noted.

3. Seismic Design

Seismic Hazard

As noted on page 5.1.2-4, the dynamic analysis is to be carried out for a Design Basis Earthquake characterized by 0.15g maximum transient horizontal ground acceleration and for an Operating Basis Earthquake characterized similarly by a 0.1g maximum horizontal ground acceleration. For vertical excitation, an earthquake characterized by 0.10g maximum transient acceleration is to be employed for the DBE and 0.05g for the OBE.

Response Spectra

The response spectra employed in the seismic design of the plant are presented in Figs. A.1-1 and A.1-2. These response spectra are in accordance with the state-of-the-art applicable to the time that the PSAR and seismic design criteria were established, and on this basis are acceptable.

Damping

The damping values applicable to the design of the Indian Point 3 unit are presented in Table A.1-1 and when used in conjunction with the spectra noted are acceptable.

Seismic Analysis of Structures, Piping and Equipment

A general description of the procedures employed for seismic design is presented in Section 5 of the FSAR. The response spectrum approach was employed. It is indicated there that the containment structure was modeled as a simple cantilever in order to ascertain the moments and shear resulting from seismic excitation. Additional information concerning the details of the seismic analysis procedures is presented in the containment design report, specifically beginning on page 5A-26. Vertical seismic response and the effects of overturning were considered in the analysis.

For items other than the major structures, the general procedure employed in the dynamic analysis is described in Appendix A beginning on page A.3-10. It is indicated there that all Class I piping 6 inches in diameter or larger, together with the 2-inch diameter high-head safety injection lines, were dynamically analyzed for seismic response. Additional information is presented in the answer to Question 5.16, where there is listed for Class I piping and other auxiliary equipment the specific methods of analysis which were employed in the design. It is noted there and in the answer to Question 5.21 that equivalent static coefficients were used for the analysis design of piping less than 6-inch diameter. The answer to Question 5.36 states that the use of equivalent static coefficients is only employed for piping and equipment items after it has been demonstrated that such an approach, when checked against rigorous dynamic analyses, gives conservative results. This approach is in accordance with the state-of-the-art applicable to this design.

The answer to Question 5.20 indicates that floor response spectra were employed in the design of equipment and piping and the general approach analyzed in derivation of the floor response spectra is described in the Answer to Question 4.32.

Buried Piping

The design criteria applicable generally to buried piping or other piping located outside the containment structure appear on page A.3-9 and again in the Answer to Questions 5.19 and 5.35. On the assumption that the design approach did consider the problem of providing adequately for stresses and deformations at support points as suggested in the Answer to Question 5.35, We believe the approach to be adequate.

Design Stresses

The design stress approach employed for Class I structures is described in Section 5, and the stress tabulations presented in the containment report, Section 5A, are helpful in demonstrating the adequacy of the design approach employed for Class I structures.

For piping, the procedures associated with techniques outlined in Topical Report WCAP-7287 were employed, but the Answer to Question 4.29 indicates that only elastic analyses were used with the cited stress limits. This approach is in line with the state-of-the-art applicable to this design.

Class I Controls and Instrumentation

The general procedures to be employed in the design and review of critical controls and instrumentation are presented in the Answer to Question 5.29. On the assumption that criteria of the type described in Report WCAP-7397-L and Supplements thereto are applicable, we believe that the design procedures adopted for the critical controls and instrumentation will be acceptable.

REFERENCES

1. "Final Facility Description and Safety Analysis Report -- Indian Point Nuclear Generating Unit No. 3, Consolidated Edison Company of New York, Inc., Vol. 1-6 and Amendments 14-16, 19-21, 23 and Supplements 10 and 11", AEC Docket No. 50-286, 1971-72.

2. Newmark, N. M., W. J. Hall and A. J. Hendron, "Adequacy of the Structural Criteria for Indian Point Nuclear Generating Unit No. 3, Consolidated Edison Company of New York, Inc.", AEC Docket No. 50-286, 20 Dec. 1968.

3. Newmark, N. M. and W. J. Hall, "Report to the AEC Regulatory Staff -- Structural Adequacy of Indian Point Nuclear Generating Unit No. 2, Consolidated Edison Company of New York, Inc.", AEC Docket No. 50-247, August 1970.

