BEFORE THE UNITED STATES

ATOMIC ENERGY COMMISSION

In the Matter of	)	
	)	
Consolidated Edison Company of New York, Inc. (Indian Point Unit No. 3)	)	Docket No. 50-286

SUMMARY OF APPLICATION

S/LR
O'R 2/25/69

February 20, 1969

SUMMARY OF APPLICATION

Table of Contents

SUMMARY OF APPLICATION

I. INTRODUCTION

In accordance with the Statement of General Policy set forth in Appendix A to the AEC's Rules of Practice (10 CFR 2, App. A), this document (a) summarizes the application, as amended, submitted by Consolidated Edison Company of New York, Inc. ("Consolidated Edison" or "Applicant") for licenses to construct and operate a facility to be part of a third nuclear generating unit at Indian Point ("Unit No. 3") and (b) evaluates the considerations important to the safety of the facility. The design of this Unit is fully described, analyzed and evaluated in the Preliminary Safety Analysis Report, together with Supplements 1-10 thereof, filed herein as exhibits to the application.

Two of the safety objectives in designing a nuclear power plant are: first, to prevent accidents from occurring, and second, to restrict the consequences of an accident should one occur. To assure itself that these objectives will be met, the Applicant has made numerous evaluations and analyses which are summarized in this document. These include a study of the site and environment of Unit No. 3; analyses of the effects of the plant upon its environment for various hypothesized accident conditions, as well as for normal operations; and the identification of areas in which developmental testing or analysis is required and the formulation of programs to carry

1 out this development and analysis.

2 Some of the terminology appearing in the application
3 has been simplified or explained in this document, and some of
4 the information appearing in scattered portions of the applica-
5 tion has been combined and characterized. The objective is to
6 inform the Board and the public, in as nontechnical terms as
7 feasible, of the evaluations and considerations which have con-
8 vinced Consolidated Edison that its Unit No. 3 can be constructed
9 and operated at Indian Point without undue risk to the health
10 and safety of the public.

11 For members of the Atomic Safety and Licensing Board
12 and others who wish to study in more detail subjects mentioned
13 in this summary, footnotes are provided containing appropriate
14 references.

15 This summary is sponsored collectively by the follow-
16 ing persons as witnesses: Messrs. Anderson, Cahill and Grob
17 of Consolidated Edison; and Messrs. Moore, Hauge, McAdoo and
18 Durfee of Westinghouse Electric Corporation. The qualifica-
19 tions of these witnesses appear in the biographical resumes
20 attached as Appendix A to this document.

II. BACKGROUND; PROJECTED POWER NEEDS; TECHNICAL AND FINANCIAL QUALIFICATIONS

A. Company Background

Consolidated Edison is one of the largest electric utilities in the country. Although its service area is small in comparison with other large private utilities in the United States, the Applicant supplies the greatest concentration of population in the country -- approximately 3 million electric customers in an area of about 600 square miles. This area encompasses the five boroughs of New York City (excluding the Rockaway peninsula) and most of Westchester County. 1/

Among the six largest utilities, Consolidated Edison ranks first as to population and electric customers served, electric utility plant investment and annual revenues from the sale of electricity; sixth in installed generating capability and maximum load; and sixth in annual generation and sales. The generating capacity of the electric system as of December 31, 1968 was 7,607,000 kilowatts. All of Applicant's currently operating electric generating plants use fossil fuels exclusively, except for Indian Point Unit No. 1, which uses nuclear fuel and also some fuel oil for superheating. In addition to its electric operations, Consolidated Edison also supplies gas and steam service in portions of its service territory. 2/

1 B. Consolidated Edison Role in Nuclear Power

2 Consolidated Edison became involved in nuclear matters
3 even prior to the passage of the Atomic Energy Act of 1954, which
4 permitted the large-scale use of special nuclear material by
5 private industry for the first time. Several of its employees
6 were assigned to laboratories and industrial organizations for
7 the purpose of research and training in the techniques and
8 development of nuclear power. It filed with the Atomic Energy
9 Commission the first application for permission to construct and
10 operate a nuclear power plant. This plant, known as Indian Point Unit
11 No. 1, first went critical in August, 1962 and has operated
12 successfully to date. The Applicant was authorized in 1966 to
13 construct a second nuclear unit at Indian Point.

14 Consolidated Edison has also participated in the
15 development of nuclear energy through membership in Empire
16 State Atomic Development Associates, Inc., which has sponsored
17 development work on advanced reactor concepts and on the
18 economics of nuclear power. This group, known as ESADA, was
19 formed by the private utilities of New York State. It has
20 sponsored programs to further the development of engineered
21 safeguards and to improve the methods of designing and inspect-
22 ing reactor vessels, among others.

23 Consolidated Edison has participated over the past
24 fourteen years in Atomic Power Development Associates, Inc., a

1 research and development organization concentrating on breeder
2 reactor technology. Applicant also helped to establish the
3 Industrial Reprocessing Group, which stimulated the industrial
4 interest in nuclear reprocessing, leading eventually to the
5 establishment of Nuclear Fuel Services, Inc. More recently,
6 the Applicant helped organize the Plutonium Export Association,
7 an association of plutonium producers. In all the organizations
8 mentioned above, both managerial and technical personnel of the
9 Applicant have been active.

10 C. Projected Power Needs

11 Applicant's maximum load is expected to increase from
12 7,350 megawatts in 1969 to 9,075 megawatts in 1974. According to
13 the Applicant's present plans, the increase in these requirements
14 would be satisfied primarily from nuclear power plants. It is
15 expected that Unit No. 3 would constitute 9.3% of the Applicant's
16 system generation capacity as of the summer of 1972. In addition
17 to Indian Point Units 1, 2 and 3, Consolidated Edison has already
18 announced the projected construction, if authorized by the Atomic
19 Energy Commission, of another nuclear unit on the Hudson River,
20 south of Indian Point, which it is planned would become operational
21 during this period.

1 D. General Policy of the Applicant Concerning Safety
2 of the facility

3 As has been the case with Consolidated Edison's
4 Nuclear Units 1 and 2, safety considerations have been vital
5 in all decisions pertaining to the design of Unit No. 3.
6 Consolidated Edison has selected as its prime contractor
7 Westinghouse Electric Corporation, which has had very extensive
8 experience in the nuclear power field. As is more fully explained
9 in Part IV of this summary, the design of Unit No. 3 is based
10 upon the design of similar pressurized water reactors which have
11 been licensed by the Atomic Energy Commission for construction
12 or operation.

13 Consolidated Edison recognizes that it has a corporate
14 responsibility for the safety of Unit No. 3. This responsibility
15 exists entirely apart from any safety requirements imposed by
16 the Atomic Energy Commission. Consolidated Edison will monitor
17 the design and construction of the facility in a number of ways
18 in order to assure itself that the plant can be operated safely.
19 Among the ways it will so check on the safety aspects of design
20 and construction are the following:

21 First, by having representatives on the
22 site to maintain surveillance over physical construction.

23 Second, by technical review of design
24 by its engineering departments and by its technical
25 consultants.

1 Third, by review of design by its produc-
2 tion department in order to determine the suitability
3 of the design and construction for operational
4 requirements.
5 Part IX and the Supplement to this summary discuss in
6 further detail the matter of the quality assurance plan which
7 will be carried out to assure that the plant is so constructed
8 that it will operate in a safe and reliable manner.

9 E. Technical Qualifications[3/]

10 As noted above, Consolidated Edison has long been active
11 in the field of nuclear energy, such experience now having exceeded
12 fifteen years. In addition to its participation in the ESADA and
13 APDA programs, the following activities have also contributed to
14 develop its skills in nuclear technology:
15 In 1955, prior to the construction of
16 Indian Point Unit No. 1, several engineers were lent
17 to organizations actively engaged in nuclear activi-
18 ties -- primarily the Naval Reactors Program. These
19 engineers actively participated in nuclear design at
20 Knolls Atomic Power Laboratory and the Westinghouse-
21 operated Bettis Atomic Power Laboratory and now hold
22 supervisory positions in the engineering and operating
23 departments of the Applicant. Others are former
24 employees of reactor manufacturers or firms in

1 related industries. Several of the Applicant's
2 engineers hold or are studying for advanced
3 degrees in nuclear engineering. A number of engineers
4 received special training at various reactor instal-
5 lations and were the original AEC-licensed operators
6 for Indian Point Unit No. 1. These engineers took
7 formal courses in nuclear engineering and gained
8 reactor operating experience at the Shippingport,
9 Vallecitos and MTR and ETR facilities. Nineteen
10 of the original operator-licensees now hold positions
11 in the engineering, construction and operating
12 departments of Consolidated Edison.
13 Applicant acted in lieu of a general con-
14 tractor for the construction of its Indian Point Unit
15 No. 1. It was found by the Atomic Energy Commission
16 in Docket No. 50-3 to be technically qualified to con-
17 struct and operate that facility. A little over two
18 years ago, in Docket No. 50-247, the Applicant was
19 found technically qualified to design and construct
20 the Indian Point Unit No. 2 facility, which is
21 very similar to the Unit No. 3 facility.
22 Applicant has safely operated Indian Point
23 Unit No. 1 for over six and a half years and as of
24 December 31, 1968 had generated over 7,424,000 mega-
25 watt hours of electricity from the facility. Many

1 of the Applicant's officers and employees in several
2 departments have become familiar with the design and
3 the features affecting safety which are incorporated
4 into Unit No. 3.

5 Applicant's principal contractor is the
6 Westinghouse Electric Corporation, which is respon-
7 sible for the design and construction of Unit No. 3,
8 including procurement of all materials and components
9 for the plant. As noted earlier, Westinghouse has
10 designed, has completed and is building a large
11 number of nuclear power reactors including Shipping-
12 port, Yankee-Rowe, San Onofre, Connecticut Yankee,
13 Ginna, Diablo Canyon, Indian Point 2 and Zion.
14 Westinghouse-designed nuclear power plants totaling
15 about 14,000 megawatts electric in capacity are
16 presently in service or under construction in the
17 United States.

