UNITED STATES OF AMERICA

ATOMIC ENERGY COMMISSION

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

APPLICANT'S PROPOSED FINDINGS OF FACT AND CONCLUSIONS OF LAW IN THE FORM OF A PROPOSED INITIAL DECISION WITH RESPECT TO MOTION FOR 50 PERCENT TESTING LICENSE PART I

FFCL

I. Preliminary Statement

1. This proceeding involves the Application for Licenses filed by Consolidated Edison Company of New York, Inc. ("Consolidated Edison" or "Applicant") with the Atomic Energy Commission ("the Commission") in December, 1965 pursuant to Section 104(b) of the Atomic Energy Act of 1954, as amended ("the Act"). In its application Consolidated Edison requested authority to construct and operate a pressurized water nuclear reactor facility designated as Indian Point Unit No. 2 ("Unit No. 2") at its Indian Point site in the Village of Buchanan, Westchester County, New York.1/

8111120157 720128
PDR ADOCK 05000247
G PDR

1/ Application for Licenses, As Amended, Applicant's Exhibit No. 1A (introduced into evidence Tr. 377) (all transcript references contained herein refer to corrected pagination); Summary of Application, Applicant's Exhibit No. 1C (introduced into evidence Tr. 377), pp. 1-2 (all references to the Summary of Application incorporate the footnotes contained therein).

2. After review of the application by the Regulatory Staff of the Commission ("the Staff") and the Advisory Committee on Reactor Safeguards ("the ACRS") a hearing was held pursuant to the Notice of Hearing published on July 30, 1966 in the Federal Register (31 F.R. 10,331). This hearing was held in Buchanan, New York before a duly constituted Atomic Safety and Licensing Board on those issues set forth in the published notice. At the conclusion of the hearing and after the Board's Initial Decision, the Commission issued Construction Permit No. CPPR-21 dated October 17, 1966 to Consolidated Edison. Pursuant to that authority Applicant commenced construction of its Unit No. 2 facility.[2/]

3. On October 15, 1968 Consolidated Edison filed with the Commission its Final Facility Description and Safety Analysis Report ("FSAR"). Since that filing Applicant has amended its application and supplemented that report a number of times and has introduced testimony

---

2/ Consolidated Edison Co., Docket No. 50-247 (Initial Decision, Oct. 3, 1966) pp. 2-3; Summary of Application, page 2.

setting forth those features incorporated into the final design of the facility.[3/] The application and its amendments have been reviewed by the Staff and the ACRS.

4. Consolidated Edison, by a letter to the Commission dated October 5, 1970 requested that a public hearing be held at the operating license stage. The Commission issued a Notice of Hearing on November 13, 1970 which was duly published in the Federal Register (35 F.R. 7679-80, Nov. 17, 1970). This notice specified those issues which pursuant to Section 50.57(a) of the Commission's regulations were to be considered and determined at the hearing for the operating license.

5. Pursuant to the notice a prehearing conference was held in Montrose, New York on December 1,

---

3/ Application for Licenses, Amendments No. 9-20, 22-25; Listing of Design Changes to the Indian Point Unit No. 2 Plant, pp. 1-4 (follows Tr. 3834); Answers of Applicant to Questions Raised by ASLB on March 24, 1971, Part II, Question 7, pp. 2-4 (follows Tr. 888). The Summary of Application contains information on the following subjects: Site and Environment, Description of Facility and Associated Plant Features, Evolution of Design of Unit No. 2 from Design of Other Facilities, Quality Assurance, Safety Analyses, Items Requiring Further Information or Development During Construction, Technical Specifications, Conduct of Operations, Initial Tests and Operation, Financial Qualifications and Alien Control - Access to Restricted Data.

1970, followed by twenty-five days of hearings (December 17-18, 1970, January 19, March 24, May 13, July 13-16, July 21, October 5, November 1-4, November 8-12, November 16-17, December 14, 1971 and January 11-12, 1972).

