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Application for Licenses

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BEFORE THE UNITED STATES

ATOMIC ENERGY COMMISSION

In the Matter of	)	
	)	
Consolidated Edison Company	)	Docket No. 50-247
of New York, Inc.	)	
(Indian Point Unit No. 2)	)	

SUMMARY OF APPLICATION

November 12, 1970

SUMMARY OF APPLICATION

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SUMMARY OF APPLICATION

I. INTRODUCTION

This document is a summary of the Application for Licenses, as amended to date, submitted to the Atomic Energy Commission ("Commission") by Consolidated Edison Company of New York, Inc. ("Con Edison" or "Applicant") for a nuclear powered generating unit known as Indian Point Unit No. 2 ("Unit No. 2"). The summary is prepared according to the Statement of General Policy set forth in Appendix A to the Commission's Rules of Practice (10 CFR Part 2, Appendix A), and it evaluates, among other things, the considerations important to the safety of the facility.

Unit No. 2 is designed to provide 873,000 net kilowatts of electric power for the Con Edison system. It is located on Con Edison's Indian Point site on the Hudson River, and is adjacent to and to the north of Unit No. 1, a nuclear powered generating unit of 265 megawatts net electrical capacity, operating since 1962. A third nuclear unit, under construction at this site, is scheduled for completion in 1973. All three units use reactors of the pressurized water type.

Con Edison filed its Application for Licenses for Unit No. 2 with the AEC in December, 1965, and received a construction permit in October, 1966 (Construction Permit No. CPPR-21). Construction of the unit has proceeded and Con Edison now seeks authority to operate the facility. This summary therefore relates to portions of the application which support Con Edison's request for operating authorization.

The design of the facility and other technical matters relating to its operation are fully described, analyzed and evaluated in the Final Facility Description and Safety Analysis Report ("FSAR"), and Supplements 1 through 15 thereof, filed herein as part of the Application as amended.

Among the matters covered by the FSAR and summarized in this document are studies of the site and environment of the facility; analyses of the radiological effects of the facility upon the environment under various hypothesized accident conditions, as well as for normal operating conditions; discussions of research and development programs carried out since the construction permit was issued; the technical and financial qualifications of the Applicant; technical specifications; conduct of operations (including emergency planning); and the common defense and security.

Some of the terminology appearing in the Application has been simplified or explained in this document, and some of the information appearing in various portions of the Application has been combined and characterized. The principal objective is to inform the Board and the public, in as non-technical terms as feasible, of the considerations which support the conclusion that the facility can be operated without undue risk to the health and safety of the public.

For members of the Atomic Safety and Licensing Board and others who wish to study in more detail subjects mentioned in this summary, appropriate references are provided in Appendix B.

II. SITE AND ENVIRONMENT

The Applicant owns a tract of land of approximately 239 acres called "Indian Point", located on the Hudson River in the village of Buchanan, Westchester County, New York. Indian Point is about 24 miles north of the New York City boundary. Unit No. 2 has been built on this site adjacent to and north of the Applicant's existing Unit No. 1. The site is shown on the map attached as Appendix A to this document. 1/

The FSAR contains data on both the present and projected populations within the area circumscribed by a 55-mile radius from the site. Based upon the 1960 census, about 55,000 people live within five miles of the site. Most of them live northeast of the site. In 1960, the population within 15 miles of the site was 326,930. Within this larger radius most people are located south of the site. The population projected to 1980 for this radius is about 670,000. 2/

The general area surrounding Indian Point is zoned principally for residential and state park usage although there is some industrial activity and a little agricultural and grazing activity. 3/

The Indian Point site consists geologically of a fine-grained phyllite, schist, and limestone with bedrock lying very close to the surface. Unit No. 2 is located on limestone which is hard although jointed. The bedrock will support any foundation loads up to 50 tons per square foot. This capacity far exceeds any load that this plant will impose on the bedrock. It will therefore provide a firm foundation for the facility. Ground water in the vicinity of the plant flows to the river; therefore, liquids accidentally spilled on the site would not contaminate ground water supplies. [4/]

According to Applicant's consultant on seismology, the Indian Point site is located in one of the safest areas relative to earthquake activity, both as to historical incident and the probability of future occurrences. This consultant has advised that the probability of a serious shock occurring in the area of this site within the next several hundred years is practically nonexistent. The highest intensity recorded in this area is the equivalent of a horizontal acceleration much less than 0.10 g. [5/] All systems and structures in the plant whose failure could cause or increase the severity of a loss of coolant accident, which could result in an uncontrolled release of excessive amounts of radioactivity, or which are vital to safe shutdown and isolation of the reactor are capable of performing their safety functions during and after an earthquake having a horizontal acceleration of 0.15 g and a simultaneously acting vertical acceleration of 0.10 g. [6]

The highest recorded water elevation at the site is 7.4 feet above mean sea level. The Applicant has performed a comprehensive analysis of possible flooding conditions at the site. Various possible combinations of floods, dam failures and hurricanes were considered. In all cases analyzed, the calculated flood-water level at the plant was below the 15-foot 3-inch level required to flood any plant structure. [7]

The site is in an area of very low tornado incidence and is surrounded by hills which provide natural protection against tornadoes. Nevertheless, the ability of the plant to withstand tornadoes has been investigated and is discussed in the FSAR. [8]

During construction of Indian Point Unit No. 1, New York University conducted a 2-year detailed study of the meteorological conditions at the site. This study was supplemented by data from the National Weather Records Center at Bear Mountain Weather Station, which was located approximately 3 miles north of the site. The most important meteorological characteristic of the site is that the winds are diurnal in nature; that is, the winds blow down the valley during the night and up the valley during the day. This is a result of the orientation of the ridges in the Hudson valley. These predominant winds occur at all altitudes within the valley and under all meteorological conditions. [9]

Atmospheric diffusion calculations based on data gathered in the above-mentioned surveys have been made to determine the rate of release of gaseous radioactivity which would result in an average concentration at the worst point on the land bounding the site equal to the limits set by Part 20 of the Commission's regulations. The technical specifications for Unit No. 2 plant require that the combined releases from Units 1 and 2 be at all times less than that calculated release rate. [10]

The data from the surveys also show that the meteorological conditions assumed for the calculations of doses from postulated accidents are conservative since periods of calm are infrequent and wind does not persist in any one direction for long periods of time. [11/]

The releases from Units 1 and 2 into the Hudson River will be below the Part 20 allowable concentrations at all times, as required by the technical specifications. [12/] Studies show that even for continuous release into the Hudson River at Part 20 concentrations during drought conditions, concentrations at Chelsea Pumping Station and the Castle Point Veteran's Hospital (the nearest sources of drinking water) would be far below those permitted by Part 20. The river flow at Indian Point is primarily the result of tides; therefore, even during periods of drought, excellent mixing is provided. [13/]

A comprehensive environmental monitoring program is being carried out by Con Edison in the vicinity of the Indian Point site. This program began with a survey of radioactivity in the vicinity of Indian Point instituted in 1958, four years prior to the startup of Indian Point 1. It has continued to the present and will continue throughout the operating lifetimes of Units 1, 2 and 3. The program is designed to provide comparative information on radioactivity in the environment both before and after the beginning of operation of each unit, so that the effects of their operation, if any, may be determined. This data supplements the primary control at the source of the effluents which assures compliance with Part 20 of the Commission's regulations. The results of these surveys are currently reported semi-annually to the AEC under Docket 50-3. 14/

The present environmental program includes continuous sampling of atmospheric dust, Hudson River water, and measurement of the gross gamma background on the Indian Point site. Surface water from a small lake on the site and drinking water from the Indian Point tap and nearby reservoirs are sampled weekly. The program includes samples of milk, Hudson River water upstream and downstream from the site, vegetation on the site, marine life from the river, and water from the Indian Point well. Vegetation and soil samples are taken regularly during the growing season and gamma radiation surveys on roads in the station vicinity are made. 15/

The New York State Department of Health has been conducting surveys in the vicinity of Indian Point since 1958 as part of its state-wide radiological survey program. In addition, the New York University Medical Center recently completed an ecological survey of the Hudson River which included extensive studies of radioactivity in the river and in the river biota. 16/

The Applicant expects that Unit No. 2 will have no deleterious radiological effect on the environment. 17/

III. DESCRIPTION OF FACILITY AND ASSOCIATED PLANT FEATURES

The FSAR describes the reactor, its components and related systems and features which are essential for safe operation. This portion of the summary describes briefly those systems most relevant to public health and safety, highlighting the features which are of greatest importance and those of special interest.