18 In addition to its own personnel who are
19 qualified in the nuclear field, Consolidated Edison
20 has engaged a number of independent consultants who
21 have rendered advice and assistance in the prepara-
22 tion of reports and material dealing with geology,
23 seismology, hydrology, meteorology, demography and
24 environmental radioactivity. Applicant further has

1 engaged Dr. C. Rogers McCullough as a general con-
2 sultant. Dr. McCullough has reviewed the overall
3 adequacy of the Unit No. 3 design from a nuclear
4 safety point of view and will testify as to his
5 conclusions at the hearing on the application.
6 While at this stage of the planning of Unit No. 3
7 the operating procedures are not yet prepared, Consolidated
8 Edison has general plans for plant operation, especially as
9 to the training of personnel and the allocation of operating
10 responsibilities within the Company. The basic nuclear train-
11 ing of the Unit No. 3 operating force has been obtained on
12 Indian Point Unit No. 1 and will be obtained on Unit No. 2
13 after it has been authorized for operation. About one year
14 prior to the startup of Unit No. 3 the personnel will receive
15 on site and off site training by Westinghouse, on the specific
16 features of Unit No. 3. All preoperational test procedures will
17 be reviewed by the operating force which will also perform these
18 tests under the technical direction of Westinghouse. Those oper-
19 ations and tests associated with fuel loading, initial criticality
20 and power testing will be under the control of the Station General
21 Superintendent with technical assistance provided by Westinghouse.
22 Administrative responsibility for operation of the
23 facility will rest with the Station General Superintendent

1 (who is also the General Superintendent for Unit No. 1 and
2 No. 2) and who, in turn, reports to the manager of the Produc-
3 tion Department. The manager of the Production Department in
4 turn reports to the vice president of the Company who is res-
5 ponsible for all generating facilities.

6 F. Financial Qualifications[4/]

7 The estimated expenses to be incurred in connection
8 with the construction of Unit No. 3 will be approximately
9 $197,000,000, which include transmission, distribution and
10 general plant costs and nuclear fuel inventory cost for the
11 first core. By way of comparison, the Applicant's net plant
12 investment at the end of 1967 was approximately $3.4 billion.
13 No special financing or financing techniques will
14 be necessary or will be used for Unit No. 3. During the
15 period 1968 through 1972 the Applicant expects to spend approxi-
16 mately $1.4 billion on its contruction program.
17 In connection with this five-year construction pro-
18 gram, the Applicant in 1968 issued at par 931,432 shares of
19 Cumulative Preference Stock ($100 par value) and $60,000,000
20 principal amount of Series FF Mortgage Bonds. The Applicant
21 estimates that because of this five-year construction program
22 it will be required to raise approximately $617,000,000 through
23 the sale of securities. The types of securities and the times

1 at which they may be issued cannot now be determined. The
2 balance of the required funds will be obtained from retained
3 earnings and from other funds available from internal sources,
4 principally provisions for depreciation.
5 The estimated plant cost of Unit No. 3 is included
6 in the above five-year estimate of construction expenditures
7 except for certain expenditures made prior to December 31, 1967.

III. FACILITY SITE AND ENVIRONMENT

Applicant owns a tract of land called Indian Point which consists of 235 acres and which is located on the Hudson River in the Village of Buchanan, Westchester County, New York. It is about 24 miles north of the New York City boundary line. The proposed facility will be built on this site adjacent to and south of the Applicant's existing Nuclear Unit No. 1.[1/] The site is shown on the map attached as Appendix B to this document.

The Preliminary Safety Analysis Report contains present population data as well as projected population figures for a 55-mile radius of the site.[2/] Based upon the 1960 census, approximately 53,000 people live within a 5-mile radius of the site, and this number is expected to increase to about 108,000 by 1980. The 1960 population within a 15-mile radius of the site was 326,930, whereas the estimated 1980 population is about 670,000. Within a 5-mile radius most of the population is located northeast of the site. Within the larger radius the majority of the people are located south of the site.[3/]

The area surrounding Indian Point is generally residential with some large parks and military reservations. The projections indicate that the land usage within 15 miles of the site will not change appreciably during the period prior to 1980.[4/]

1 The Indian Point site consists geologically of a
2 fine-grained phyllite, a schist, and limestone, with bedrock
3 lying very close to the surface. Unit No. 3 will be located
4 on limestone, which is hard although jointed. The bedrock
5 will support any foundation loads up to 50 tons per square
6 foot, which capacity far exceeds any load that this plant will
7 superimpose on the bedrock. It will therefore provide a firm
8 foundation for the facility.5/
9 According to the Applicant's consultant on seismology,
10 the Indian Point site is located in one of the safest areas
11 relative to earthquake activity, both as to historical inci-
12 dence and the probability of future occurrence. This consultant
13 is of the opinion the probability of a serious shock occurring
14 in the area of the site within the next several hundred years
15 is practically nonexistent. The highest intensity recorded in
16 this area is the equivalent of a horizontal acceleration of
17 less than 0.1 g. The design earthquake, defined as horizontal
18 acceleration of 0.1 g, acting simultaneously with a vertical
19 acceleration of 0.05 g, has been used as a design criterion
20 for the containment building and for other structures and
21 equipment of Unit No. 3 which are important to safety (Class
22 I). The plant is also designed in such a manner as to be
23 capable of safe shutdown in the event of a hypothetical earth-
24 quake having a horizontal acceleration of 0.15 g acting

1 simultaneously with a vertical acceleration of 0.10 g.[6/]

2 The Plant design precludes leakage of radioactive
3 liquids from the processing buildings. Even if such leakage
4 were hypothesized, sources of ground water would not be
5 susceptible to contamination.[7/]

6 The combined routine releases of radioactivity to
7 the Hudson River from all three Indian Point Units will be
8 far below limits imposed by applicable AEC regulations
9 (10 CFR Part 20) at the discharge canal; that is, they will
10 meet the permissible limits for drinking water as they leave
11 the discharge canal.[8/] The Chelsea Pumping Station owned by
12 the City of New York uses the Hudson River as a source of
13 drinking water. It is located approximately 22 miles north
14 of the site.[9/] The City of Poughkeepsie, which is about 30
15 miles north of Indian Point, also uses the river as a source
16 of drinking water.[10/] The river flow at Indian Point is primarily
17 the result of tidal dynamics and, therefore, even during periods
18 of drought excellent mixing is provided. The peak tidal flow
19 past Indian Point is 80 million gallons per minute. A compre-
20 hensive study of discharges of radioactive wastes to the
21 river demonstrates it would be incredible that discharges
22 from Unit 3 would result in allowable limits for drinking
23 water being exceeded at either Chelsea or Poughkeepsie.[11/]

1 Applicant has performed a comprehensive analysis of
2 possible flooding conditions at the site. The plant's design
3 will include the capability to prevent equipment required to
4 maintain the plant in a safe condition from being jeopardized
5 by water in the case of the maximum hypothesized flood.[12/]
6 When Indian Point Unit No. 1 was being constructed,
7 New York University conducted a 2-year detailed study of the
8 meteorological conditions at the site. This study was supple-
9 mented by data from the National Weather Records Center at
10 Bear Mountain Weather Station, which was located approximately
11 three miles north of the site.[13/] The most important meteoro-
12 logical characteristic of the site is the prevalent north-south
13 wind direction. This is a result of the orientation of the
14 ridges in the Hudson Valley.[14/] These predominant winds hold
15 at all altitudes within the valley and for lapse, neutral and
16 inversion conditions. When winds aloft are calm or light, valley
17 winds which are diurnal in nature occur, that is, they go down
18 the valley (or south) during the night and up the valley (or
19 north) during the day.[15/] Atmospheric diffusion calculations
20 which have been made confirm that off-site doses due to normal
21 releases of gaseous radioactivity will be far less than the
22 limits set by Part 20 of the Commission's regulations and that
23 calculated off-site doses due to theoretical leakage under hypo-
24 thetical accident conditions would fall well within the AEC's

1 reactor site criteria (10 CFR Part 100).16/

2 Tornadoes are not to be expected at Indian Point.
3 Nevertheless, features of the facility required for safe shut-
4 down and long term core cooling will be protected against
5 tornadoes with wind speeds of 300 miles per hour tangential
6 velocity, 60 miles per hour traverse velocity, and a differential
7 pressure drop of 3 psi in 3 seconds.17/

8 Continuous monitoring of radiation in the vicinity
9 of Indian Point started some 11 years ago when Unit No. 1 was
10 under construction. Since then, samples have been taken con-
11 tinually of the river water, nearby reservoirs, vegetation,
12 marine life, soil and airborne particulate.18/ These environ-
13 mental data will provide a background reference for checking
14 on the radioactivity discharged from Unit No. 3.

IV. EVOLUTION OF DESIGN OF INDIAN POINT UNIT 3 FACILITY FROM DESIGN OF OTHER FACILITIES

This section describes the evolution of the design of the Indian Point Unit 3 facility, including associated engineered safeguards, from those of other facilities previously approved by the Commission. Particular emphasis is placed upon the similarities to and differences from the design of the Indian Point Unit 2 facility. A tabular comparison of Unit No. 3 with other nuclear plants is attached as Appendix C.

A. Reactor

Unit No. 3 will utilize a pressurized water reactor. This reactor type has demonstrated successful and safe operation beginning with the 239 MWt Shippingport plant in 1957. Consolidated Edison began operating Indian Point Unit No. 1 in 1962 at 585 MWt and in 1965 increased the rating to 615 MWt. Yankee-Rowe began operation in 1961 at 392 MWt and is now licensed at 600 MWt. More recent pressurized water reactors are San Onofre (1347 MWt), which began operation in 1967 and Connecticut Yankee (1473 MWt), which began operation in 1967. Indian Point Unit No. 3 will have an initial power level of 3025 MWt. Among the pressurized water reactors with comparable power ratings already approved for construction are Indian Point Unit 2, Diablo Canyon, Salem Units 1 and 2, and the two Zion reactors.

1 The evolution of nuclear reactors has been charac-
2 terized by increases in power density. The peak linear heat
3 rate for Indian Point Unit No. 1 is 12.1 kw/ft compared to
4 17.6 kw/ft in Unit No. 3. The latter is comparable to current
5 reactor designs, such as Zion (18.9 kw/ft), Diablo Canyon
6 (18.9 kw/ft), and Palisades (15.3 kw/ft).
7 Unit No. 3 will utilize zircaloy-clad uranium oxide
8 fuel, which has been used in all Westinghouse power reactors
9 approved since San Onofre and Connecticut Yankee. The system of
10 reactivity control -- chemical shim and control rods -- first
11 was demonstrated in the Yankee-Rowe, Saxton, Trino Vercellesi
12 (formerly known as SELNI) and SENA nuclear power plants and is
13 used in Indian Point Unit No. 1. All central station Westinghouse
14 pressurized water reactors since Yankee-Rowe have utilized this
15 method of control.
16 Current Westinghouse pressurized water reactors, begin-
17 ning with San Onofre and Connecticut Yankee, employ rod cluster
18 control assemblies which are designed to reduce power peaking
19 and thus provide more favorable spatial power distribution. Part
20 length control rods are included in the Unit No. 3 reactor to
21 control axial xenon oscillations should they occur. This concept
22 is also utilized in the Ginna, Diablo Canyon, Indian Point Unit 2,
23 Zion and other current reactors.