6. In addition to the Applicant and the Staff five persons participated as parties to the proceeding by way of intervention. Three of the interventions - those by the Citizens Committee for Protection of the Environment, the Environmental Defense Fund, Inc. and the Hudson River Fishermen's Association - were in opposition to the application. This is therefore a "contested proceeding" within the meaning of Section 2.4(n) of the Commission's Rules of Practice. The fourth and fifth - the Atomic Energy Council of the State of New York and the State of New York - chose to intervene under Section 2.714 of the Rules of Practice of the Commission but did not take a position with respect to the application. A sixth person - Mary Hays Weik - was permitted to intervene in opposition to the application but withdrew her intervention shortly after the hearing began. In addition numerous persons participated in the hearing by way of limited appearances pursuant to Section 2.715(a) of the Commission's Rules of Practice.

7. The entire application as amended has been reviewed by the Staff and the ACRS, and the conclusions of both are favorable to the granting of an operating license.[4/]

8. On September 24, 1971 Applicant filed a motion pursuant to 10 CFR 50.57(c) and 10 CFR 50, Appendix D, Section D.2, requesting the Board, among other things, to authorize the Director of Regulation to issue a license to Applicant to operate Unit No. 2 at a steady state power level of 2482 megawatts thermal (i.e., 90 percent of full power) pending completion of the review required by the National Environmental Policy Act of 1969 ("NEPA"), or such other power level short of full power operation as the Board determines to be justified on the basis of the record in this proceeding. This motion was supplemented on October 19, 1971, at which time Applicant requested that the Board, among other things:

(a) Consider first and on an expedited basis the evidence as it relates to the issuance of a license for limited operation of Unit

---

4/ Staff Safety Evaluation ("SSE"), pp. 2-3, 79-80, 88-92 (follows Tr. 405).

No. 2 for testing purposes at a power level up to 1379 Mwt (50 percent of full power);

(b) Order that any hearing with respect to the issuance of such a license shall commence immediately following the conclusion of the hearings on radiological safety issues commencing on November 1, 1971, and that the hearing on the limited operation license shall not exceed three days in length;

(c) Issue an Initial Decision and Order authorizing the Director of Regulation immediately to issue a license to Applicant to operate Unit No. 2 for testing purposes at a power level up to 551 Mwt (20 percent of full power), and further authorizing the Director of Regulation to issue a license to Applicant to operate Unit No. 2 for testing purposes at a power level up to 1379 Mwt (50 percent of full power);

(d) Refer the Board's order to the Atomic Energy Commission for its specific approval insofar as it authorizes operation of Unit No. 2 at a power level in excess of 551 Mwt.

9. This is an Initial Decision with respect to section (a) of the October 19, 1971 supplement to Applicant's motion for issuance of a license authorizing limited operation of Unit No. 2, i.e. a motion for authority to operate Unit No. 2 for testing purposes at power levels up to 50 percent of full power. Only the Citizens Committee is opposing the issuance of such a license.[5/] Pursuant to the Stipulation, the Applicant and the Citizens Committee are relying upon the entire record in this proceeding to date for purposes of the Board's ruling upon the Applicant's above-referenced motion.

10. The nature and expected duration of the tests which Applicant wishes to conduct in accordance with its motion for authority to operate Unit No. 2 for testing purposes up to 50 percent of full power are

---

5/ Response of the Citizens Committee for the Protection of the Environment to Applicant's Motion for Issuance of a License Authorizing Limited Operation, served October 1, 1971; Stipulation Among Applicant and Intervenors CCPE, EDF and HRFA Concerning Further Conduct of Proceedings ("Stipulation"), served November 4, 1971.

set forth in the FSAR and Applicant's testimony. The duration of the testing program is not reasonably expected to exceed 100 days.6/

6/ FSAR, Sections 13.3.1 to 13.3.3; Testimony of Applicant in Support of its Motion for Issuance of a License Authorizing Limited Operation dated October 19, 1971 ("Applicant's Oct. 19, 1971 Testimony"), pp. 1-16 (follows Tr. 4013); Tr. 4705-06.