A. Reactor and reactor coolant system

Unit No. 2 contains a pressurized water reactor with an initial rating of 2758 megawatts thermal and 873 megawatts net electric.[1] The reactor will operate at a nominal pressure of 2250 psia and an average temperature of 569.5 degrees F.[2]

The reactor core is approximately eleven feet in diameter and twelve feet in height. It will contain 193 fuel assemblies, each containing 204 fuel rods in square array and held in place by grids. These fuel rods are manufactured from Zircaloy tubes and loaded with fuel pellets of slightly enriched uranium dioxide. Placed in each fuel assembly are 21 guide tubes. These guide tubes provide structural integrity and positioning for control rods, fixed burnable neutron absorbers and in-core instrumentation.[3]

Core reactivity will be controlled by a combination of fixed burnable neutron absorbing rods, movable absorber rods and neutron absorber (boric acid) dissolved in the coolant. The movable absorber rods are grouped in clusters and used for short-term reactivity changes, such as those accompanying unit load changes. Some of the rods are full-length and others part-length, the latter being available to control axial xenon oscillations should they occur. Upon reactor trip the full-length rod cluster control (RCC) rods fall into the core by gravity. These assemblies can shut down the reactor at any time during power operation, even if any one rod cluster fails to fall.[4]

The boric acid concentration in the reactor coolant will be changed to compensate for reactivity changes associated with fuel depletion and build-up and decay of fission products xenon and samarium. It will also be used to keep the reactor subcritical when the reactor is cooled down and to provide a safe shutdown margin during refueling.[5/]

The reactor vessel is a steel cylinder with a hemispherical bottom and a hemispherical top which is removable to permit refueling and inspection. All interior surfaces of the vessel are clad with corrosion-resistant stainless steel.[6/] The vessel and its internals are designed to permit removal of the internals for inspection of the internals and the reactor vessel during plant life.[7/] The internals will remain in a safe condition under the combined effects of the hypothetical earthquake and of a loss-of-coolant accident.[8/] A surveillance program will be instituted to ascertain the effect of radiation on the reactor vessel material with samples of the vessel material that will be placed within the vessel. This program will verify design margins and mechanical properties of the vessel.[9/]

There are four cooling loops which will be used to carry the heat from the reactor. In each loop reactor coolant will be pumped through a stainless steel piping system to a steam generator and then back to the reactor.[10/] As the coolant passes through tubes in the steam generators, the tubes are heated and in turn boil water which produces steam for the turbine generator.[11/]

Connected to one of the loops, a surge tank, approximately half filled with reactor coolant, acts as a pressurizer to control reactor system pressure.[12/] The pressure is maintained by a combination of electric immersion heaters to raise pressure and a spray of reactor coolant to lower pressure.[13/]

All materials and components which form a part of the reactor coolant system pressure boundary meet or exceed the requirements of applicable codes[14/] and together with their supports can withstand the combined effects of the hypothetical earthquake and of a loss-of-coolant accident.[15/]

B. Containment

The containment for Unit No. 2 consists of a massive reinforced concrete structure lined with steel plate. The containment completely encloses the reactor and reactor coolant system and is intended, together with associated engineered safeguards described below, to contain any radioactive material which might accidentally be released from the reactor coolant system.[16/]

The containment structure is a flat bottomed cylinder with a hemispherical dome with an inside diameter of 135 feet and vertical sidewalls of 148 feet. The base is nine feet thick, the side walls are 4-1/2 feet thick, and the dome is 3-1/2 feet thick. The inside of the containment is lined with steel plate.[17/]

Pipes which penetrate the containment are provided with valves that serve to isolate the containment atmosphere should this be required. To assure that no leakage of containment atmosphere will occur through these valves, an isolation valve sealing system is provided.[18] Also included is a containment penetration and weld channel pressurization system which, by using double barriers on all the containment penetrations, doors and liner welds with continuous pressurization of the space between the barriers, will assure an essentially leak-tight containment system.[19]

Prior to operation of the facility, the containment will be tested for structural integrity and leak tightness. The structural integrity test will be conducted at 115% of design pressure.[20] The structure will be tested at design pressure to establish that the leak rate of the containment structure is less than 0.1% of the free volume per day, even with the penetration and weld channel pressurization system open to the containment atmosphere.[21]

The containment structure can withstand various loading combinations, including those associated with the combined effects of an earthquake and the most severe loss-of-coolant accident.[22] Missiles generated either by a tornado or by a turbine-generator failure will not penetrate the containment structure.[23] Protection is also provided against missiles which might be generated from the reactor coolant system.[24] The ability of the containment and associated safeguards to contain fission products resulting from various postulated accidents is discussed in Section VI. below.

C. Engineered safeguards

In addition to the pressurization system for containment penetrations and liner weld channels and the isolation valve sealing system described above, the following engineered safeguards are incorporated for the protection of the public in the unlikely event of an accident:

1. A safety injection system, which in the event of a loss-of-coolant accident provides borated water to cool the core and thus limits damage to the reactor core and also limits the energy and fission products released from the reactor into the containment. The system includes four accumulator tanks, a boron injection tank, a refueling water

storage tank, three high-head safety injection pumps, two low-head residual heat removal pumps and two low-head recirculation pumps. [25]

2. A containment spray system, which in the event of a loss-of-coolant or steam-break accident inside the containment, provides a spray of cool, chemically treated borated water to the containment atmosphere to reduce the pressure inside the containment and to remove elemental iodine. [26]

3. A containment air recirculation cooling and filtration system is used for cooling the containment atmosphere during normal operation. This system also serves to limit the pressure inside the containment following a loss-of-coolant or steam line break accident. It is a self-contained system equipped with cooling coils, demisters, roughing filters, absolute filters, fans and a bypass loop with charcoal filters. Under design-basis accident conditions the bypass flow through the charcoal filters is capable of removing organic iodides from the containment atmosphere. [27]

4. Redundant hydrogen recombiners which, following a loss-of-coolant accident, can be used to prevent hydrogen from building up in the containment atmosphere.[28/] As a backup to this system a containment venting system will be installed within the first two years of operation.[29/]

D. Instrumentation and control

The facility is equipped with a central control room which contains all controls, alarms and instrumentation displays necessary for the safe startup, operation, and shutdown of the plant, as well as for the detection and control of accident situations. The control room is designed to be occupied on a continuous basis, even under accident conditions.[30/]

The reactor control system provides for control of reactor power during plant startup, operation and shutdown. The system provides for automatic or manual control and is specifically designed to assure that the reactor power level changes in response to changing demand for electric power while maintaining all plant conditions within prescribed operational limits. Other plant control systems include the chemical and volume control system, the steam and feedwater flow control system, the turbine and turbine bypass controls and the pressurizer heaters and sprays.[31]

The reactor protective system provides for protection of the reactor core and coolant systems during all phases of plant operation. The protective system shuts down the reactor power output by dropping the control rod assemblies into the core and starting any engineered safeguards that may be required to maintain a safe condition. The reactor protective system is activated by signals from process sensors such as nuclear power and reactor coolant system temperature, pressure, level and flow, if the prescribed safety limits are approached.[32]

The protective system is separate from the control system, except that some process sensors are used for both control and protection. Where the process signals from one sensor are used for both control and protection, electrical isolation is used to insure that no failure on the control system side of the isolation device can affect the protective system.[33]

Studies performed by the Applicant at the request of the Commission's Advisory Committee on Reactor Safeguards have shown that even if the reactor protective system should fail to drop the control rods following anticipated transients such as loss of load or reactor coolant pump failure, the resulting situation could be tolerated without primary coolant system failure.[34]

A large number of different and highly reliable instruments are included in the plant to provide information to the reactor operator and to the control and protective systems. These instruments are described in detail in the FSAR.[35] All the instruments and electronics associated with the reactor protective system meet the requirements of the Institute of Electronics and Electrical Engineers Criteria for Nuclear Power Plant Protection Systems (IEEE279).[36]

E. Electrical supply

Unit No. 2 auxiliary and emergency A-C power is provided from the following different sources:

1) The main generator

2) A 138 kv overhead line from Buchanan Substation

3) A 13.8 kv underground feeder from Buchanan Substation

4) A 21 megawatt on-site gas turbine generator

5) Three on-site emergency diesel generators

In addition, emergency power for instruments and control is provided by two independent 125 v. D-C station batteries.

The main source of normal auxiliary power during plant operation is the main generator of the unit itself. Power will be supplied by way of a unit auxiliary transformer that is connected to the main generator.

A portion of normal auxiliary power as well as the standby power required during plant startup, shutdown, and after reactor trip will be normally supplied through the 138 kv overhead line from the Buchanan Substation.

A backup to the 138 overhead line is provided by a 13.8 kv underground feeder. The 13.8 kv feeder also runs from the Buchanan Substation but is supplied from a different bus at Buchanan than the 138 kv line and supplies the plant by way of a different transformer.

The 13.8 kv and 138 kv lines both originate from the 138 kv Con Edison system. The output of Unit No. 2 feeds directly into Con Edison's 345 kv system. Studies have been made to show that the 138 kv and 13.8 kv supplies are stable (that is, they remain fully available) following loss of any single generating unit on the Con Edison system, including Indian Point 2.

There is a 21 megawatt gas turbine generator on the Indian Point site. The output from this generator feeds through the same transformer as the 13.8 kv underground feeder from Buchanan.