1 B. Reactor Coolant System

2 The Unit No. 3 reactor coolant system is similar to
3 Westinghouse systems designed for San Onofre, Ginna, Diablo
4 Canyon, Zion and other recent plants and is almost identical to
5 Indian Point Unit 2. All these systems have design pressures
6 of 2500 psia and design temperatures of 650° F. The coolant
7 system design utilizes a number of independent loops that pro-
8 vide sufficient heat removal capacity for each plant. Thus,
9 Ginna (1300 MWt) will use two loops, and Indian Point Unit 2
10 (2758 MWt), Diablo Canyon (3250 MWt), and Indian Point No. 3
11 (3025 MWt) will use four loops. Each loop contains a steam
12 generator and pump which are similar in design features in all
13 aforementioned plants but which may vary slightly in design
14 parameters to fit the plant operating characteristics.

15 C. Containment

16 The Unit No. 3 steel-lined, reinforced concrete
17 containment will be similar to that used in Indian Point Unit
18 2, Diablo Canyon and other plants.

19 The weld channel pressurization system and the
20 isolation valve seal water system, which are intended to
21 provide an essentially leak-tight containment system, are
22 also utilized in Indian Point 2 and Zion.

1 D. Engineered Safeguards

2 The engineered safeguards in Unit No. 3 will be
3 similar to those used in all Westinghouse plants following
4 the Ginna plant of Rochester Gas and Electric.

5 The safety injection system will include one passive
6 accumulator on each primary coolant loop. Three high head
7 and two low head pumps located outside the containment will
8 also inject water into the reactor during accident conditions.
9 Indian Point No. 2 and No. 3 utilize two additional low head
10 pumps located inside containment to recirculate water following
11 an accident.

12 Redundant containment fan cooling units and spray
13 systems are provided to reduce containment pressure following
14 an accident. This is similar to other plants. Sodium hydroxide
15 in the spray water will remove elemental iodine from the post-
16 accident containment atmosphere, thereby minimizing the leakage
17 of radioactivity from containment.

18 Charcoal filters will be included in the Unit No. 3
19 containment air recirculation units to remove organic iodine
20 from the post-accident containment atmosphere. Other plants
21 also include charcoal filters, but these were included to
22 reduce short-term leakage of all volatile forms of iodine.
23 As spray technology progressed it was shown that sodium
24 hydroxide additive in the containment spray provided a much

1 faster means of removing all but the organic form. Because
2 the amount of fission product iodine in the organic form is
3 relatively small, its contribution to the accident consequences
4 becomes important only after the other forms have been removed
5 by the spray. For Unit No. 3, the charcoal filters are relied
6 upon to reduce the total leakage dose from organic iodine
7 during the course of the accident.
8 Two hydrogen recombiners are to be utilized in the
9 containment following a loss-of-coolant accident to remove hydrogen
10 generated by the metal-water reaction, radiolysis of water, and
11 alkaline reaction with aluminum. Similar recombiners are being
12 installed in the Ginna plant.

V. PRINCIPAL ARCHITECTURAL AND ENGINEERING CRITERIA

The principal architectural and engineering criteria for the Indian Point Unit 3 facility, as set forth in section 2.4 of the Preliminary Safety Analysis Report, are reproduced below. The statement of criteria was based upon the AEC's general design criteria which were proposed at the time the Application was filed. Nevertheless, the design of the facility has been shown in a supplement to the Preliminary Safety Analysis Report [1/] to meet the intent of the currently proposed AEC 70 General Design Criteria for Nuclear Power Plant Construction Permits.[2/]

1. Quality and Performance Standards

Those features of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to the mitigation of their consequences shall be designed, fabricated, and erected to:

(a) Quality standards that reflect the importance of the safety function to be performed. Approved design codes shall be used when appropriate to the nuclear application.

(b) Performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces imposed by the most severe earthquakes, flooding conditions, winds, ice, or

other natural phenomena characteristic of the proposed site.

2. Reliability

Sufficient redundancy and independence shall be provided in systems so that no single failure of any active component of the system can prevent action necessary to prevent an unsafe condition.* These systems should be designed so that effects of such conditions as gross disconnection of the system, loss of energy (electric power, instrument air), and adverse environment (extreme heat, cold, fire, steam, water, etc.) cause the system to go into its safest state (fail-safe) or are tolerable on some other basis. Redundancy and independence of static elements such as piping and wiring are necessary only if the event to be protected against can cause damage to the static element and thereby prevent a necessary safety action.

3. Testing

Capability shall be provided for demonstrating by analysis or test the functional operability of systems or

* As used in these criteria, an unsafe condition means a condition which would increase significantly the likelihood of release of unacceptable quantities of radioactivity to the public environment. The term also takes into account any significant increases in the likelihood of exposing the public to unacceptable levels of direct radiation. Unacceptable quantities of radioactivity release and unacceptable levels of radiation exposure under both normal and abnormal circumstances have been defined by the AEC in 10 CFR 20 and 10 CFR 100, respectively.

components necessary to prevent an unsafe condition.

4. Control

The reactor facility shall be designed so that all actions can be controlled or monitored as necessary to maintain safe operational status of the plant at all times.

5. Electric Power Supplies

Sufficient normal and emergency supplies of electrical power shall be provided to assure a capability for prompt shutdown and continued maintenance of the reactor facility in a safe condition under all credible circumstances.

6. Protection Against Dynamic Effects

Protection shall be provided against dynamic effects resulting from plant equipment failures and causing an unsafe condition.

7. Nil-Ductility Temperature Limits

Components of the primary coolant and containment systems which are potentially subject to propagation-type failure shall be designed and operated so that no substantial pressure or thermal stress will be imposed on the structural materials unless their temperatures are sufficiently above the nil-ductility temperatures.

8. Reactor Protection System

A reliable protection system shall be provided to automatically initiate appropriate action whenever such action is necessary to prevent an unsafe condition.

9. Oscillations and Transients

The reactor system shall be designed to accommodate or readily suppress, without causing an unsafe condition, oscillations or transients resulting from anticipated events such as tripping of the turbine generator or loss of power to the reactor recirculation pumps.

10. Fuel Performance

The fuel shall be designed to accommodate throughout its design lifetime all normal and abnormal modes of anticipated reactor operation, including the design overpower condition, without failure that would result in fission product inventories in the primary coolant or in storage facilities that would preclude continued operation within the limits imposed by applicable regulations for normal release and potential accident releases.

11. Reactivity Insertion

The maximum reactivity worth of control rods or elements and the rates with which reactivity can be inserted shall be held to values such that no single credible control

system malfunction could cause a reactivity transient capable of causing an unsafe condition.

12. Control Rod Ejection

The reactor shall be designed and operated so that a control rod ejection brought about by failure of a rod drive housing does not cause further rupture of the primary system.

13. Shutdown Margin

Reactivity shutdown capability shall be provided to make the core sub-critical from any credible operating condition with the most reactive control rod withdrawn.

14. Primary Shutdown System Capability

The primary shutdown system shall be designed to be operable under abnormal conditions anticipated at the site.

15. Secondary Shutdown Capability

Secondary or backup reactivity shutdown capability shall be provided that is independent of primary means of reactivity shutdown. This system must have the capability to shut down the reactor from any operating condition.

16. Decay Heat Dissipation

The design shall provide means of dissipating core decay heat under all anticipated abnormal and credible conditions,

1 such as isolation from the main condenser or complete or partial
2 loss of primary coolant from the reactor.

3 17. Chemical Reactions

4 Provisions shall be included to limit the extent and
5 the credible consequences of chemical reactions that could cause
6 or materially augment the release of hazardous amounts of fission
7 products from the facility.

8 18. Containment Integrity

9 The containment structure, including access openings
10 and penetrations, shall be designed and fabricated to accommodate
11 without failure credible transients of pressure and temperature.
12 These transients shall be analyzed with allowance for appropriate
13 operating and failure modes of engineered safeguards. If part
14 of the primary coolant system is outside the primary reactor
15 containment, appropriate safeguards shall be provided for that
16 part as necessary to protect the health and safety of the public,
17 in case of an accidental rupture in that part of the system.

18 19. Containment Cooling

19 Provision shall be made for the removal of heat from
20 within the containment structure as necessary to maintain the
21 integrity of the structure under accident conditions. If active
22 heat dissipation systems are needed to prevent containment vessel

failure due to heat released under such conditions, at least two independent systems shall be provided, preferably of different principles.

20. Containment Isolation

A reliable containment isolation system shall be provided where necessary to assure containment integrity.

21. Containment Leakage

The containment shall be designed so that its maximum integrated leakage under accident conditions shall meet the site exposure criteria set forth in 10 CFR 100.

22. Access Provisions

The facility shall be provided with adequate radiation protection to permit access, even under accident conditions, to equipment as necessary to maintain the facility in a safe condition.

23. Effluent Release

Where environmental conditions can be expected to require limitations upon the release of operational radioactive effluents to the environment, appropriate holdup capacity shall be provided for retention of gaseous, liquid or solid effluents.

24. Fuel and Waste Facilities

Fuel and waste storage and handling systems shall be

1 designed and operated in such a manner that credible accidental
2 release of radioactivity will not exceed the limits set forth
3 in 10 CFR 100. The fuel handling and storage facilities shall be
4 designed to prevent criticality and to maintain adequate
5 shielding and cooling for spent fuel under all anticipated
6 normal and abnormal conditions and credible accident conditions.

VI. DESCRIPTION OF FACILITY AND ASSOCIATED PLANT FEATURES

The application 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. 3 for Indian Point will utilize a pressurized water reactor with an initial rating of 3025 megawatts thermal and 965 megawatts electric.[1/] The reactor will operate at a pressure of 2250 psia and an average temperature of 579°F.[2/]

The reactor core will be approximately eleven feet in diameter and twelve feet long. It will be made up of 193 fuel assemblies, each containing a square array of 204 fuel rods. These fuel rods will be fabricated from Zircaloy tubes filled with fuel pellets of slightly enriched uranium dioxide. There will be 21 unoccupied spaces in the fuel rod array, 20 of which will be occupied by guide tubes and by control rod absorbers or burnable poison rods. The remaining one is available for an in-core instrumentation thimble.[3/]

Core reactivity will be controlled by a combination of fixed burnable poison rods, movable absorber rods and neutron absorber dissolved in the coolant. The movable absorber rods

will contain an alloy of silver-indium-cadmium encapsulated in stainless steel, and the soluble neutron absorber will be boric acid dissolved in the primary coolant. The movable absorber rods will be grouped in clusters and used for short term reactivity changes, such as those accompanying unit load changes. Some of the rods will be full-length and others part-length, the latter being available to control xenon oscillations should they occur. The full-length cluster control (RCC) assemblies can shut down the reactor from full power level at any time with the reactor coolant system at normal operating temperature and pressure. The RCC assemblies will be actuated by individual magnetic-latch type drive mechanisms located on the reactor vessel head. Upon reactor trip the rods fall into the core by gravity. The part-length RCC assemblies are driven by a mechanism which does not allow them to fall when the reactor is tripped.