WJHall

APPENDIX B

NATHAN M. NEWMARK
CONSULTING ENGINEERING SERVICES

1114 CIVIL ENGINEERING BUILDING
URBANA, ILLINOIS 61801

12 February 1973

STRUCTURAL ADEQUACY
OF THE
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
AEC Docket No. 50-286

by

W. J. Hall and N. M. Newmark

After our review of the FSAR, including Supplements 1, 2, 5, 6, 7, 8 and Amendments 15, 16, 22, it is believed that the design of the Indian Point Nuclear Generating Unit No. 3 can be considered adequate in terms of provisions for safe shutdown for a Design Basis Earthquake of 0.15g maximum transient horizontal ground acceleration and capable otherwise of withstanding the effects of an Operating Basis Earthquake of 0.10g maximum horizontal ground acceleration.

Our review was based on consideration, among other things, of the design criteria and results of the analysis presented by the applicant for the foundations and the seismic design criteria including seismic hazard, response spectra, damping, seismic analysis, buried piping, design stresses, Class I controls and instrumentation.

We believe that the procedures used in the design and analysis are in accord with the state-of-the-art. It is our conclusion that the design incorporates an acceptable range of margins of safety for the hazards considered.

W J Hall

Appendix C

Report of the Department of the Army

Coastal Engineering Research Center

DEPARTMENT OF THE ARMY
COASTAL ENGINEERING RESEARCH CENTER
5201 LITTLE FALLS ROAD, N.W.
WASHINGTON, D.C. 20016

CEREN-DE

28 March 1973

RECEIVED
APR 4 1973
U.S. ATOMIC ENERGY COMMISSION
Regulatory
Mail Section

Dr. J. M. Hendrie
Dep. Director for Technical Review
Directorate of Licensing
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Dr. Hendrie:

Reference is made to your letters regarding Docket Nos. 50-247, 286, 342 and 343, Consolidated Edison Company of New York's Indian Point Nuclear Generating Units 2, 3, 4 and 5, and our letter dated 21 November 1969.

Pursuant to our arrangements, Mr. R. A. Jachowski of my staff has continued to review all information pertaining to the application for an operating license for Unit 3 (Docket No. 50-286) and to advise your staff on the acceptability of the applicant's implementation of the design bases still water level in which we have previously concurred in the referenced letter. Our review has included consideration of the storm surge associated with Probable Maximum and Standard Project Hurricanes, and wind-generated wave analyses associated with severe water levels.

We agree with your staff that wind-generated wave activity associated with severe water levels such as that resulting from the PMH surge could exceed plant grade in the vicinity of the intake structures by several feet, and that with appropriate emergency procedures should be developed so as to protect essential structures from flooding.

Sincerely yours,

THORNDIKE SAVILLE, JR
Acting Director

Appendix D

Evaluation of Financial Qualifications

FINANCIAL QUALIFICATIONS

The Commission's regulations which relate to the financial data and information required to establish financial qualifications for applicants for operating licenses are 10 CFR 50.33(f) and 10 CFR 50, Appendix C. The basic application of Consolidated Edison Company of New York (Con Ed), Amendments No. 9, 12, 27, the amendment of April 13, 1973, and the accompanying certified annual financial statements of the applicant provide the financial information required by the Commission's regulations. This information includes the estimated annual costs of operating the Indian Point Nuclear Generating Unit No. 3 for a five-year period plus the estimated cost of permanently shutting down the facility and maintaining it in a safe shutdown condition.

Our evaluation of the financial data submitted by the applicant, summarized below, provides reasonable assurance that the applicant possesses or can obtain the necessary funds to meet the requirements of 10 CFR 50.33(f) to operate the Indian Point Nuclear Generating Unit No. 3, and if necessary permanently shut down the facility and maintain it in a safe shutdown condition.

Indian Point Nuclear Generating Unit No. 3 will be used to augment the applicant's present electrical generating capacities. Operating revenues will provide the funds to cover cost of operations. The costs of operating for the five-year period 1975-79 are presently estimated by the applicant to be (in millions of dollars) $72.4; $74.1; $71.7; $72.5; and $73.5 in that order. These costs include amounts for operation and maintenance, fuel, insurance, depreciation, interest on investment, and taxes. In addition, the applicant estimates that (based on 1973 cost levels and technology) the cost of permanently shutting down the facility will be approximately $3 million, and an annual cost of $300,000 will be incurred to maintain the facility in a safe shutdown condition. Funds for permanent shutdown will come from retained earnings and funds to maintain the facility in a safe shutdown condition will be provided by future operating revenues.