In the event that power from all the above on-site and off-site sources is lost, emergency power will be immediately available from three on-site emergency diesel generators. Any two of the three diesels can carry all loads required for safety under normal or accident conditions. Each diesel has two independent starters. All diesels are signalled to start automatically on either loss of power or engineered safeguards actuation. Each diesel is completely independent and has no automatic interconnections.[37]

F. Waste disposal

Unit No. 2 contains a number of facilities for treatment and disposal of liquid, gaseous, and solid radioactive wastes. These facilities are designed to insure that the discharge of radioactive effluents and shipments of solid radioactive wastes are in accordance with applicable governmental regulations.

The bulk of the radioactive water discharged from the reactor coolant system is processed and retained inside the plant by the chemical and volume control system. This minimizes liquid input to the waste disposal system which processes relatively small quantities of generally low activity level wastes.[38]

Radioactive water which may leak from the reactor coolant system and other equipment, along with small amounts of other liquid wastes such as laboratory samples, is collected in tanks where it is sampled and analyzed to determine the quantity of radioactivity present, with an isotopic breakdown if necessary. It is then processed before release to the cooling water discharge.

Processing is done on a batch basis; that is, one tank is processed while another is being filled. The radioactive water is evaporated thus leaving most radioactivity behind as solid residue which is removed for drumming and shipment off site. The evaporated water is condensed and held in tanks where it is again sampled before it is released under controlled conditions to the cooling water discharge. Provision is made for reprocessing of the purified water if further reduction of activity is required. The discharge lines are monitored and activity recorded. If for any reason the effluent exceeds specified levels an automatic cutoff is provided as well as an alarm.

During plant operation some radioactive gases which may be dissolved in the water in various systems are removed, collected and pumped by compressors to decay tanks where they are held prior to release. Samples are periodically taken from the tanks to determine when the radioactivity is low enough for release. An alarm warns the operator if activity in these tanks is too high. The radioactive gases in those tanks are discharged through the plant vent. There are three monitors in the discharge line - two for radioactive gases and one for particulates. There is also an automatic cutoff on this system to prevent inadvertent releases.

All components of the liquid, gaseous and solid radioactive waste systems which contain significant radioactivity are located in the containment or the primary auxiliary building. The atmosphere in the primary auxiliary building is continuously exhausted through filters to the plant vent by way of the monitors mentioned above. This insures that any gases which might leak from any tank are collected and monitored.

The containment system has recirculating fans and filters which reduce the radioactivity in the containment air before this air is exhausted to the plant vent. Any radioactive liquids which leak or are spilled in either the containment or primary auxiliary building are collected in drains and piped to the liquid waste system.

Solid wastes may be accumulated in several ways: resins which are used to purify plant water become radioactive through use; solid wastes are collected in the waste evaporators which are part of the liquid waste system; and solids such as rags and laboratory equipment may become contaminated. A drumming area is provided for the preparation of solid wastes for disposal off site. Solid wastes are packaged in suitable containers and stored on site until shipped off site for disposal.

In addition to the monitors provided on tanks and discharge lines, monitors are provided in various areas throughout the plant to warn of leakage or spills and to permit appropriate operator action. [39/]

With all equipment operating as designed and with the maximum expected fuel failure of 1%, it is estimated that gaseous radioactivity released from Unit 2 will be about 1.5% of that allowed by Part 20 while liquid releases will be less than .1% of allowable. [40/]

As is presently the case for Unit No. 1, it will be Con Edison's policy to operate Unit No. 2 including the waste disposal equipment, in such a manner that radioactive releases will be kept as low as practicable below the limits specified in Part 20 of the Commission's regulations. Available waste disposal equipment will be utilized to the maximum extent practicable, and every effort will be made to restore the availability of equipment which may be shut down for servicing or repair.

G. Auxiliary and emergency systems

The auxiliary and emergency systems support the reliable operation and servicing of the reactor coolant system. The most important of these systems are:

1) Chemical and volume control system
2) Closed loop auxiliary coolant systems
3) Service water system
4) Fuel handling system
5) Sampling system
6) Primary and auxiliary building ventilation system
7) Control room heating and ventilation system

The chemical and volume control system adds and removes boron in the reactor coolant to control the reactivity of the core. The system also adds chemicals to the reactor coolant to control corrosion, removes gases from the reactor coolant for treatment by the radioactive waste system, adds additional water to the reactor coolant system as required, continuously cleans the reactor coolant water, and provides seal water for the reactor coolant pump seals. [41/]

The closed loop auxiliary coolant systems transfer heat from various parts of the plant to the service water system. They remove heat from the core and reduce the temperature of the reactor coolant system following normal reactor shutdown; heat generated by stored spent fuel elements in the spent fuel pit; heat from the letdown flow heat exchangers; and heat from various primary plant and other components. 42/

The service water system supplies cooling water from the river intake to remove heat loads from both the primary and secondary portions of the plant. A continuous flow of cooling water is provided to those systems and components necessary for plant safety either during normal operation or under abnormal and accident conditions. 43/

The fuel handling system provides a means of safely transporting and handling fuel from the time the fuel reaches the plant in an unirradiated condition to the time it is cooled after irradiation and is removed from the plant. The reactor is refueled under water by equipment designed to provide careful handling of the spent fuel assemblies, from the time it leaves the reactor until it is placed in a cask and shipped from the site. The water surrounding the spent fuel assemblies during their transfer provides an effective, transparent radiation shield, as well as a reliable cooling medium for removal of the residual heat. [44/] The building storing the irradiated fuel is equipped with a ventilation exhaust system. Filtration equipment will be added to the ventilation exhaust system during the first year of operation to reduce radioactive particulates and halogens in the event of a fuel handling accident. [45/]

The sampling system provides the equipment necessary to obtain liquid and gaseous samples from the reactor plant system. [46/]

The primary auxiliary building ventilation system maintains operating temperatures in the primary auxiliary building and provides for exhausting of the primary auxiliary building atmosphere to the plant vent. [47/]

The control room heating and ventilation system maintains a proper environment in the control room, for both personnel and vital instrumentation systems and equipment. [48/]

H. Steam and power conversion system

The steam and feedwater system is designed to remove heat from the primary coolant system and uses it to drive the turbine. Two steam driven feedwater pumps supply water to the four steam generators. The steam then passes to the turbine which drives the generator. From the turbine, the steam is quenched in three condensers and returned to the feedwater pumps by three motor driven condensate pumps. The condensers are cooled by six circulating water pumps which draw water from the Hudson River. [49/]

The turbine is a three-stage 1800 rpm. tandem unit, comprised of one high pressure and three low pressure cylinders. Six combination moisture separator-reheater units are employed to dry and superheat the steam between the high and low pressure turbine cylinders. [50/] The turbine-generator is capable of a 50% loss of external electrical load without turbine or reactor trip. [51/] The turbine is equipped with a redundant overspeed protection system to prevent it from reaching excessive speeds in the event that power produced in the primary system exceeds the electrical load. [52/]

IV. EVOLUTION OF DESIGN OF UNIT NO. 2 FROM DESIGN OF OTHER FACILITIES

This section describes the evolution of the design of the Indian Point Unit No. 2 facility and its associated engineered safeguards, from those of other facilities previously approved by the Commission for operation. [1/] The section also refers to reactors of the same "generation" as Unit No. 2 which have been approved for construction but which will probably not receive operating authorization before Unit No. 2.

A. Reactor

Unit No. 2 uses a pressurized water reactor. This reactor type has demonstrated successful and safe operation in electric power plant service beginning with the 239 MWt Shippingport plant in 1957. Yankee-Rowe began operation in 1961 at 392 MWt and is now licensed at 600 MWt. Consolidated Edison began operating Indian Point Unit No. 1 in 1962 at 585 MWt and in 1965 increased the rating to 615 MWt. More recent pressurized water reactors are San Onofre (1347 MWt), which began operation in 1967, Connecticut Yankee (1825 MWt) which began operation in 1967, and Ginna (1300 MWt) which began operation in 1969. In addition, Point Beach (1518 MWt) and H. B. Robinson Unit No. 2 (2200 MWt) have both received operating authorizations in 1970, and have successfully achieved criticality.

Indian Point Unit No. 2 will have an initial power level of 2758 MWt. It is the first of a series of pressurized water reactors with comparable or higher power ratings which have been approved for construction. This series includes Indian Point Unit 3, Diablo Canyon Unit 1, Salem Units 1 and 2, Zion Units 1 and 2, D. C. Cook Units 1 and 2, and Sequoyah Units 1 and 2.

The evolution of nuclear reactors has been characterized by increases in linear heat rates of the fuel rods. The peak design linear heat rate for Indian Point Unit No. 1 is 12.1 kw/ft; Connecticut Yankee is 13.7 kw/ft; San Onofre is 15.0 kw/ft; Ginna is 16.5 kw/ft; Point Beach is 16.0 kw/ft and H. B. Robinson Unit No. 2 is 17.9 kw/ft. By comparison Indian Point Unit No. 2 is 18.4 kw/ft. The latter linear heat rate is comparable to those reactors currently under construction, such as Zion (18.9 kw/ft), and Diablo Canyon (18.9 kw/ft). In addition, the

Saxton reactor has been successfully operated at a peak design linear heat rate of 20.8 kw/ft.