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 at room temperature and at atmospheric pressure and to provide a safe shutdown margin during refueling. During the first fuel cycle, burnable poison rods will be used to ensure a negative moderator temperature coefficient of reactivity at operating temperatures.[4/]

The reactor vessel will be a cylinder 14-1/2 feet in inside diameter, with a hemispherical bottom and a bolted removable hemispherical head. Nozzles above the top of the core connect the vessel to the reactor coolant loops at the sides. The vessel will be constructed of a low alloy steel with all interior surfaces clad with corrosion-resistant stainless steel.[5] 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.[6] The internals are designed to withstand the combined effects of the hypothetical earthquake and a loss of coolant accident. 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.[7]

Four cooling loops will be used to carry the heat from the reactor. Reactor coolant will be pumped through a stainless steel piping system to a vertical inverted U-tube steam generator in each loop. Coolant will enter and leave at the bottom, passing through the inside of the steam generator tubes before being pumped back to the reactor by a single stage centrifugal coolant pump driven by a motor of conventional design. The pumps will have controlled leakage shaft seals and will each have a design capacity of 88,500 gpm.[8]

A vertical surge tank approximately half filled with reactor coolant will act as a pressurizer to control system pressure. The operating pressure of 2,250 psia will be maintained by a combination of electric immersion heaters and a spray of reactor coolant to condense steam in the dome of the pressurizer to limit pressure during load changes.[9]

All materials and components which form a part of the reactor coolant system pressure boundary will meet or exceed the requirements of applicable ASME Codes and together with their supports are designed to withstand the combined effects of the hypothetical earthquake and a loss of coolant accident.[10]

B. Containment

The design of Unit No. 3 includes a massive reinforced concrete containment 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.

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 steel liner has a minimum thickness of 1/4 inch.

The containment is a self-contained free-standing structure that does not require anchorage to the ground.[11/]

The design includes a containment isolation valve seal water system which permits, when required, automatic rapid sealing of pipes which penetrate the containment. In all cases, redundancy of isolation is provided through either the use of multiple independent valves, or where permissible because of the inherent isolation of the system itself, a single valve.[12/] 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. The pressurization system also provides a means of continuously monitoring the leakage status of the containment.[13/]

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 (54 psig).[14/] The structure will be tested at design pressure (47 psig) 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 atmosphere.[15/]

Analyses have been made which confirm the ability of the containment structure to withstand various loading combinations, including those associated with the simultaneous occurrence of an earthquake and the most severe loss-of-coolant accident.[16/] Results of other analyses confirm that missiles generated either by a tornado or by a turbine-generator failure will not penetrate the containment structure.[17/] Protection is also provided against missiles which might be generated from the reactor coolant system.[18/] The Applicant recognizes the importance of assuring that the main-current-pump flywheel receives special attention in matters of design, materials, quality assurance, and in-service inspection. Further information on these matters, in addition to that already presented,[19/] will be reviewed with the AEC regulatory staff in the latter part of this year. Applicant will take any additional steps determined by the AEC staff to be necessary to assure the integrity of the flywheel assembly. The ability of the containment and associated safeguards to contain fission products resulting from various postulated accidents is discussed later in this summary.

C. Engineered Safeguards

In addition to the pressurization system for containment penetrations and liner weld channels and the seal water injection system described above, the following engineered safeguards are incorporated for the protection of the public:

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 both damage to the reactor core and also the energy and fission products released from the reactor into the containment. The system includes four accumulator tanks, a boron injection tank, three high-head safety injection pumps, two low-head residual heat removal pumps[20/] and two low-head recirculation pumps.

2. A containment spray system, which in the event fission products are released to the containment, provides a spray of cool, chemically treated borated water to the containment atmosphere to reduce the pressure inside the containment and also to peovide an elemental iodine removal capability.[21/]

3. A containment air recirculatinn cooling and filtration system which is used for cooling the containment atmosphere and for removing organic iodides. It is a self-contained system equipped with fans, demisters, absolute filters, cooling coils, and charcoal filters.[22/]

4. Redundant hydrogen flame recombiners which, following a loss-of-coolant accident, will function to prevent hydrogen from building up in the containment atmosphere. In this connection, a testing program to confirm acceptable performance of the flame recombiner is being conducted by Westinghouse. Applicant and Westinghouse will continue to

investigate other recombiner concepts to determine their feasibility and performance as new developments arise. Measures will be included in the system design to prevent inadvertent introduction of hydrogen into the containment.[23/]

Applicant has considered the possibility of failure of the reactor vessel as a result of thermal shock caused by action of the emergency core cooling system in the event of a loss of coolant accident during the later portion of vessel life. Provisions have been made in the design of Unit No. 3 to install a reactor vessel cavity-flooding system to provide for covering and cooling the core in case of such a failure. Studies are presently underway to evaluate the likelihood of such a failure, taking into account vessel irradiation levels expected late in plant life. Thermal transients experienced by the hot reactor vessel wall when deluged with cold safety injection water after a loss of coolant accident have been analyzed. Results of this analysis indicate that no loss of reactor vessel integrity would occur even if flaws were assumed to be present in the vessel wall. If research on the effects of thermal shock shows that loss of the reactor vessel integrity is credible, the Applicant intends to install such a system at a future time, after review with the AEC regulatory staff.[24/] The reactor vessel cavity walls will be designed to withstand the mechanical forces which would result if a highly unlikely vessel split were

1 to occur with the primary system pressurized. Design of the
2 facility is such that the reactor vessel could be annealed, if
3 this should become necessary.

4 D. Instrumentation and Control

5 The facility is equipped with a central control room
6 which contains all controls, alarms and instrumentation displays
7 necessary for the safe startup, operation, and shutdown of the
8 plant, as well as for the detection and control of accident
9 situations. The control room is designed to be occupied on a
10 continuous basis, even under accident conditions.[25/]

11 The instrumentation and control systems are designed
12 in accordance with the proposed IEEE criteria for Nuclear Power
13 Plant Protection Systems (IEEE 279).[26/] Protection system
14 redundancy is provided so that no single failure will result in
15 loss of the protective function.[27/]

16 The Unit No. 3 nuclear plant design does not have com-
17 plete separation of control and protection instrumentation but
18 rather makes effective use of process signals for both control and
19 protection, thereby achieving a high degree of functional
20 diversity.[28/] Recognizing the comment by the Advisory Committee
21 on Reactor Safeguards with respect to common failure modes of
22 the instrumentation design, the Applicant and Westinghouse are
23 currently reviewing the instrumentation and control systems
24 for Unit No. 3 with the AEC regulatory staff. The review is

1 directed toward determining the appropriate balance of redundancy,
2 separation, functional diversity, equipment diversity, surveil-
3 lance and qualification testing. The final design of Unit No. 3
4 will reflect the results of this review.

5 Neutron flux distribution information is provided by
6 movable in-core instrumentation and fixed out of core instru-
7 mentation. Information on fuel assembly temperatures at
8 selected core locations is also provided by fixed in-core
9 temperature instrumentation. The plant will have the capability
10 for installation of fixed in-core neutron flux detectors, if
11 operation of large pressurized water reactors indicates that
12 such a system is necessary.[29/]

13 The non-nuclear regulating process and containment
14 instrumentation measures temperatures, pressure, flow and
15 levels in the reactor coolant system, steam systems, contain-
16 ment and other auxiliary systems. The quantity and type of
17 process instrumentation provided ensures safe and orderly opera-
18 tion of all systems and processes over the full operating range
19 of the plant.[30/]

20 Westinghouse is currently engaged in a program of
21 testing performance of instrumentation for prompt detection of
22 fuel failure.[31/] The program will be completed in the fourth
23 quarter of 1969. Applicant will review the results of this
24 program and will select a system to be installed in Unit No. 3

for prompt detection of an abrupt gross failure of a fuel element.

E. Electrical Supplies

Unit No. 3 will be supplied with normal, standby, and emergency power with four separate and independent sources available as follows:

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

2. Stand-by power required during plant startup, shutdown, and after reactor trip will be normally supplied from a 138 kv/6.9 kv station auxiliary transformer supplied from one of the buses at the existing 138 kv Buchanan Substation. This power source is backed up by a 13.8 kv supply from a different bus at Buchanan through a 13.8/6.9 kv transformer. The 138 kv feeder to Unit No. 2 can also be used to supply the 138 kv/6.9 kv Unit No. 3 auxiliary transformer. In addition power will be available from a 21 megawatt on-site gas turbine generator.

3. Emergency power will be automatically available from three on-site emergency diesel generators provided for the exclusive use of Unit No. 3, any two of which can perform the required function. Applicant recognizes that the Advisory Committee on Reactor Safeguards has expressed its belief that the on-site power sources for Unit No. 3 should have greater

1 independence than in the system proposed. In light of this
2 expression of the ACRS's views, the Applicant intends to modify
3 the proposed system, after appropriate review with the AEC's
4 regulatory staff, so that there will be no automatic system for
5 cross-connecting sources and loads.

6 4. Emergency power for instruments and control is pro-
7 vided by 125 volt direct current station batteries.[32/]

8 F. Waste Disposal

9 The waste disposal system will contain all equipment neces-
10 sary to collect, process and prepare for disposal all radioactive
11 liquid, gaseous and solid wastes produced as a result of reactor
12 operation.

13 Liquid wastes will be collected and evaporated, and after
14 appropriate cleaning and filtering, the evaporation condensate will be
15 reused or discharged to the river in such amounts that 10 CFR 20 limits
16 for drinking water will not be exceeded at the common outflow of
17 Indian Point Units 1, 2 and 3. Gaseous wastes will be collected,
18 stored and discharged in such a manner that the combined discharge from
19 all three units will be less than 10 CFR 20 limits.

20 These systems are designed to ensure that there will be no
21 accidental release of radioactive wastes to the environment.[33/]

22 Solid wastes will be packaged and shipped in accordance with
23 applicable governmental regulations for ultimate disposal at an autho-
24 rized location.