II. Financial and Technical Qualifications of Applicant and Related Matters

11. The operation of Unit No. 2 will be financed, as was its construction, as an integral part of the Applicant's regular course of business. The record indicates that the Applicant is one of the largest privately owned gas and electric utilities in the United States. There is no doubt that the Applicant is financially qualified to carry out the activities for which it seeks authorization.[7/]

12. Experience, technical depth and planned training programs and procedures provide the requisite technical qualifications for Consolidated Edison to operate Unit No. 2. The organization which has proven effective during the more than eight years of safe operation of Unit No. 1 has been expanded to provide for the administrative and technical needs of Unit No. 2. Applicant's engineering personnel have participated in the design and Commission review of both Units No. 1 and 2. Extensive training programs for supervisory personnel,

---

7/ Summary of Application, page 80; Application for Licenses, Amendment 21; SSE, pages 74-75, 116 (follows Tr. 405).

control room operators and key maintenance personnel have continued since December 1968 under the direction of the Applicant, equipment manufacturers and the Public Health Service.[8/] The Nuclear Facilities Safety Committee will review, audit and inspect all aspects of the operation of the plant and changes in the facility.[9/] Complete records of facility operations will be maintained at the plant.[10/]

13. Applicant has satisfied its financial protection requirements under 10 CFR Part 140 of the Commission's regulations by furnishing to the Commission proof of financial protection in the amount of $82,000,000, the maximum amount required by the Commission's regulations for a full power license for a facility of this size in the form of a Mutual Atomic Energy Liability Underwriter's Policy No. MF-29 and a Nuclear Energy Liability Insurance Association Policy No. NF-100, each of which is on file with the

---

8/ Summary of Application, pp. 60-62; SSE, pp. 64-67 (follows Tr. 405); FSAR, Section 12.2, Questions 12.1, 13.4; Appendix A to Proposed Facility Operating License - Technical Specifications and Bases ("Tech. Specs."), No. 6.1A, Supplement No. 1 to Staff's Exhibit No. 1 (introduced into evidence Tr. 678).

9/ Summary of Application, pp. 62-64; SSE, pp. 66-67 (follows Tr. 405); Tech. Spec. No. 6.1.B.1.

10/ Summary of Application, pp. 62-63; Tech. Spec. No. 6.5; FSAR, Section 12.4, Question 12.4.

Commission. In addition Applicant has entered into Indemnity Agreement No. B-19 with the Commission which provides protection in the amount of $478,000,000. Therefore Applicant has provided aggregate financial protection of up to $560,000,000.[11]

14. Consolidated Edison is a public utility corporation organized and existing under the laws of the State of New York. Its principal offices are at 4 Irving Place, New York, New York. All directors of Consolidated Edison are citizens of the United States. Except for one officer, all officers of Consolidated Edison are citizens of the United States. The officer who is not an American citizen is a citizen of Canada. Consolidated Edison is not owned, controlled or dominated by an alien, a foreign corporation or a foreign government. The activities to be conducted do not involve any Restricted Data. The Applicant has agreed to safeguard any such data which might become involved in accordance with the requirements of

---

11/ Testimony of John F. Dembeck, pp. 1-2 (follows Tr. 382); SSE, pp. 76-78 (follows Tr. 405); 36 F.R. 20,621-22 (Oct. 27, 1971); Tr. 401-402, 4586-87.

Part 50 of the Commission's regulations. The Applicant will obtain fuel from sources of supply available for civilian purposes so that no diversion of special nuclear material for military purposes is involved.[12/]

12/ Summary of Application, page 81; SSE, page 73 (follows Tr. 405).