Unit No. 2 will use zircaloy-clad uranium oxide fuel, which has been used in all Westinghouse power reactors approved beginning with Ginna. The system of reactivity control --dissolved neutron absorber and control rods--first was demonstrated in the Yankee-Rowe, Saxton, Trino Vercellesi (formerly known as SELNI) and SENA nuclear power plants and is used in Indian Point Unit No. 1. All central station Westinghouse pressurized water reactors since Yankee-Rowe have used this method of control.

Current Westinghouse pressurized reactors, beginning with San Onofre and Connecticut Yankee, employ rod cluster control assemblies which are designed to reduce power peaking and thus provide more favorable spatial power distribution. Part length control rods are included in the Unit No. 2 reactor to control axial xenon oscillations should they occur. This concept is also used on all Westinghouse reactors beginning with Ginna.

B. Reactor coolant system

The Unit No. 2 reactor coolant system is similar to that used in all other Westinghouse-designed reactors. The coolant system design uses a number of independent loops that provide appropriate heat removal capacity for each plant. Thus, Ginna (1300 MWt) uses two loops, H. B. Robinson (2200 MWt) uses three loops and Indian Point Unit 2 (2758 MWt), Diablo Canyon (3250 MWt), and Indian Point No. 3 (3025 MWt) will use four loops. Each loop contains a steam generator and pump which are similar in design features but which may vary slightly in design parameters to fit the plant operating characteristics.

C. Containment

The Unit No. 2 steel-lined, reinforced concrete containment is structurally similar to that used in Connecticut Yankee.

The isolation valve sealing system and the containment penetration and weld channel pressurization system provide added protection by assuring an essentially leak-tight containment system. These systems are additional to those usually provided.

They are also used in H. B. Robinson and will be used in Indian Point 3 and Zion.

D. Engineered safeguards

The engineered safeguards in Unit No. 2 are similar in principle to those used in the Ginna plant and in all other subsequent Westinghouse-designed plants which have been licensed.

In addition to the usual external recirculation system Indian Point No. 2 includes in its safety injection system two low head pumps located inside containment to recirculate cooled water to the core following a loss-of-coolant accident.

Charcoal filters are included in the Unit No. 2 containment air recirculation units to remove organic iodides from the post-accident containment atmosphere. Other plants such as Ginna also include charcoal filters, but generally those filters were included to reduce short-term leakage of all volatile forms of iodine. As spray technology progressed, it was shown that chemically treated containment spray provided a much faster means of removing all but the organic form of iodine. Because the amount of fission product iodine in the organic form is relatively small, its contribution to the accident consequences becomes important only after the other forms have been removed by the spray. The charcoal filters in the Unit No. 2 containment air recirculation units are relied upon to reduce the total leakage dose from organic iodides during the course of the accident.

Some previous plants have either installed (or made provision for) recombiners, or have made provision for venting systems, for the removal of hydrogen from the post-accident containment atmosphere. Both systems will be installed in Unit No. 2, as described in Section III. C. 4 above.

V. QUALITY ASSURANCE

The Applicant's quality assurance program for the design and construction of Unit No. 2 is described in the FSAR, as supplemented. [1] The program has been and is being carried out for all components, systems, and structures important for safety and all areas of activity affecting quality, including design (drawings and specifications), manufacture, field erection and installation, pre-operational testing, and related activities such as document control, cleanliness control, and shipment, storage and handling of components and equipment.

The program delineates the quality assurance responsibilities of each organization involved in the project, emphasizing the way the Applicant is assuring the quality of the completed product. The plant is being constructed under a "turnkey" arrangement, with Westinghouse having responsibility for design, engineering and construction.

Westinghouse in 1969 created a wholly owned subsidiary called "WEDCO" which has carried out Westinghouse's engineering and construction functions since that time. United Engineers & Constructors was selected by Westinghouse to provide architectural and engineering services and has continued to perform those functions under the direction of WEDCO. [2/]

The Applicant carries out its quality assurance responsibilities by: (1) insuring that Westinghouse, WEDCO and United Engineers & Constructors have adequate quality assurance programs and procedures; (2) monitoring the Westinghouse, WEDCO and United Engineers & Constructors' activities in critical areas through an independent detailed vendor surveillance program during manufacture of components; (3) carrying out a continuous on-site surveillance program; and (4) reviewing all engineering and safety analysis activities.

The United States Testing Company, as surveillance agent for Con Edison, visits manufacturing facilities, reviews test procedures and fabrication techniques, and witnesses selected tests. United States Testing Company's written reports of all surveillance visits are forwarded to Con Edison for review.

Con Edison maintains at the site a permanently assigned Superintendent of Construction (Resident Construction Manager) and his full-time staff whose prime function is to insure that on-site work is accomplished according to contractual requirements and on-site quality control programs.

The quality assurance program describes in detail the review procedures and surveillance programs carried out by Westinghouse, WEDCO and United Engineers & Constructors throughout the design and construction of the plant. The program also describes the internal organization of Westinghouse, WEDCO and United Engineers & Constructors and their relationship with Con Edison for quality assurance. Quality control and inspection records are maintained and will be permanently available to Con Edison.

Pre-operational testing and startup testing are important steps in assuring that the completed plant will operate as designed. These phases are discussed in Section X. below.

The technical specifications include a detailed program for inspection of the primary coolant system pressure boundary during the plant lifetime to assure the continued integrity of this boundary. [3/] In the event that any repairs must be made on the reactor coolant system pressure boundary in the future, the technical specifications require quality control testing equivalent to that done during the initial construction of the plant. [4/]

The technical specifications require frequent testing and maintenance of all critical plant components to be carried out during the life of the plant to insure that the high quality of the plant is maintained through its life. [5/]

VI. SAFETY ANALYSES

Con Edison and its contractors have analyzed the consequences of a variety of assumed abnormal operating conditions or equipment failure. For most of the situations analyzed the conclusion is that no radioactivity would be released from the plant. For the more severe postulated accidents, particularly those involving large breaks in the reactor coolant pressure boundary, the conclusion is that even with only partial effectiveness of the engineered safe-guards systems public exposure would be well within the guidelines set forth in Part 100 of the Atomic Energy Commission's regulations.[1]

Two general classes of accidents were considered: mechanical accidents and reactivity accidents. Of the former, the most severe is the postulated loss-of-coolant accident resulting from the rupture of a pipe in the reactor coolant system. This accident has been analyzed assuming rupture of various sizes of pipe up to and including a hypothetical doubleended rupture of the largest reactor coolant pipe. Loss of coolant is effectively controlled by normal action of the charging pumps for very small breaks. For larger breaks, reactor trip and safety injection are automatically initiated by high containment pressure or by the coincidence of both low water level and low pressure in the pressurizer. For the hypothetical rupture of the largest coolant pipe, injection of borated water ensures sufficient flooding of the core to limit core damage and any resulting energy-producing chemical reaction between the zirconium cladding and the coolant water. Even in this unlikely event compounded by failure of all external sources of electric power to the plant and the simultaneous occurrence of an earthquake, the facility with its emergency on-site power will be capable of protecting the public. [2/]

In this case the amount of fission products released in the containment would be small when there is full operation of the engineered safeguards on external power. Even with only on-site emergency diesel power, the fission product release is limited.

The containment isolation system and the pressurized penetration and weld channels essentially eliminate leakage to the environment after the accident. The calculated post-accident releases and off-site exposure levels for both of the above conditions are only a small fraction of the exposure guidelines given in 10 CFR 100. [3/]

In another calculation it was further assumed that a failure in the penetration and weld channel pressurization system or in the containment isolation valve sealing system permitted the design leak rate of the containment to exist and release fission products to the environment. It was further postulated that, concurrent with this accident, all external sources of electric power failed and only those safeguards would function which are operable from two of the three on-site diesel-generator units. Even under such extremely improbable conditions, the calculated exposures of the public will still be within the guidelines of 10 CFR Part 100. [4/]

Other mechanical accidents which would have a potential for off-site exposure include the steam generator tube rupture, the secondary system steam line break, a failure in the gaseous waste disposal system, rupture of the volume control tank, and a fuel handling accident. For these assumed accidents as well, potential off-site exposure is well below the 10 CFR 100 guidelines. 5/

Of the reactivity accidents the only one in which some fuel damage could occur is the highly unlikely rod ejection accident. In this hypothetical accident rapid withdrawal of a control rod is assumed to result from a rupture of a control rod drive mechanism housing, after which a rod control cluster assembly would be ejected from the core in a very short time by the system pressure.