G. Fuel Storage and Handling

The fuel handling system is designed to provide a safe, effective means of transporting and handling fuel from the time it reaches the plant in an unirradiated condition until it leaves the plant after post-irradiation cooling. The reactor is refueled with equipment designed to provide careful underwater handling of the spent fuel from the time it leaves the reactor until it is placed in a cask for shipment from the site. Underwater transfer of spent fuel provides an effective, economical and transparent radiation shield, as well as a reliable cooling medium for removal of residual heat.[34/]

VII. ANALYSIS OF POTENTIAL ACCIDENTS

Applicant 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 safeguards 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 double-ended 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 initiated by the coincidence of both low water level and low pressure in the pressurizer or from high containment pressure. For the hypothetical rupture of the largest coolant pipe, injection of borated water ensures sufficient flooding of the core to limit greatly core damage and any resulting zirconium-water reaction. Even in this unlikely event

1 compounded by failure of all external sources of electric power to the
2 plant and the simultaneous occurrence of an earthquake, the facility
3 with its emergency on-site power will be capable of protecting the pub-
4 lic.[2/]

5 In this case the amount of fission products released in
6 the containment would be small when there is full operation of the
7 engineered safeguards on external power.[3/] Even with only on-site
8 emergency diesel power, the fission product release is limited.[4/]
9 The containment isolation system and the pressurized penetration and
10 weld channels essentially eliminate leakage to the environment after
11 the accident. The calculated post-accident releases and off-site
12 exposure levels for both of the above conditions are only a small
13 fraction of the exposure guidelines given in 10 CFR 100.[5/]

14 In another calculation it was further assumed that a
15 failure in the penetration and weld channel pressurization system or
16 in the containment isolation valve sealing system permitted the design
17 leak rate of the containment to exist and release fission products to
18 the environment. To make the evaluation even more conservative, it
19 was postulated that, concurrent with this accident, all external
20 sources of electric power failed and only those safeguards would
21 function which are operable from two of the three on-site diesel-
22 generator units. Even under such extremely improbable conditions,
23 the calculated exposures of the public will still be within the guide-
24 lines of 10 CFR Part 100.[6/]

1 Other mechanical accidents which would have a potential
2 for off-site exposure include the steam generator tube rupture, the
3 secondary system steam line break, a failure in the gaseous waste
4 disposal system and a fuel handling accident.[7/] For these assumed
5 accidents as well, potential off-site exposure is well below the
6 10 CFR 100 guidelines.[8/] Applicant is cognizant of the advice of the
7 Advisory Committee on Reactor Safeguards about the consequence of a
8 fuel handling accident. The Applicant and Westinghouse will review
9 and resolve with the AEC regulatory staff the adequacy and conserva-
10 tism of the analysis of such an accident.
11 Of the reactivity accidents the only one in which some
12 fuel damage could occur is the rod ejection accident.[9/] In this
13 hypothetical accidental rapid withdrawal of a control rod is assumed
14 to result from a rupture of a control rod drive mechanism housing,
15 after which a rod control cluster assembly would be ejected from the
16 core in a very short time by the system pressure. Such a rupture is
17 considered incredible because the housings are of conservative design
18 and initially hydrostatically tested prior to operation; and stress
19 levels in the housing are not affected by systems transients at power
20 or by thermal movement of the coolant loops. If, however, a rupture
21 of control rod mechanism housing were assumed to occur, the resulting
22 loss of coolant would be small compared with the prior discussed
23 hypothetical loss-of-coolant accident.
24 The resultant power pulse following a rod ejection
25 accident is limited by the Doppler reactivity effect of the increased

1 fuel temperature and terminated by reactor trip actuated by high
2 nuclear power signals.[10/] Analyses show that in the event of
3 ejection of the rod of maximum worth further failure of the reactor
4 coolant pressure boundary would not occur and that the resulting
5 power pulse would not cause excessive damage of fuel or other core
6 damage such that the effectiveness of the safety injection system
7 would be impaired. During the development of the final design of
8 the facility further analyses will be made to confirm this conclu-
9 sion.[11/]

10 Unit No. 3 does not share safety-related facilities with
11 either of the other two units on the site. The three units do have
12 a common discharge canal, and there are certain other ties between
13 them such as backup electrical power supplies, city water, and
14 sanitary facilities. Unit No. 3 is therefore virtually independent
15 of the other two units, and an accident at one unit could not cause
16 an accident at another.[12/]

VIII. RESEARCH AND DEVELOPMENT

Research and development programs will be conducted on each of four safety features or components, described below, to be utilized in Indian Point Unit No. 3. This work is being conducted in Westinghouse laboratories, in operating reactors and in AEC facilities. A description of these evolving research and development programs, including a schedule for their completion, is contained in the Preliminary Safety Analysis Report.[1/] This section summarizes the status of these programs as presented in the PSAR.

The schedule for developing this technical information is compatible with the schedule for completion of construction of Unit No. 3. That is, definite results will be available before the plant design is complete, and in time to consider alternatives in development programs and changes in design or in plant operating conditions in the event that the program results do not corroborate their objectives. Considerable information is on hand to indicate that the anticipated program results will be obtained.[2/] The Final Safety Analysis Report will include details on these programs.

A. Core Stability Evaluation[3/]

This program is designed to establish means for the control and detection of potential xenon oscillations in the reactor core and for the shaping of the axial power distribution for improved core performance. Part length control rods have been incorporated

in the design of the core of the Indian Point 3 reactor as a means of controlling xenon oscillations should they occur and to permit a more optimum control of core power distribution. The research and development program includes an evaluation in operating reactors of the use of these rods for controlling xenon oscillations and of various means of detecting such oscillations by out-of-core measuring devices.[4/]

B. Rod Burst Program[5/]

This program will determine fuel clad deformation characteristics and the extent of flow blockage under simulated loss-of-coolant accident conditions. This is to confirm that rod bursting during a loss-of-coolant accident will not cause gross core geometry distortion and that the core will remain in place and essentially intact to such an extent that effective core cooling is not impaired. In addition, experimental data will be obtained on the behavior of the fuel rod during the core reflooding stage of the loss-of-coolant accident in order to establish a realistic upper bound for the peak fuel clad temperature criteria for use in design evaluations. This upper bound is expected to be well above the present peak temperature predictions for the accident conditions.

The overall program consists of the following tests:

A. Rod Burst Tests - Unirradiated Clad

B. Rod Burst Tests - Unirradiated Hydride Clad

C. Quench Tests - Unirradiated Hydride Clad

D. Rod Burst Tests - Irradiated Clad

C. Containment Spray Program 6/

The purpose of this program is the development of design details for a containment spray system utilizing chemically reactive materials to promote radioactive iodine absorption.

Following a loss-of-coolant accident, sprays are actuated in the containment to reduce the pressure of steam. Boric acid is a required constituent of these sprays to prevent dilution of the boron concentration in water collected by the recirculation sump and subsequently used for core cooling. Due to the low solubility of elemental iodine in boric acid compared with that in alkaline solutions it has been decided to mix a sodium hydroxide solution with the boric acid solution in the event of a major accident producing a solution which is an aggressive absorber for iodine when contacted with the containment atmosphere.

It is possible that the containment sprays can remove iodine at a rate sufficient to reduce the integrated two-hour leakage of elemental iodine by a factor of 20-60, depending on physical dimensions and spray rates required for containment cooling. Design of the spray system is being studied and modified to obtain maximum benefit from it and to minimize the leakage of iodine from the containment.

A subsidiary program, which is the study of radiolysis in emergency core cooling water, was initiated as a part of the spray additive evaluation. Results will provide a basis for assessing potential hydrogen buildup when the containment is held isolated

for extended periods of time.

The following technical considerations and areas are being investigated in order to demonstrate the full capability of the spray system:

1. In extending the height of the chamber in which spray absorption takes place, the possibility of more interaction (i.e. coalescence) between droplets arises due to their longer residence time.

2. Simplifying assumptions which were made in preliminary analyses and verified in intermediate size tests, namely, that absorption rate is gas-film controlled, must be reexamined to determine whether liquid phase mass transfer and/or chemical reaction may influence overall absorption rate in a large system.

3. The effect of nonuniformity of spray droplet size on the surface area for absorption has been incorporated in previous performance analyses. It is desired to consider nonuniformity effects on other aspects of the problem, including (in addition to collision frequency mentioned above) the increased residence time of small drops, gas phase mixing, and the depletion of the capacity of small drops to react with iodine due to their smaller volume-to-surface ratio.

4. It must be shown that the use of chemical additives does not promote corrosion or other degradation of the integrity of the containment emergency core cooling system such that the safety function of these systems could be impaired.

5. The maximum rate of hydrogen generation from corrosion or radiolysis of water under post accident conditions must be assessed in order to establish the level of protective action to be taken against the accumulation of a flammable or explosive atmosphere. It is necessary that the basis for such an assessment include any effect on hydrogen production due to the presence of spray additive chemicals.

D. Charcoal Filters for Removal of Organic Iodine 7/

Iodized activated charcoal absorbers (filters) will be installed in this facility to decontaminate the post-accident containment atmosphere with respect to organic iodines. These filters are installed in the Air Recirculation Cooling and Filtration Units and will process part of the air-steam mixture flow after it passes through the cooling coils, demister and filters, and before it is returned to the containment via the ventilation system distribution ducts. The filters reduce organic iodine vapor activity by a process of isotopic exchange.

The effects of water on carbon bed performance are an important consideration, since the reactor containment atmosphere will be near 100 percent relative humidity during the post-accident period. It has been reported that the bed performance for organic iodine decontamination will be significantly reduced at the conditions of near 100 percent relative humidity. This conclusion, however, is not clearly substantiated by the available test data, nor is it supported by a careful examination of the carbon properties for water absorption at the test conditions. A clear distinction has not been made between tests conducted at high humidity conditions and those with test beds flooded with water. To further delineate the effects of moisture on the efficiency of organic iodine decontamination by the charcoal filter system to be installed in this Unit, additional research and development is proposed.

Since test data exist which can be interpreted to reflect low performance at high relative humidity conditions, without clarification of the effects due to flooding, additional tests are being planned. These tests will supplement the existing data and are expected to illustrate more clearly the effects of moisture on bed performance.

The following specific objectives are sought:

A. To show the relationship of filter performance with moisture content when moisture is derived solely from

1 water vapor absorption in a saturated and near-saturated atmos-
2 phere.

3 B. To determine whether sufficient dewatering
4 of a flooded filter can be achieved under saturated condi-
5 tions to restore useful trapping efficiency for organic
6 iodine.