We have examined the financial information submitted by Con Ed to determine whether it is financially qualified to meet the above estimated costs. The information contained in Con Ed's calendar year 1972 financial report indicates that operating revenues for 1972 totaled $1,479.9 million; operating expenses were $1,244.6 million, of which $112.3 million represented depreciation. The net income for the year was $148.1 million, of which $134.8 million was distributed as dividends to stockholders and the remainder of $13.3 million was retained for use in the business. As of December 31, 1972, the Company's assets totaled $5,262.0 million, most of which was invested in utility plant ($4,840.6 million); retained earnings amounted to $546.9 million. Financial ratios computed from the 1972 statements indicate an adequate financial condition, e.g., long-term debt to total capitalization - .51, and to net utility plant - .53; net plant to capitalization - .97;

the operating ratio - .84; and the rates of return on common - 6.4%, on stockholders' investment - 6.0%, and on total investment - 5.4%. The record of Con Ed's operations over the past 5 years reflects that operating revenues increased from $930.8 million in 1967 to $1,479.9 million in 1972; net income increased from $122.9 million to $148.1 million; and net investment in plant from $3,433.2 million to $4,840.6 million; while the number of times interest earned declined from 2.7 to 2.1. Moody's Investors Service rates the Company's first mortgage bonds as A (upper medium grade). The Company's current Dun and Bradstreet credit rating is 5A1.

A copy of our financial analysis of the company reflecting these ratios and other pertinent financial data is attached as an appendix.

CONSOLIDATED EDISON COMPANY OF NEW YORK
DOCKET NO. 50-286
FINANCIAL ANALYSIS

	(dollars in millions) Calendar Year Ended December 31		
	1972	1971	1970
Long-term debt	$2,543.1	$2,408.1	$2,256.6
Utility plant (net)	4,840.6	4,424.8	4,106.8
Ratio - debt to fixed plant	.53	.54	.55
Utility plant (net)	4,840.6	4,424.8	4,106.8
Capitalization	4,999.3	4,657.6	4,242.1
Ratio of net plant to capitalization	.97	.95	.97
Stockholders' equity	2,456.2	2,249.5	1,985.5
Total assets	5,262.0	4,888.2	4,448.9
Proprietary ratio	.47	.46	.45
Earnings available to common equity	108.4	160.4	94.2
Common equity	1,705.2	1,573.3	1,309.1
Rate of earnings on common equity	6.4%	10.2%	7.2%
Net income	148.1	198.6	128.4
Stockholders' equity	2,456.2	2,249.5	1,985.4
Rate of earnings on stockholders' equity	6.0%	8.8%	6.5%
Net income before interest	284.3	317.9	234.6
Liabilities and capital	5,262.0	4,888.2	4,448.9
Rate of earnings on total investment	5.4%	6.5%	5.3%
Net income before interest	284.3	317.9	234.6
Interest on long-term debt	134.7	118.6	105.5
No. of times long-term interest earned	2.11	2.68	2.22
Net income	148.1	198.6	128.4
Total revenues	1,528.9	1,403.3	1,152.5
Net income ratio	.10	.14	.11
Total utility operating expenses	1,244.6	1,085.4	917.9
Total utility operating revenues	1,479.9	1,313.9	1,128.5
Operating ratio	.84	.83	.81
Utility plant (gross)	5,918.2	5,480.2	5,093.2
Utility operating revenues	1,479.9	1,313.9	1,128.5
Ratio of plant investment to revenues	4.00	4.17	4.51

	1972		1971	
Capitalization:	Amount	% of Total	Amount	% of Total
Long-term debt	$2,543.1	50.9%	$2,408.1	51.7%
Preferred stock	751.0	15.0	676.2	14.5
Common stock & surplus	1,705.2	34.1	1,573.3	33.8
Total	$4,999.3	100.0%	$4,657.6	100.0%

Moody's Bond Rating: A
Dun & Bradstreet Credit Rating: 5A1