The resultant power pulse following a rod ejection accident is limited by the Doppler reactivity effect of the increased fuel temperature and terminated by reactor trip actuated by high nuclear power signals. Analyses show that in the event of ejection of the rod of maximum worth further failure of the reactor coolant pressure boundary would not occur and that the resulting power pulse would not cause excessive damage of fuel or other core damage such that the effectiveness of the safety injection system would be impaired. [6]

Unit No. 2 does not share safety-related systems with either of the other two units on the site. Units 1 and 2 will have a common control room, but the controls for each unit are physically separate. The three units also have a common discharge canal, and there are certain other ties between them such as backup electrical power supplies, city water, and sanitary facilities. Unit No. 2 is therefore virtually independent of the other two units. [7]

VII. ITEMS REQUIRING FURTHER INFORMATION OR DEVELOPMENT DURING CONSTRUCTION

A. Research and development programs

At the time of issuance of the Unit No. 2 construction permit several areas had been identified as requiring further research and development prior to completion of construction. These areas were: (1) verification of core design details and parameters; (2) emergency core cooling system; (3) failure of core cooling systems and means to ameliorate consequences; (4) control rod ejection analysis; (5) reactor coolant pump controlled leakage seals; and (6) air recirculation system halogen filters. Research and development programs have been carried on in these areas, and information developed from these programs has been included in the FSAR. The results are summarized below.

1. Verification of core design details and parameters

The detailed final core design and thermal-hydraulics and physics parameters have been completed. The core design incorporates fixed burnable neutron absorber rods in the initial loading to ensure a negative moderator reactivity temperature coefficient at operating temperature. This improves reactor stability and lessens the consequences of a rod ejection or loss of coolant accident.[1/]

Core stability has been analyzed and design provisions for detection and control of potential axial xenon oscillations have been finalized. The core design incorporates part length control rods for controlling these axial xenon oscillations and shaping the axial power distribution. Tests in operating reactors have demonstrated the ability of the out-of-core instrumentation to give accurate indication of power redistribution and provide the operator information necessary to monitor redistributions and control axial oscillations by moving the part-length rods. This capability will also be verified for Unit No. 2 during startup tests.[2/]

2. Emergency core cooling system

The emergency core cooling system (ECCS) was identified as an area where additional design information would be provided to show that the system adequately meets the design criteria.

Additional development effort on ECCS design has resulted in the modification of the system to include pressurized accumulator tanks for rapid core reflooding.[3] This increased flooding capability assures effective cooling of the core in the event of a loss-of-coolant accident.[4] The system design incorporates redundancy of components such that the minimum required water addition rates can be met assuming any single active component fails. During the long term period of post-accident core decay heat removal, minimum water addition rates will continue to be met assuming a passive or active component failure in either the safety injection or service water systems, or an active failure in the component cooling water system.[5] Because of the incorporation of this revised emergency core cooling system the reactor pit crucible which was proposed during the construction permit review has been deleted from the plant design.[6/]

3. Failure of the core cooling system (as proposed for construction) and means to ameliorate consequences

To assure the effectiveness of the revised ECCS, limits on peak fuel clad temperature and local metal water reaction and effects of flow blockage resulting from rod burst have been established experimentally.[7/]

The effectiveness of the ECCS was determined by improved analytical techniques that have been developed to analyze the loss-of-coolant accident. These techniques represent the culmination of extensive research and development programs carried out by Westinghouse and augmented by AEC-funded research and development. A complete description of the behavior of the core during a loss-of-coolant accident has been provided which includes discussions of the computer programs and assumptions used in analysis as well as the experimental verification of the programs or correlations used in the program. In addition a parametric study was performed that showed the sensitivity of the results to the various important core cooling phenomena.

It was concluded that the improved ECCS will keep the core intact and the peak clad temperature well below the point where Zircaloy-water reaction might have an adverse effect on clad ductility and, hence, on the continued structural integrity of the fuel elements. The ECCS will therefore perform adequately at the proposed power level.[8/]

4. <u>Control rod ejection analysis</u>

A control rod ejection analysis was performed for the final core design, rod worths, rod position limits, and moderator reactivity temperature coefficient. The analysis shows that, with the final core design (including burnable neutron absorbing rods) and the insertion limits for the control rods, no consequential damage to the reactor coolant system and no fuel or clad melt will occur for ejection of the highest worth rod.[9/]

5. <u>Reactor coolant pump controlled leakage seals</u>

The reactor coolant pump controlled leakage seal design for this plant has been fully developed.[10/] Successful operation of the seal design used on Unit No. 2 reactor coolant pumps has been demonstrated in operating plants such as San Onofre, Connecticut Yankee, Ginna, and H.B. Robinson.

6. <u>Air recirculation system halogen filters</u>

Investigations of the effectiveness of the air recirculation system halogen filters were undertaken in a charcoal filter testing program.

The charcoal filters in Unit No. 2 are intended to remove methyl iodide. To achieve this by isotopic exchange, the banks of activated charcoal filters are impregnated with non-radioactive iodides. As the post-accident containment atmosphere is forced through the filters by the containment fan coolers, some fraction of the radioactive methyl iodide is exchanged with the non-radioactive form.

Prior to the work done under this program, it had been established that iodine, as radioactive methyl iodide, could be trapped efficiently from a simulated post-accident containment atmosphere of high humidity. However, there was some uncertainty that the high efficiency of the filters was maintained at and near the conditions of 100% relative humidity which may exist in the containment following a loss-of-coolant accident, and the purpose of the program was to investigate this aspect of charcoal filter performance.

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

From the results of this program it has been concluded that the Westinghouse charcoal filter design used in Unit No. 2 has an initial removal efficiency of at least 70% per pass for all post accident containment atmosphere environmental conditions up to and including 100% relative humidity. Based on these results it has been determined that the charcoal filters are more than adequate to assure that the guidelines in Part 100 of the AEC regulations will not be exceeded.[11]

B. Other items

1. Containment spray program[12]

Review of this and other applications since issuance of the Unit No. 2 construction permit indicated a need for additional information to substantiate the iodine removal effectiveness of the containment sprays with chemical additives following a loss-of-coolant accident. The purpose of the containment spray program has been the development of this technical information.

The containment spray system as described in the FSAR was designed to be activated automatically following such an accident to condense steam, thus reducing the containment internal pressure, and simultaneously to cause absorption of elemental iodine vapor. In order to show that adequate iodine removal could be achieved with the spray system operating at minimum capacity, an analysis was presented in the FSAR based on the "single drop" model derived from Griffiths' work.[13/] Simplifying assumptions were made in this model which were subject to verification before a final assessment could be made of the margins of safety inherent in the system design. Moreover, it was desired to test the validity of the model against measured data obtained in the most realistic practicable simulations of actual containment conditions. The containment spray program resulted in improved modeling capability whereby the assumptions of the preliminary analysis were investigated. Results substantially confirmed the previously calculated performance margins in the spray system.

VIII. TECHNICAL SPECIFICATIONS

The Applicant, with the benefit of discussions with the AEC Staff, has prepared proposed technical specifications as a part of the Application as amended. From these the AEC prepares technical specifications which become a part of the operating license. The technical specifications will govern the operation of the plant throughout its life, and they are provided to assure that the plant will be operated in a safe manner and that plant safety systems will be properly maintained and tested. Deviations from or changes to the technical specifications are not permitted without prior AEC approval.

Some typical examples of items included in the technical specifications are given below (the list is by no means comprehensive):

1. Requirements are placed on all safeguards systems (e.g., safety injection, sprays, filters, etc.). The specifications prescribe the frequency of testing of the systems and associated instrumentation;[1]

specify which components and systems must be operable when the plant is started up,[2] and specify under what conditions of system inoperability the plant must shut down.[3]

2. Set points are specified for instrumentation required to protect the reactor core or to initiate safeguards following an accident.[4] The plant would be automatically shut down by the reactor protective system if any of these set points were exceeded.[5]

3. Periodic tests are required to show that the reactor containment leak rate is below the design value. The types and frequency of such tests are specified.[6]

4. A program is specified for periodic inspection of the primary coolant piping and components, to assure that the primary coolant system retains its integrity.[7]

5. Limits are specified on radioactivity in the primary[8] and secondary coolant systems[9] and on the allowable release of gaseous and

liquid wastes.[10] Operability of instruments and sampling procedures and of effluent processing systems which assure that the abovementioned limits are not exceeded are also specified.[11]

6. Requirements are set forth for plant organization and the numbers and qualifications of personnel. Requirements are also placed on the availability of operating procedures for various normal and abnormal conditions and for the maintenance of records and the reporting of any abnormal occurrences.[12]

IX. CONDUCT OF OPERATIONS

A. Organization and responsibility

The organization for operating Unit No. 2 will be similar to that now used for operating Unit No. 1. The organization is being expanded to provide for the administrative and technical needs of Units 1 and 2. This organization has proven effective during eight years of operation of Unit No. 1. Supervisory personnel in most positions in the Unit No. 1 organization will perform the same functions in the expanded organization for Units 1 and 2. 1/

Plant organization charts, giving the number of persons serving in each capacity, are presented in the FSAR. Resumes of each of the supervisory persons for Unit No. 2 operations are also presented. Most of these supervisors have operating experience on Unit No. 1; and all key personnel hold Senior Reactor Operator Licenses on Unit No. 1. 2/

B. Technical qualifications; training programs

Training of supervisory personnel for Unit No. 2 began in December 1968. Soon after this, training of control room operators was also begun.