7 In addition to the above-described four programs, other
8 research and development is being conducted primarily to provide
9 technical information which can be applied for component or system
10 optimization in future plants. While these programs will give
11 added confirmation of the conservatism of the proposed design for
12 Unit No. 3, their completion is not essential for the resolution
13 of outstanding safety questions. These programs include the follow-
14 ing:

15 A. Burnable Poison Program
16 B. Saxton Loose Lattice Irradiation Program
17 C. Zorita Irradiation Program
18 D. In-Core Detector Program
19 E. ESADA DNB Program
20 F. Failed Fuel Monitor Program
21 G. Loss of Coolant Analysis Program
22 H. FLECHT (Full Length Emergency Cooling Heat Transfer
23 Test) Program
24 I. Flashing Heat Transfer Program

1 J. Blowdown Forces Program

2 K. Reactor Vessel Thermal Shock Analysis Program 8/

3 The term "research and development" as used in this
4 section is the same as that used by the Commission in Section 50.2
5 of its regulations, as follows:

6 "(n) 'Research and development' means (1)
7 theoretical analysis, exploration or experi-
8 mentation; or (2) the extension of investi-
9 gative findings and theories of a scientific
10 or technical nature into practical application
11 for experimental and demonstration purposes
12 including the experimental production and
13 testing of models, devices, equipment,
14 materials and processes."

IX. QUALITY ASSURANCE

Applicant has a quality assurance plan for the design and construction of the Unit No. 3 facility which is described in the Preliminary Safety Analysis Report as supplemented [1] and, in greater detail, in the Supplement to this Summary of Application. The plan is comprehensive, covering all components, systems and structures important for safety and covering all areas of activity affecting quality, including design (drawings and specifications), manufacture, field erection and installation, preoperational testing, and related activities such as document control, cleanliness control, and shipment, storage and handling of components and equipment.

The plan delineates the quality assurance responsibilities of each organization involved in the project, with emphasis upon the manner in which the Applicant will assure itself of the quality of the completed project. Since this is a "turnkey" arrangement with Westinghouse having the direct responsibility for design and construction, the Applicant carries out its quality assurance responsibilities principally by (1) insuring that its principal contractors (Westinghouse and United Engineers and Constructors) have adequate quality assurance programs and procedures, and (2) monitoring the Westinghouse and United Engineers and Constructors activities in critical areas through an independent detailed vendor surveillance program during manufacture of components, a continuous on-site surveillance program, and a general review of engineering and safety

1 analysis activities. In fulfilling these responsibilities the
2 Applicant utilizes the services of its own personnel; of its
3 quality control and quality assurance surveillance agency, the
4 United States Testing Company; and of its nuclear engineering
5 consultant, Southern Nuclear Engineering, Inc.
6 The internal organization of the companies involved in
7 this project -- particularly the degree of functional independence
8 of groups responsible for quality assurance -- is described in
9 Consolidated Edison's quality assurance plan. Finally, the plan
10 describes the steps which are being taken by Consolidated Edison
11 and the other organizations to establish and document quality
12 assurance procedures in important areas and to establish a system
13 to insure that appropriate quality assurance and quality control
14 records are maintained and accessible.

X. CONCLUSION

The application, as amended, summarized herein has described the preliminary design of Unit No. 3 and has set forth the principal architectural and engineering criteria on which the final design will be based. The major features and components incorporated for the protection of the public have been identified. Aspects of the design and components requiring further research and development have been identified, as have the research and development programs themselves.

The application reflects that Applicant's directors and principal officers are U. S. citizens. The Applicant is not owned, controlled or dominated by an alien, a foreign corportion 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 which might become involved in accordance with the Commission's regulations.[1/]

Consolidated Edison believes that all safety questions will be satisfactorily resolved prior to the completion of the facility and that the application as amended together with this summary adequately show that the proposed facility can be constructed and operated at Applicant's Indian Point site without undue risk to the health and safety of the public.

1 QUALIFICATIONS
2 ARTHUR N. ANDERSON
3 VICE PRESIDENT
4 CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

5 My name is Arthur N. Anderson. My business address
6 is 4 Irving Place, New York, New York 10003.
7 I joined Consolidated Edison in 1933, the year I
8 received my bachelor's degree in mechanical engineering
9 from New York University. I was employed in various
10 production, engineering and construction activities in
11 the Company and in 1951 received a leave of absence to
12 serve with the Atomic Energy Commission in Washington,
13 D. C. in its Reactor Development Division. I was later
14 associated with the Atomic Power Development Associates
15 in Detroit, a nuclear study group working toward the
16 development of a prototype breeder reactor.
17 After rejoining Consolidated Edison I became a
18 general superintendent of the Sherman Creek Electric
19 Generating Station in 1954 and in 1955 became general
20 superintendent of the Waterside Generating Station. I
21 was elected an assistant vice president in 1958 and in
22 1961 was elected vice president of the Company in charge
23 of its construction activities. I am presently vice
24 president of mechanical, nuclear, structural and design

1 engineering.

2 I am a member of the American Society of Mechanical

3 Engineers, the American Gas Association and the Engineers

4 Club.

QUALIFICATIONS
WILLIAM J. CAHILL, JR.
ASSISTANT VICE PRESIDENT
CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

My name is William J. Cahill, Jr. My business address is 4 Irving Place, New York, New York 10003.

I graduated from Polytechnic Institute of Brooklyn with a degree of Bachelor of Mechanical Engineering in 1949. I was employed by Consolidated Edison upon graduation and participated in the design of steam power plants until 1954. In that period I also taught evening classes in heat power engineering and thermo-dynamics at the Polytechnic Institute of Brooklyn. From 1954 to 1956, while on loan to the Knolls Atomic Power Laboratory, I participated in the design of the nuclear power plants for the submarines Seawolf and Triton. I also aided in the test operation of the prototype nuclear plant for the Seawolf and in the construction and operation of a demonstration nuclear electric power plant at West Milton, New York.

During the major part of 1957 I was employed by Nuclear Development Associates in the design of the BR-3 test reactor for installation at Mol, Belgium and in the development program for a sodium cooled-heavy water

1 moderated reactor proposed for Anchorage, Alaska.

2 From late 1957 to the present I have been engaged
3 in assignments of successively increasing responsibility
4 for Consolidated Edison in which I have participated in
5 the design of Indian Point No. 1, the proposed Ravenswood
6 Nuclear Plant, and Indian Point No. 2 and No. 3, as well
7 as several conventional steam power plants.

8 At present I am Assistant Vice President in charge
9 of mechanical, nuclear and design engineering.

10 I am a member of the Technical Committee of ESADA,
11 an organization of New York State Electric Companies
12 engaged in the support of research and development for
13 nuclear power plants, and the Reactor Safety Committee
14 of the Atomic Industrial Forum. I am a licensed pro-
15 fessional engineer in New York State and a member of the
16 New York State Society of Professional Engineers, the
17 ASME and the American Nuclear Society.

1 QUALIFICATIONS
2 JOHN J. GROB, JR.
3 ASSISTANT MECHANICAL PLANT ENGINEER
4 CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

5 My name is John J. Grob, Jr. My business address
6 is 4 Irving Place, New York, New York 10003.
7 I graduated from Stevens Institute of Technology,
8 Hoboken, New Jersey with an ME degree in Mechanical Engineer-
9 ing in 1951. I was employed by Consolidated Edison upon
10 graduation and went on a leave of absence for military ser-
11 vice shortly thereafter. Commissioned an officer in the
12 Civil Engineer Corps of the U. S. Navy, my assignments
13 included Assistant Design Officer and Construction & Main-
14 tenance Officer at the Norfolk Naval Air Station. My
15 assignment as Construction & Maintenance Officer included
16 the administration of several hundred employees in the
17 various construction trades, along with the administration
18 of inspection and other controls on work done by contractors.
19 Upon return to Consolidated Edison in 1955, I par-
20 ticipated in the design of steam power plants. I also par-
21 ticipated in 1957-1958 as a student in a course on nuclear
22 physics and reactor theory which was conducted by the Vitro
23 Corporation for Consolidated Edison for a period of approxi-
24 mately 178 classroom hours. During 1967 I was a student at
25 a two-week course in reactor safety given at MIT under the

1 direction of Dr. T. J. Thompson.

2 For six months during 1960 I was on loan to the
3 Philips Petroleum Company and participated in the operation
4 of the Materials Test Reactor and the Engineering Test
5 Reactor which were operated by Philips Petroleum for the AEC
6 at the National Reactor Testing Station, Idaho. Subsequently
7 I spent four months in the Nuclear Operating Training Program
8 at Shippingport Atomic Power Station, as a trainee.

9 Upon my return to Consolidated Edison in late 1960,
10 I was assigned to the Indian Point Nuclear Generating Station
11 and participated in the preparation of test procedures for
12 systems and components of this plant. I also participated
13 in the performance of these tests and was one of the original
14 Atomic Energy Commission licensed startup crew members for
15 operation of this plant and training of the regular operating
16 crew.

17 In March of 1963 I was reassigned to Consolidated
18 Edison's Mechanical Engineering Department. From that time
19 to April, 1968 I was in charge of the Nuclear Division of
20 this Department and participated in engineering projects for
21 Indian Point Unit No. 1, the proposed Ravenswood Nuclear Plant,
22 Indian Point Units 2 & 3, as well as for the recently announced
23 Nuclear Unit No. 4. Presently I am the Assistant Mechanical

1 Plant Engineer of the Mechanical Engineering Department,
2 directing the work of approximately 50 engineers, engaged
3 in the engineering aspects of power plants for Consolidated
4 Edison.

5 I was a member of the American Standards Asso-
6 ciation N2 Sectional Committee on General and Administrative
7 Standards for Nuclear Energy, and am a member of its suc-
8 cessor USASI N12 Committee. I am a member of a task force
9 chosen by the USASI Standards Committee N45 - (Location,
10 Design, Construction & Maintenance of Nuclear Reactors) -
11 to develop requirements for quality assurance codes and
12 standards for use during the construction phase of nuclear
13 power plants. I am also a member of the Nuclear Task
14 Force of the Edison Electric Institute which reports to
15 the Atomic Power Subcommittee of the EEI Prime Movers Com-
16 mittee.

QUALIFICATIONS
JAMES S. MOORE
ASSISTANT ENGINEERING MANAGER-LICENSING
NUCLEAR ENERGY SYSTEMS
WESTINGHOUSE ELECTRIC CORPORATION

My name is James S. Moore. My residence address is 2021 Garrick Drive, Pittsburgh, Pennsylvania 15235. I am Assistant Engineering Manager responsible for licensing for the PWR Division, Nuclear Energy Systems, Westinghouse Electric Corporation and have served in this capacity since January, 1968. In my present position I have overall responsibility for the licensing activities for all Westinghouse supplied commercial nuclear power reactors. Currently these projects include Zorita (Spain), Beznau (Switerland), Ginna, Indian Point, Turkey Point, Robinson Unit 2, Mihama (Japan), Point Beach, Zion, Cook, Surry, Diablo Canyon, Kewaunee, Prairie Island, North Anna, TVA and Salem.

I was graduated from Carnegie-Mellon University in 1961 with a Bachelor of Science degree in Mechanical Engineering and received a Master of Science degree in Nuclear Engineering from Carnegie-Mellon University in 1963.

I joined the Nuclear Energy Systems in 1956 and have been actively engaged in analysis and design of control

1 system for over eight years. I was lead engineer for the
2 control and protection system on the Enrico Fermi Plant for
3 SELNI at Torino, Italy with on-site participation in the
4 initial power tests. I was also lead engineer for the
5 San Onofre and SENA plants prior to assuming a position as
6 Manager of Systems Transient Analysis in February, 1964.
7 In this position I had overall responsibility for the func-
8 tional design of reactor control and protection system and
9 for the analyses of accident conditions for presentation
10 to licensing authorities.
11 I am a member of the American Nuclear Society,
12 in which I serve as a member of the Standards Executive
13 Committee and as Chairman of the ANS Standards Systems
14 Engineering Subcommittee and serve as a member of the ANS
15 Standards Subcommittee ANS-4, Reactor Dynamics and Control.

1 QUALIFICATIONS
2 OLIVER M. HAUGE
3 MANAGER-PROJECT ENGINEERING
4 CONSOLIDATED EDISON NUCLEAR POWER PLANTS
5 TURNKEY PROJECTS
6 WESTINGHOUSE ELECTRIC CORPORATION

7 My name is Oliver M. Hauge. My residence is 122
8 Cornwall Drive, Pittsburgh, Pennsylvania, 15238. I am the
9 Westinghouse Project Engineering Manager, Turnkey Projects -
10 Nuclear Energy Systems. In this position, I have the overall
11 responsibility for all Westinghouse and Architect Engineer
12 design on the Indian Point Projects for Consolidated Edison
13 Nuclear Power Plants.
14 I was graduated from North Dakota State University
15 in 1951 with a Bachelor of Science degree in Industrial
16 Engineering. Graduate work was also done at the University
17 of California and the University of Idaho.
18 I joined the Atomic Power Divisions in January,
19 1967 and have been associated with the Indian Point Projects
20 since that time. Prior to January, 1967, I was Engineering
21 Manager for Phillips Petroleum Company at the National
22 Reactor Testing Station in Idaho, engaged in nuclear safety
23 research and development for the Atomic Energy Commission.
24 I am a member of the American Nuclear Society and
25 the American Society of Mechanical Engineers.

QUALIFICATIONS
JOHN D. McADOO, JR.
MANAGER
ENGINEERED SAFEGUARDS SYSTEMS
NUCLEAR ENERGY SYSTEMS
WESTINGHOUSE ELECTRIC CORPORATION

My name is John D. McAdoo, Jr. My residence address is 3500 MacArthur Drive, Murrysville, Pennsylvania 15668. I am employed by the Westinghouse Nuclear Energy Systems - Engineering, as Manager of Engineered Safeguards Systems.

I graduated from Carnegie-Mellon University in 1951 with a Bachelor of Science degree in Chemical Engineering.

From 1951 until 1956 I was employed by the Kellex Corporation, later renamed the Vitro Corporation of America, in their Jersey City and West Orange, New Jersey, laboratories where I participated in a variety of research and development projects related to the study of the chemical and physical behavior of uranium and fission products. During that period I held lead responsibility for development work on homogeneous reactor fuel reprocessing under subcontract to the Oak Ridge National Laboratory. Since coming to Westinghouse in 1956, I have held engineering assignments related to systems design for a large

1 homogeneous power reactor, technical coordination of
2 reactor plant engineering, and hazards evaluation. From
3 1961 to 1966 I was engaged in the evaluation of safeguards
4 and potential hazards for the following projects: Yankee
5 Atomic Electric Company Reactor, Carolinas Virginia Tube
6 Reactor, Saxton Reactor, San Onofre Nuclear Steam Generating
7 Station, Connecticut-Yankee Nuclear Plant, Malibu Nuclear
8 Plant, Brookwood Nuclear Station, Indian Point Unit No. 2
9 and Turkey Point Units Nos. 3 and 4. In my present position
10 I am responsible for design of shielding and engineered
11 safeguards systems, and for analysis of loss-of-coolant
12 accidents for all current PWR projects.
13 During my employment at Vitro and Westinghouse
14 I have completed post-graduate courses in nuclear engineer-
15 ing at New York University and in advanced heat and mass
16 transfer and fluid dynamics at Carnegie-Mellon University.

1 QUALIFICATIONS
2 CHARLES GIBSON DURFEE, JR.
3 MANAGER, QUALITY CONTROL AND RELIABILITY
4 PWR SYSTEMS DIVISION
5 NUCLEAR ENERGY SYSTEMS
6 WESTINGHOUSE ELECTRIC CORPORATION

7 My name is Charles Gibson Durfee, Jr. My address
8 is 8983 Eastwood Road, Pittsburgh, Pennsylvania 15235. My
9 position is Manager, Quality Control and Reliability. In
10 this capacity I supervise four managers who in turn super-
11 vise 24 quality control engineers and about 25 other
12 quality control and reliability personnel. My department
13 is responsible for the assurance of quality of all compon-
14 ents within the scope of the nuclear steam supply system.
15 At present, equipment is in manufacture for many projects
16 such as the Consolidated Edison Indian Point Plants 2 and
17 3, the Florida Power and Light Company Turkey Point Plants
18 3 and 4, the Pacific Gas and Electric Diablo Canyon Plants
19 1 and 2 and the Commonwealth Edison Zion Plants 1 and 2.
20 I attended Yale University and was graduated
21 with a Bachelor of Engineering degree in mechanical engineer-
22 ing.
23 I have been employed by Westinghouse for four
24 years in various management and administrative positions
25 for both commercial and Naval nuclear divisions of the company.

1 Prior to that time my experience includes six years in
2 Naval and commercial shipyards building, overhauling,
3 refueling and repairing submarines and other ships, both
4 nuclear and conventional. Assignments included management
5 positions in quality control, production, planning and
6 estimating.

7 I have authored a text, "Shipyard Quality Con-
8 trol", General Dynamics/Electric Boat, 1964, and the
9 quality assurance procedure NAVSHIPS Instruction 9020.32.

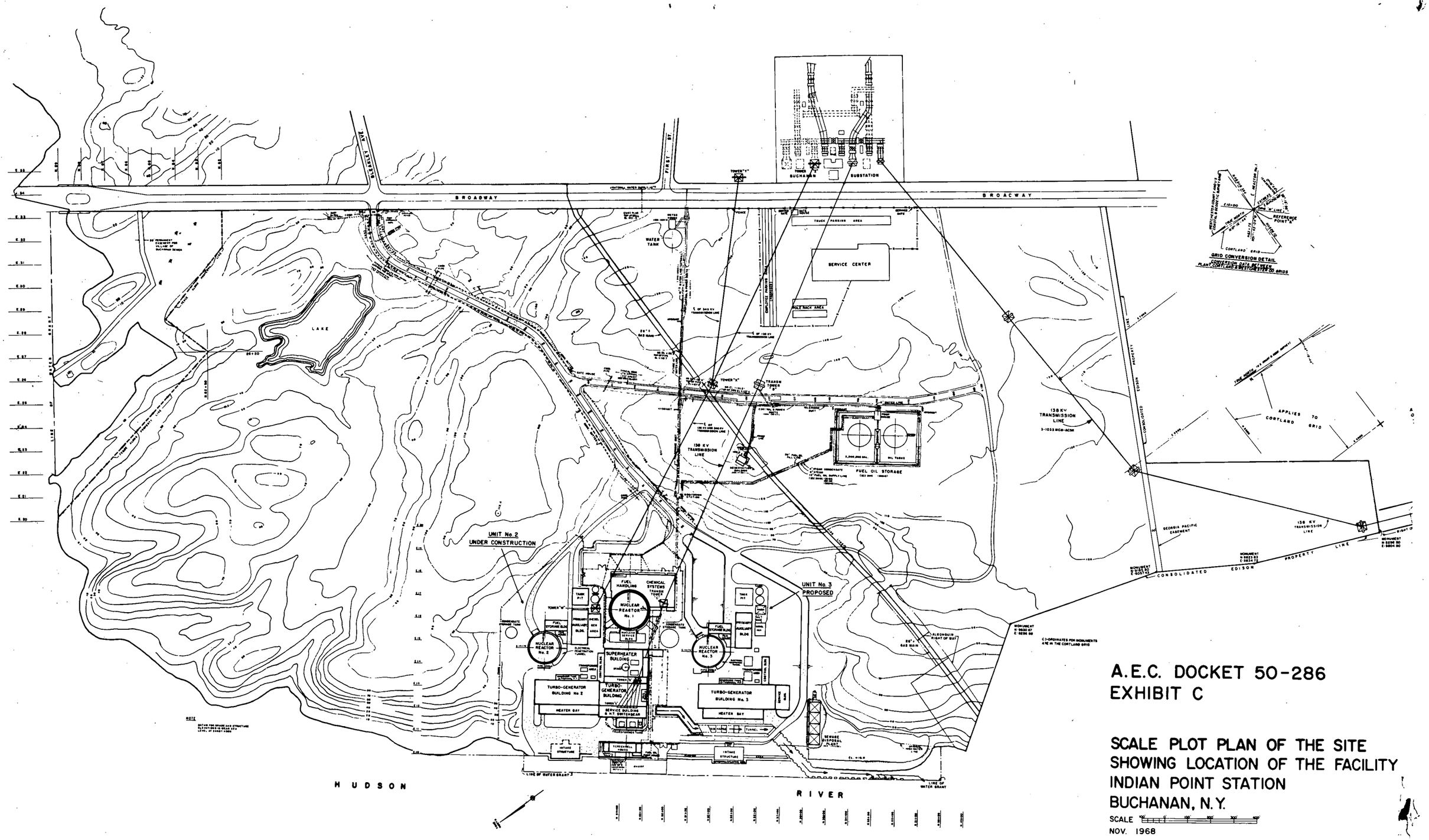
10 I am a member of the Planning Committee of the
11 Quality and Reliability Assurance Advisory Committee of
12 the National Security Industrial Association and am chair-
13 man of the AEC Liaison Subcommittee of that group.