The training includes classroom work and trips to various manufacturing facilities. It is being carried out under the supervision of Con Edison personnel as well as representatives of manufacturers of certain specialized equipment. The total training program involves about 2100 hours of in-service training for supervisory personnel and 1240 hours of in-service training for control room operators. Prior to startup of Unit No. 2 the control room operator position and key supervisory positions will be filled with persons who have received AEC licenses as reactor operators on Unit No. 2. Most of these persons hold operators licenses on Unit No. 1. Replacement operators for Unit No. 1 are receiving similar thorough training. [3]

Key maintenance personnel have attended several courses in the operation and maintenance of the various control and instrumentation systems for the plant. These courses have been taught by the equipment manufacturers and the Public Health Service. [4]

Technical backup for operating personnel is provided by the Applicant's engineering organization and by consultants in specialized areas. The experience of Con Edison engineering personnel includes participation in the design and AEC review of both Units 1 and 2. Many management personnel in engineering as well as other Con Edison departments were involved in the design, construction, startup and operation of Indian Point No. 1. 5/

C. Written procedures

The Applicant has prepared written procedures covering all phases of normal plant operation. Deviations from or modifications to these procedures will be made only with the approval of the General Superintendent and the Applicant's Nuclear Facilities Safety Committee. In addition, there will be detailed procedures covering abnormal or accident conditions for all plant systems related to plant safety. 6/

D. Records

Complete records of facility operations, including radiation exposures, releases of radioactivity, fuel inventories, and maintenance activities, will be maintained at the plant. 7/

E. Review and audit of operations

Con Edison established a Nuclear Facilities Safety Committee in 1962. This Committee advises the Executive Vice President, Central Operations, and through him, the President and the Chairman of the Board, concerning the safety aspects of the operation of all nuclear power facilities owned by Con Edison. 8/

The Committee is required by the technical specifications to include in its membership persons specializing in radiation safety, nuclear medicine, reactor physics, nuclear chemistry, nuclear fuel, electrical engineering, mechanical engineering and environmental engineering.

The Committee is also required by the technical specifications to:

1. Conduct a yearly audit of all plant procedures used in the operation, maintenance and environmental monitoring of each nuclear power plant.

2. Review all proposed changes in the plant facilities or procedures relating to safety and all proposed changes to the technical specifications.

3. Review the safety of all tests and experiments and any emergency or infrequent condition that may affect safety.

4. Conduct unannounced spot inspections of plant and monitoring operations, not less frequently than quarterly.

The results of all these reviews are reported directly to the Executive Vice President, Central Operations, the Chairman of the Board and the President of the Company. 9/

F. Emergency plans

In addition to the written procedures for various abnormal plant operating conditions, the Applicant has prepared plans for four highly unlikely contingencies: earthquake, tornado, fire, and radiation incident. These plans are given in detail in the FSAR. 10/

The earthquake, tornado, and fire contingency plans describe actions to be taken to minimize damage done to the plant and to determine that the safety of the plant has not been impaired. The purpose of the radiation contingency plan is to define clearly responsibilities and the organization of plant personnel following an accidental release of radioactivity and to provide these personnel with procedures to be followed during incidents. These procedures are intended to insure that any abnormal release of radioactivity is quickly terminated and to insure that the resulting radiation doses to persons both on and off site are minimized. The radiation contingency plan designates persons assigned to the contingency team, sets forth the responsibilities of the members of the contingency team and designates a contingency coordinator. The contingency coordinator has the responsibility to determine the level of incident which exists. [11]

Three levels of radiation incidents are defined: (1) a local incident, which involves only areas within the plant; (2) a site incident, which involves the total plant site but does not involve areas external to the site boundary; and (3) a general incident, which involves areas beyond the site boundary. For each of these categories, the plan sets forth the conditions which must exist for an incident to be declared and the responsibilities of each member of the contingency team who must deal with the situation. Detailed procedures for dealing with each level of incident may be found in the plan in the FSAR along with detailed lists of persons to be notified in the event of a contingency.

The planned response to a site incident consists basically of evacuation and blocking off of the area, making radiation surveys, terminating the release of radioactivity and cleaning up the areas contaminated. Should a site incident be declared all persons would congregate at a central location and a radiological survey would be made of the site. If necessary, non-essential persons would be evacuated from the site. The contingency coordinator would notify the AEC, New York State and Westchester County that a site incident had been declared.

Meanwhile, efforts to control the event on-site would be initiated and, if necessary, an off-site radiological survey would be made to determine whether a general incident exists. [12]

Should a general incident be declared the site incident plan would be followed. In addition, if necessary, the New York State Department of Health, Bureau of Radiological Health Services, the Westchester County Department of Health and the AEC would be notified that assistance is required. Plant personnel would present data to the State and County to enable them to determine what further action is required.

The tornado, earthquake, fire and radiation contingency plans will all be available at the plant at all times for personnel who would be responsible for implementing them. Periodic drills of plant personnel will be conducted to insure that they are completely familiar with the organization and procedures to be followed in the event of any of these contingencies. [13]

X. INITIAL TESTS AND OPERATION

Prior to full power operation of Unit No. 2 the plant will undergo a thorough, systematic testing program which successively demonstrates the capability and safety of the plant to proceed to each following stage of testing until full power is achieved and maintained.[1/] WEDCO, a wholly owned subsidiary of Westinghouse, has the overall responsibility for engineering, construction management, and initial startup testing.[2/] The initial startup tests are subdivided into several stages, each to be completed before the next stage is undertaken. Following the initial startup and testing program, periodic system and plant performance tests will be performed as described in the technical specifications.

A. Tests prior to initial reactor fueling[3/]

The first stage of the initial tests is a comprehensive testing program which ensures that equipment and systems perform in accordance with design criteria prior to fuel loading. As the installation of individual components and systems is completed, they are tested and evaluated according to predetermined and approved written testing techniques, procedures, or check-off lists. Field and engineering analyses of test results are made to verify that systems and components are performing satisfactorily and to recommend corrective action, if necessary.

The program includes tests, adjustments, calibrations, and system operations necessary to assure that initial fuel loading and subsequent power operation can be safely undertaken. In general, the types of tests are classified as installation, flushing, hydrostatic, hot functional, and preoperational tests. These tests are aimed at verifying that the system or equipment is capable of performing the function for which it is designed. Where practical, preoperational tests involve actual operation of the system and equipment under design or simulated design conditions. In addition, the reactor protection and safeguards instrumentation system will be performance tested prior to core loading.

The reactor coolant system vibration testing program will overlap the plant testing program. Data for this particular program will be taken during cold hydro and hot functional testing prior to fuel loading and also during the low power physics tests which follow fuel loading.[4/]

B. Core loading[5/]

Fuel loading does not begin until the prerequisite system tests and operations as defined in the detailed core loading procedures are satisfactorily completed and the facility operating license is obtained. Upon completion of fuel loading, the reactor upper internals and pressure vessel head are installed and additional mechanical and electrical tests are performed. The purpose of these activities is to prepare the system for nuclear operation and to establish that all design requirements necessary for operation are achieved.

The overall responsibility and direction for initial core loading is exercised by the General Superintendent assisted by the Unit No. 2 Acting

Superintendent. During the initial core loading operation the WEDCO Refueling Manager is in charge of the Westinghouse activities. The process of initial core loading is, in general, directed from the operating floor of the containment structure. Standard procedures for the control of personnel and the maintenance of containment security are established prior to fuel loading. The core configuration is specified as part of the core design studies conducted well in advance of station startup and as such is not subject to change at startup. The core is assembled in the reactor vessel, submerged in water containing sufficient quantities of boric acid to maintain the fully loaded core substantially subcritical.

The core loading procedure documents include a detailed tabular check sheet which prescribe and verify the successive movements of each fuel assembly and its specified inserts from its initial position in the storage racks to its final position in the core. Multiple checks are made of component serial numbers and types at successive transfer points to guard against possible inadvertent exchanges or substitutions of components. The results of each loading step are evaluated by the Applicant's licensed senior reactor operator and the WEDCO Refueling Manager before the next prescribed step is started.

Core loading procedures prevent inadvertent dilution of the boron in the reactor coolant, restrict the movement of fuel to preclude the possibility of mechanical damage, prescribe the conditions under which loading can proceed, identify chains of responsibility and authority and provide for continuous and complete fuel and core component accountability.

C. Post core loading tests[6/]

Upon completion of core loading, the reactor upper internals and the pressure head are installed. An operational leak test is conducted after filling of the reactor coolant system and final reactor vessel head stud tensioning is completed.

Mechanical and electrical tests are performed on the control rod drive mechanisms. These tests include a complete operational checkout of the mechanisms. In addition, tests are performed on the reactor trip circuits to test manual trip operation and, by use of dummy signals, the reactor control and protection system is made to produce trip signals for the various unit abnormalities that require tripping. Finally, a complete functional electrical and mechanical check is made of the movable nuclear detector system (cold shutdown).

D. <u>Initial testing in the operating reactor</u>[7/]

After satisfactory completion of fuel loading and final precritical tests, nuclear operation of the reactor is initiated. This final stage of startup and testing includes initial criticality, low power testing and power level escalation. The purpose of these tests is to establish the operational characteristics of the unit and core, to verify design prediction, to demonstrate that license requirements are being met, and to ensure that the next prescribed step in the test sequence can be safely undertaken. Reactor control set point verification will also be performed during this stage of startup testing.

(1) Initial criticality.[8] Initial criticality is established by sequentially withdrawing the shutdown and control groups of control rod assemblies from the core and then slowly diluting the heavily borated reactor coolant until the chain reaction is self-sustaining.

Written procedures specify alignment of fluid systems to allow controlled start and stop and adjustment of the rate at which the approach to criticality can proceed, indicate values of core conditions under which criticality is expected, specify allowed deviations in expected values, and identify reactor operation responsibilities.

(2) Low power testing.[9] A prescribed program of reactor physics measurements is undertaken to verify that the basic static and kinetic characteristics of the core are as expected and that the values of the kinetic coefficients assumed in the safety analysis are conservative.

The measurements are made at low power and primarily at or near operating temperature and pressure. Measurements include verification of control rod assembly group reactivity worths, of isothermal temperature coefficient under various core conditions, of differential boron concentration reactivity worth and of critical boron concentrations as functions of control rod assembly group configuration. In addition measurements of the relative power distributions are made. Concurrent tests are conducted on the instrumentation including the source and intermediate range nuclear channels.

Detailed procedures are prepared to specify the sequence of tests and measurements to be conducted and conditions under which each is to be performed to ensure both safety of operation and the relevancy and consistency of the results obtained. If significant deviations from design predictions exist, unacceptable behavior is revealed, or apparent anomalies develop, the testing is suspended and the situation reviewed by the Applicant and the WEDCO/Westinghouse technical advisors to determine whether a question of safety is involved, and what corrective action must be taken, prior to resumption of testing. The ultimate responsibility for these determinations rests with Con Edison.

(3) Power level escalation.[10] When the operating characteristics of the reactor and unit are verified by the preliminary zero power tests, a program of power level escalation in successive stages brings the unit to its full-rated power level. Both reactor and unit operational characteristics are closely examined at each stage and the safeguards analysis assumptions verified before escalation to the next programmed level is effected.

Additional reactor physics measurements are made and the ability of the reactor control and protection system to respond effectively to signals from primary and secondary instrumentation under a variety of conditions encountered in normal operations is verified. At prescribed power levels the dynamic response characteristics of the reactor coolant and steam system are evaluated.

The sequence of tests, measurements and intervening operations is prescribed in the power escalation procedures together with specific details relating to the conduct of the several tests and measurements. The measurement and test operations during power escalation are similar to those during normal operation.

E. Initial operation responsibilities[11/]

A Joint Test Group consisting of responsible WEDCO and Con Edison personnel review and concur in release of test procedures for implementation. Technical responsibility for each individual phase of actual startup resides with the functional group most directly concerned with the results of the test. WEDCO and Westinghouse have on-site representatives of supporting functional groups to provide technical advice, recommendations and assistance in planning and executing the respective stages of unit startup. Specific responsibilities during each stage of testing are discussed in preceding respective sections.

All system operations in the testing program are performed by station operators in accordance with the approved written procedures. These procedures include such items as delineation of administrative procedures and test responsibilities, equipment clearance procedures, test purpose, conditions, precautions, limitations, and sequence of operations. Procedural changes are made only in accordance with an approved standard operating procedure that requires review and approval of the changes by experienced supervisory personnel.

Test procedures stating the test purpose, conditions, precautions, limitations and criteria for acceptance are prepared for each test by WEDCO and/or Westinghouse technical advisors. All such procedures are reviewed and concurred in by the

Joint Test Group in accordance with approved standard operating procedures prior to implementation.

All test results will receive a preliminary review and evaluation by Con Edison site personnel. Cognizant WEDCO/Westinghouse startup engineers and technical advisors will determine the adequacy of test data for verification of design objectives. Detailed analyses of test results and issuance of final test reports will be performed by WEDCO site startup and/or Westinghouse engineering and design personnel with input from Con Edison where appropriate. Con Edison will review all final test results to determine that design objectives and criteria have been met and will give final approval as to the acceptability of plant components, systems and operating characteristics of the facility.

XI. FINANCIAL QUALIFICATIONS

Consolidated Edison's financial qualifications to engage in the activities to be authorized by the operating license it now seeks are evidenced in the record of its application, which includes the Applicant's annual reports to its stockholders for the years 1968 and 1969. As shown in the annual report for 1969, the company's assets exceeded four billion dollars and its net income for the year 1969 was over $127 million. This net income was derived from sales of electricity, gas and steam which produced operating revenues of slightly over one billion dollars.

The operation of Unit No. 2 will be financed, as was its construction, as an integral part of the company's regular course of business. The estimated annual operating and maintenance expense for the unit, including fuel, will average about $12 million for the first five years of operation. In 1969 the company's total operating and maintenance expense was $494 million. [1/]

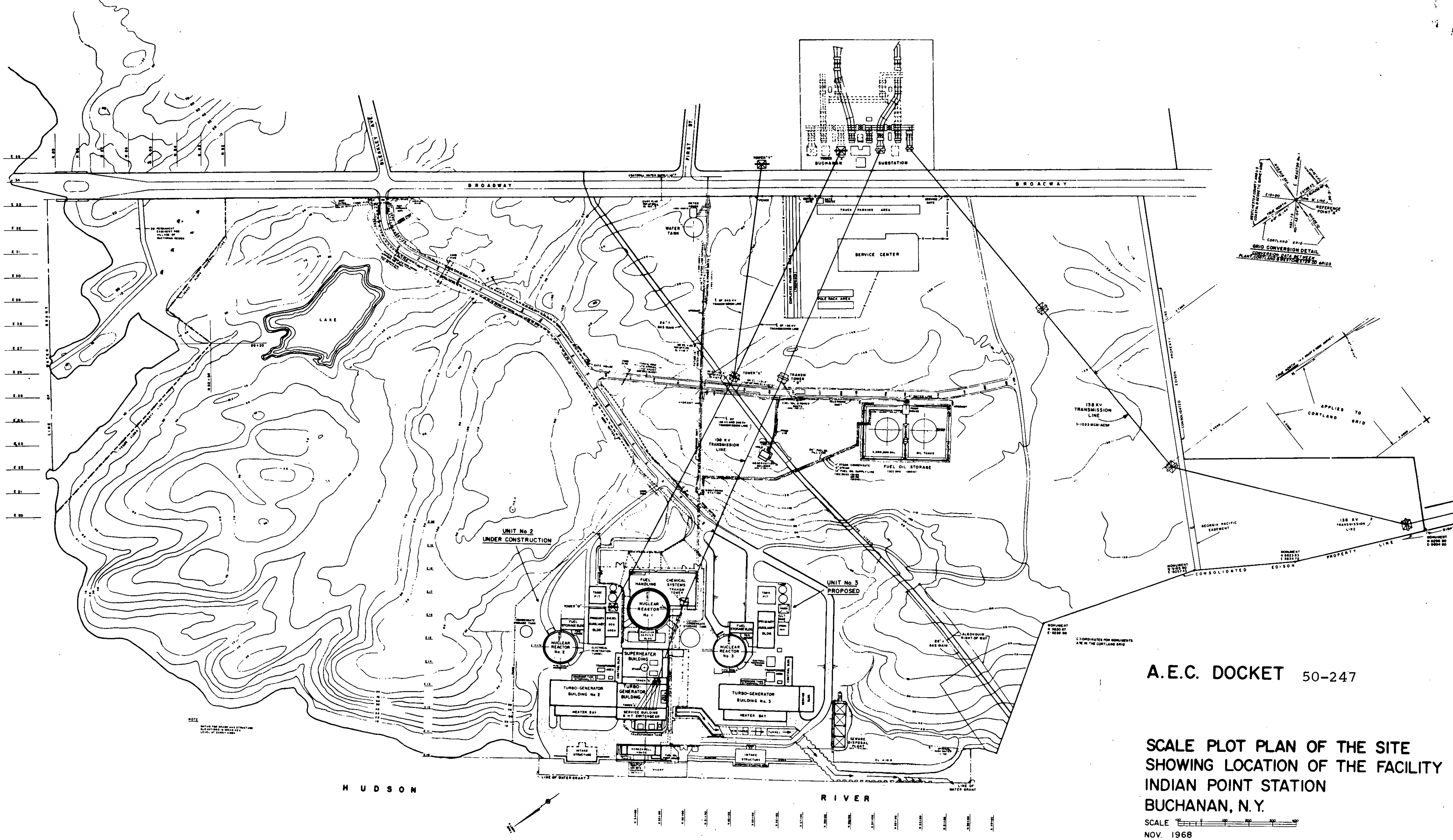
XII. ALIEN CONTROL; ACCESS TO RESTRICTED DATA

All directors and principal officers of Con Edison are citizens of the United States, with the exception of the Vice President, Treasurer, who is a citizen of Canada. Con Edison is not owned, controlled, or dominated by an alien, foreign corporation or foreign government. Con Edison has agreed that it will not permit any individual to have access to Restricted Data until the United States Civil Service Commission shall have made an investigation and report to the Commission on the character, associations and loyalty of such individual, and the Commission shall have determined that permitting such person to have access to Restricted Data will not endanger the common defense and security. [1]

XIII. CONCLUSION

The Application, as amended, summarized herein has described the final design and safety analysis of the facility. Information on the Applicant's financial and technical qualifications to operate the facility have been described. A complete set of proposed technical specifications governing the operation of the reactor has been supplied, as has the Applicant's plan for conduct of operation of the facility.