AEC Docket
No. 50-286

Appendix B to

Summary of Application

(Map of Site)



A.E.C. DOCKET 50-286
EXHIBIT C

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

SCALE

NOV. 1968

APPENDIX C

TABULAR COMPARISON OF UNIT NO. 3 WITH OTHER NUCLEAR PLANTS

	Ginna	Indian Point 2	Diablo Canyon	Zion	Indian Point 3
Startup Date	1969	1970	#1 - 1972	#1 - 1972	1972
Reactor Heat Output, MW_t	1300	2758	3250	3250	3025
Net Elect. Output, MW_e	420	873	1060	1050	965
System Pressure, Nominal, PSIA	2250	2250	2250	2250	2250
Coolant Nominal Inlet Temperature, ^oF	551.9	543	539	539	549.9
Coolant Average In-Core Temperature Rise, ^oF	52	55.5	68.6	68.6	63.2
DNB Ratio @ Nominal Conditions	2.15	2.00	1.81	1.81	1.82
Specific Power, Max./Avg.	16.5/4.88 = 3.38	18.4/5.7 = 3.23	18.9/6.7 = 2.82	18.9/6.7 = 2.82	17.6/6.24 = 2.82
Specific Power @ Overpower, kw/ft	18.5	20.6	21.2	21.2	19.7
Total Coolant Flow Rate, lb/hr	67.3×10^6	136.3×10^6	135×10^6	135×10^6	133×10^6
Avg. Velocity Along Fuel Rods, ft/sec	14.7	15.4	15.7	15.7	15.7
Avg. Heat Flux, Btu/hr-ft^2	150,500	175,600	207,000	207,000	193,000
Max. Heat Flux, Btu/hr-ft^2	508,700	567,300	583,000	583,000	543,000
Fuel Assembly Design	RCC Canless 14 x 14	RCC Canless 15 x 15	RCC Canless 15 x 15	RCC Canless 15 x 15	RCC Canless 15 x 15

	Ginna	Indian Point 2	Diablo Canyon	Zion	Indian Point 3
Fuel Rod O.D.	0.422	0.422	0.422	0.422	0.422
Fuel Clad Material	Zircaloy	Zircaloy	Zircaloy	Zircaloy	Zircaloy
Fuel Pellet Material	UO_2	UO_2	UO_2	UO_2	UO_2
Fuel Pellet Diam., Inches	0.3669	0.3669	0.3669	0.3669	0.3669
Fuel Burnup, MWD/MTU (Avg. First Cycle)	14,126	14,200	12,000	12,000	13,600
Control Rod Absorber Material	Cd-In-Ag	Cd-In-Ag	Cd-In-Ag	Cd-In-Ag	Cd-In-Ag
Control Rod Cladding Material	304 SS Cold Worked	304 SS Cold Worked	304 SS Cold Worked	304 SS Cold Worked	304 SS Cold Worked
No. of Absorber Rods per Control Assembly	16	20	20	20	20
Total Rod Worth, Hot	6.8%	8.5%	7.0%	7.0%	7.0%
Moderator Temp. Coefficient $\Delta k/k/^{o}F$ (startup)	0.3×10^{-4} to -3.5×10^{-4}	0.3×10^{-4} to -3.0×10^{-4}	-0.2×10^{-4} to -3.0×10^{-4}	-0.2×10^{-4} to -3.0×10^{-4}	-0.2×10^{-4} to -3.0×10^{-4}
Moderator Void Coefficient $\Delta k/k/\%$ Void	-0.1 to + 0.3 $\Delta k/k/gm/cc$	0.03 to + 0.3 $\Delta k/k/gm/cc$	-0.2×10^{-3} to -3.0×10^{-3}	-0.2×10^{-3} to -3.0×10^{-3}	-0.2×10^{-3} to -3.0×10^{-3}
Doppler Coefficient $\Delta k/k/^{o}F$	-1×10^{-5} to -1.6×10^{-5}	-1.1×10^{-5} to 1.8×10^{-5}	-1×10^{-5} to -2×10^{-5}	-1×10^{-5} to -2×10^{-5}	-1×10^{-5} to -2×10^{-5}

	Ginna	Indian Point 2	Diablo Canyon	Zion	Indian Point 3
Type	Steel-lined, reinforced concrete vertical cylinder flat bottom and hemispherical dome prestressed in vertical direction.	Reinforced steel dome, cylindrical walls and base mats are high-strength deformed billet steel bars.	Reinforced concrete vertical cylinder flat bottom and hemispherical dome.	Concrete reinforced with steel. Cylindrical portion prestressed. Inner steel liner.	Reinforced concrete vertical cylinder flat bottom and hemispherical dome.
Inside Diameter, ft.	105	135	140	140	135
Height, ft.	151.5	215.5	212	212	215.5
Free Vol., ft^3	997,000	2.61×10^6	2.61×10^6	2.61×10^6	2.61×10^6
Reference Incid. Pressure, PSIG (Design)	60	47	47	47	47
Concrete Thickness Vertical Wall, ft.	3-1/2	4-1/2	3-1/2	3-1/2	4-1/2

	Ginna	Indian Point 2	Diablo Canyon	Zion	Indian Point 3
Dome, ft.	2-1/2	3-1/2	2-1/2	2-1/2	3-1/2
Engineered Safeguards		4 Total			
Safety Injection System No. of High Head Pumps	3	3	3	2	3
No. of Low Head Pumps	2	2	2	2	2
No. of Stored Energy Tanks	2	4	4	4	4
Containment Fan Coolers No. of Units	4	5	5	5	5
Air Flow Capacity of each @ Accident Condition, cfm	40,200 flow rate max.	65,000	65,000	65,000	65,000
Post Accident Filters					
No. of Units	2	5	None	None	Yes
Containment Spray					
No. of Pumps	2	2	2	2	2
No. of Diesel Generator Units	2	3	2 per unit + 1 shared	5 shared	3

FOOTNOTE REFERENCES

I. INTRODUCTION
(No references)

II. BACKGROUND; PROJECTED POWER NEEDS; TECHNICAL AND FINANCIAL QUALIFICATIONS

1. Application, Section I
2. Application, Sections I and II; Amendment No. 12 to Application
3. Application, Section III
4. Application, Section II; Amendment No. 9 to Application; Amendment No. 12 to Application

III. FACILITY SITE AND ENVIRONMENT

1. PSAR Section 1.2
2. PSAR Section 1.4, Figure 1.4-1
3. PSAR Section 1.4, Figures 1.4-1 and 1.4-2
4. PSAR Section 1.4, Appendix Part IIIB
5. PSAR Section 1.7
6. PSAR Section 1.8
7. PSAR Section 1.5 and Appendix ;
   PSAR Section 12.2.3
8. PSAR Section 1.5
9. PSAR Section 1.5 and Figure 1.5-2
10. PSAR Section 1.5, Figure 1.5-2

11. PSAR Section 1.5; Appendix prepared by Quirk, Lawler and Matusky Engineers

12. Supplement 7, Answer to Question 1

13. PSAR Section 1.6.1

14. PSAR Section 1.6.2 and Appendix Section 1.1 and 1.2

15. PSAR Section 1.6.2, Figure 1.6-1 and 1.6-2, and Appendix Section 4.1

16. PSAR Sections 1.6.2 and 12.3.3

17. Supplement 1, Answer to Question 6

18. PSAR Section 1.9

IV. <u>EVOLUTION OF DESIGN OF INDIAN POINT UNIT 3 FACILITY FROM DESIGN OF OTHER FACILITIES</u>
(No references)

V. <u>PRINCIPAL ARCHITECTURAL AND ENGINEERING CRITERIA</u>

1. Supplement 1, Item 1

2. <u>Ibid</u>.

VI. <u>DESCRIPTION OF FACILITY AND ASSOCIATED PLANT FEATURES</u>

1. PSAR Section 2.3.11 and table following

2. <u>Ibid</u>.

3. PSAR Section 3.2.3, table 3-6

4. PSAR Section 3.2.1; Supplement 1, Item 4 (page 12); Supplement 1, Item 9

5. PSAR Sections 4.1 and 4.4.1 (pages 4-11)

6. <u>Ibid</u>.

7. PSAR Section 4.4.1 (pages 4-17)

8. PSAR Section 4.4.3

9. PSAR Section 4.4.4

10. Supplement 1, Item 15; PSAR Section 4.1

11. PSAR Section 5.1.2.1

12. PSAR Section 6.2.5

13. PSAR Section 5.2.3

14. PSAR Section 5.2.2

15. Ibid.

16. PSAR Section 5.1.1

17. Supplement 1, Item 16 (E-4.5)

18. Supplement 7, Answer to Question 3

19. Ibid.

20. PSAR Section 6.2.1.3

21. PSAR Section 6.2.2

22. Supplement 7, Answer to Question 5

23. Supplement 5, Item 5

24. Supplement 1, Item 4 (pages 31-33)

25. Supplement 1, Item 16 (E-7.1); Supplement 1, Item 1 (pages 16-17)

26. Supplement 5, Item 10

27. Supplement 1, Item 1 (page 31)

28. Supplement 1, Item 18, Answers 1 and 2

29. PSAR Section 7.5

30. PSAR Section 7.4

31. Supplement 1, Item 4 (pages 20-22)

32. Supplement 1, Item 1 (page 31)

33. PSAR Section 11.1.2.2; Supplement 7, Answer to Question 6(K) (pages 1-2)

34. PSAR Section 9.4

VII. ANALYSIS OF POTENTIAL ACCIDENTS

1. PSAR Chapter 12

2. Supplement 1, Item 15 (page 5);
Supplement 1, Item 17 (Attachment F, Question 1.0)

3. Supplement 7, Answer to Question 5

4. Supplement 1, Item 17 (Attachment F, Question 1.0)

5. Supplement 7, Answer to Question 5

6. Supplement 1, Item 17 (Attachment F, Question 1.0);
Supplement 7, Answer to Question 5 (page 5.4-1)

7. PSAR Section 12

8. Ibid.

9. Supplement 7, Answer to Question 4

10. PSAR Section 12.2.7

11. Ibid.

12. PSAR Section 2

VIII. RESEARCH AND DEVELOPMENT

1. Supplement 1, Item 4

2. Ibid.

3. PSAR Section 3.2.1.1 (pages3-12);
Supplement 1, Item 4 (pages 3-4)

4. Ibid.

5. Supplement 1, Item 4 (pages 5-8)

6. Supplement 1, Item 4 (pages 9-11)

7. Supplement 1, Item 13; Supplement 7, Answer to Question 5

8. Supplement 1, Item 4 (pages 12-33)

IX. QUALITY ASSURANCE

1. Supplement 1, Item 5; Supplement 5, Item 4

X. CONCLUSION

1. Application (pages 1-2); Amendment No. 12 to Application