Con Edison believes that this information, together with other information to be supplied on such matters as the substantial completion of the facility and compliance with the requirements of Part 140 of the Commission's regulations, is sufficient to entitle it to a facility operating license and to support the findings which must be made prior to the issuance of such a license.



A.E.C. DOCKET 50-247

SCALE PLOT PLAN OF THE SITE SHOWING LOCATION OF THE FACILITY INDIAN POINT STATION BUCHANAN, N.Y.

SCALE

NOV. 1968

REFERENCES

Unless otherwise specified references are to portions of the Application for Licenses for Unit No. 2, as amended. The following abbreviations apply:

"FSAR" means Final Facility Description and Safety Analysis Report, as supplemented.

"Application" refers to the 15-page original pleading filed with the Commission in 1965.

"Tech. specs." refers to proposed technical specifications supplied as a separate volume of the FSAR.

"ans. to q. ___" refers to answers to AEC Staff questions appearing in Volumes 5 and 6 of the FSAR.

I. INTRODUCTION
(No references)

II. SITE AND ENVIRONMENT

1. FSAR Sec. 2.2
2. FSAR Sec. 2.4
3. *Ibid.*
4. FSAR Sec. 2.7
5. FSAR Sec. 2.8
6. FSAR Appendix A (Volume 4)
7. FSAR Sec. 2.5
8. FSAR Sec. 2.3; ans. to q. 1.11
9. FSAR Sec. 2.6
10. Tech. specs. Sec. 3.9; FSAR Sec. 2.6; ans. to q. 11.1, 11.2
11. FSAR Sec. 14.3.5, Sec. 2.6
12. Tech. specs. Sec. 3.9
13. FSAR Sec. 2.5, 14.3.2; ans. to q. 11.1
14. FSAR Sec. 2.9; ans. to q. 11.1
15. *Ibid.*
16. *Ibid.*
17. *Ibid.*

III. DESCRIPTION OF FACILITY AND ASSOCIATED PLANT FEATURES

1. FSAR Sec. 1.1
2. FSAR Sec. 1.4
3. FSAR Sec. 3.2.3
4. FSAR Sec. 3.2.1
5. FSAR Sec. 3.2.1; 9.2
6. FSAR Sec. 4.2.2
7. FSAR Sec. 4.5
8. FSAR Sec. 14.3.3; FSAR Appendix A, (Volume 4); ans. to q. 4.7, 4.9
9. FSAR Sec. 4.5
10. FSAR Sec. 4.2.2
11. FSAR Sec. 4.2.2 and 4.1.4
12. FSAR Sec. 4.1.4
13. FSAR Sec. 4.2.2
14. FSAR Sec. 4.1.7, 4.2.5
15. FSAR Sec. 4.1.2, 5.1.5; FSAR Appendix 4 B (at end of Sec. 4); ans. to q. 1.9
16. FSAR Sec. 5.1.1
17. FSAR Sec. 5.1.2
18. FSAR Sec. 6.5
19. FSAR Sec. 6.6
20. FSAR Sec. 5.1.8.2
21. FSAR Sec. 5.1.8.7, 6.6.5
22. FSAR Containment Design Report (Volume 6, Sec. IV.)
23. FSAR Containment Design Report Appendix B (Volume 6, Sec. IV.); FSAR Appendix 14 A (at end of Sec. 14)
24. FSAR Sec. 5.1.2.5
25. FSAR Sec. 6.2
26. FSAR Sec. 6.3
27. FSAR Sec. 6.4
28. FSAR ans. to q. 6.8, 14.19
29. FSAR ans. to q. 6.11
30. FSAR Sec. 7.7
31. FSAR Sec. 7.3
32. FSAR Sec. 7.2
33. <u>Ibid</u>.
34. FSAR ans. to q. 14.7
35. FSAR Sec. 7.4, 7.5, 7.6
36. FSAR Sec. 7.2.1

37. FSAR Sec. 8; FSAR ans. to q. 8.1
38. FSAR Sec. 9.2
39. FSAR Sec. 11.1, 11.2
40. FSAR Sec. 11.1; ans. to q. 11.1
41. FSAR Sec. 9.2
42. FSAR Sec. 9.3
43. FSAR Sec. 9.6.1
44. FSAR Sec. 9.5
45. Ans. to q. 14.6
46. FSAR Sec. 9.4
47. FSAR Sec. 9.8
48. FSAR Sec. 9.9; ans. to q. 7.19
49. FSAR Sec. 10.1, 10.2.4, 10.2.6
50. FSAR Sec. 10.2.2
51. FSAR Sec. 10.1.2
52. FSAR Appendix 14 A (end of Sec. 14)

IV. EVOLUTION OF DESIGN OF UNIT 2 FROM DESIGN OF OTHER FACILITIES

1. FSAR Sec. 1.4

V. QUALITY ASSURANCE

1. FSAR Appendix B (Volume 4)
2. FSAR "Project Reorganization" (Volume 6)
3. Tech. specs. Sec. 4.2
4. Tech. specs. Sec. 4.3
5. Tech. specs. Sec. 4

VI. SAFETY ANALYSES

1. FSAR Sec. 14
2. FSAR Sec. 8.2.3, 14.3; FSAR Appendix A (Volume 4)
3. FSAR Sec. 14.3.5
4. *Ibid*.
5. FSAR Sec. 14.2
6. *Ibid*.
7. FSAR Sec. 6, 7.7, 8

VII. <u>ITEMS REQUIRING FURTHER INFORMATION OR DEVELOPMENT DURING CONSTRUCTION</u>

1. FSAR Sec. 3.2
2. FSAR Sec. 3.1; ans. to q. 3.2 through 3.6
3. FSAR Sec. 6.2
4. FSAR Sec. 14.3
5. FSAR Sec. 6.2
6. FSAR Sec. 1.5; ACRS letter re Unit No. 2 - September 1970
7. FSAR Appendix 14 B (at end of Sec. 14)
8. <u>Ibid</u>.
9. FSAR Sec. 14.2.6
10. FSAR Sec. 4.2
11. Ans. to q. 14.1
12. Ans. to q. 6.2
13. FSAR Appendix 6 A (at end of Sec. 6); FSAR Sec.14.3.5

VIII. <u>TECHNICAL SPECIFICATIONS</u>

1. Tech. specs. Sec. 4.5
2. Tech. specs. Sec. 3.3
3. <u>Ibid</u>.
4. Tech. specs. Sec. 2
5. Tech. specs. Sec. 3.10
6. Tech. specs. Sec. 4.4
7. Tech. specs. Sec. 4.2
8. Tech. specs. Sec. 3.1
9. Tech. specs. Sec. 3.4
10. Tech. specs. Sec. 3.9
11. Tech. specs. Sec. 3.5
12. Tech. specs. Sec. 6

IX. <u>CONDUCT OF OPERATIONS</u>

1. FSAR Sec. 12.1
2. Ans. to q. 12.1
3. FSAR Sec. 12.2
4. Ans. to q. 12.1
5. Application, page 5; Amendment No. 25 to Application
6. FSAR Sec. 12.3; ans. to q. 12.2

7. FSAR Sec. 12.4; ans. to q. 12.4
8. FSAR Sec. 12.5; ans. to q. 12.3
9. Tech. specs. Sec. 6.1
10. Ans. to q. 12.3
11. Ibid.
12. Ibid.
13. Ibid.

X. INITIAL TESTS AND OPERATION

1. FSAR Sec. 13 (including Appendix 13 A, which is substantially the same as this section of the Summary).
2. FSAR ans. to q. 13.4
3. FSAR Sec. 13.1
4. FSAR ans. to q. 13.1
5. FSAR Sec. 13.2.1
6. FSAR Sec. 13.2.2
7. FSAR Sec. 13.3
8. FSAR Sec. 13.3.1
9. FSAR Sec. 13.3.2
10. FSAR Sec. 13.3.3
11. FSAR Sec. 13.4.2; ans. to q. 13.4(1)

XI. FINANCIAL QUALIFICATIONS

1. Amendment No. 21 to Application; Con Edison Annual Report to Stockholders for 1968, filed with AEC July, 1969.

XII. ALIEN CONTROL; ACCESS TO RESTRICTED DATA

1. Application, pages 1, 2; Amendment No. 25 to Application Exhibit A-2

XIII. CONCLUSION

(No references)