III. Safe Operation

15. The Applicant is required to comply with the technical specifications for this facility which will be incorporated in the operating license. These technical specifications contain the necessary guidelines to assure that the plant will be operated in a safe manner and that such operation will comply with the requirements of the Atomic Energy Act and of the Commission's rules and regulations.13/

16. The technical specifications do not permit considerations of the need for power, however pressing, to be used as a basis for departing from the operating practices and procedures specified therein.14/

17. In addition, extensive operator training is directed toward operation of Unit No. 2 in a safe condition regardless of other considerations.15/

13/ Summary of Application, pp. 57-59; SSE, page 70 (follows Tr. 405).

14/ Additional Testimony of Applicant, Part II, dated July 8, 1971, pp. 1-2 (follows Tr. 894); FSAR, Sections 12.3, 12.5; Tr. 1387-91.

15/ Additional Testimony of Applicant, Part II, dated July 8, 1971, pp. 1-2 (follows Tr. 894); FSAR, Section 12.2; FSAR, Question 12.1; Tr. 1386-89, 1428-29.

18. There is adequate assurance that any need for power from Unit No. 2 will not compromise the safe operation of the facility.16/

19. The plant is designed with safety equipment independent of the control of the operator which provides for automatic scram of the reactor and shutdown of the plant if unsafe operating conditions are approached.17/

20. The presence or absence of load frequency control on Indian Point Unit No. 2 would not significantly affect the safe operation of the facility.18/

16/ References in footnotes 14-15.

17/ FSAR, pp. 7.2-1 to 7.2-2, 7.2-21 to 7.2-30; Tr. 1418-19.

18/ Set J Responses, Questions J-4(a) and (d); Applicant's Responses to Round Two Questions Submitted by the CCPE on March 9, 1971, Part I, dated March 29, 1971 ("Set H Responses") Question H-21, CCPE Exhibit H (introduced into evidence Tr. 929); Tr. 1403-09.

IV. Status of Facility Construction

21. Since the issuance of Provisional Construction Permit No. CPPR-21 inspections by the Compliance Division of the Commission have been conducted at the construction site, vendor shops and the Applicant's offices to verify that the licensee has met its safety responsibilities and that the completed facility conforms to the construction permit, the application as amended and the rules and regulations of the Commission.[19/] After having determined that the facility had been constructed in accordance with the application and the construction permit for a license permitting fuel loading and subcritical testing the Division of Reactor Licensing issued Facility Operating License No. DPR-26 on October 19, 1971 (36 F.R. 20,621-22, Oct. 27, 1971).

22. Certain equipment and instrumentation contained in the Primary Auxiliary Building ("PAB") for Unit No. 2 were damaged by a fire which occurred in the PAB on

---

19/ Staff Safety Evaluation, Supplement No. 1 ("SSE, Supp. 1") pp. 1-6, App. A (follows Tr. 405); Staff Safety Evaluation, Supplement No. 2 ("SSE, Supp. 2"), App. A (follows Tr. 914).

November 4, 1971.[20/] The Applicant has kept the Board and all parties to the proceeding informed regarding the inspection, testing and restoration program undertaken to repair the damage resulting from this fire.[21/] Qualified engineers, technicians and construction and technical specialists are conducting an extensive analysis of the effects of the fire. This analysis includes visual inspection of the entire PAB and other appropriate areas of the facility and testing of equipment and components as appropriate to determine the type and extent of damage.[21a/] Based upon the results of this inspection and testing, the Applicant has undertaken a complete repair and restoration program which includes

---

20/ Tr. 3811-13, 3815, 4195.

21/ See Tr. 4014.

21a/ Indian Point Station Unit No. 2 Restoration Plan for Primary Auxiliary Building & Equipment, December 6, 1971 ("Restoration Plan"), Sections B, D (follows Tr. 4169); Responses to "Interrogatories of the CCPE Directed to the Applicant with Respect to the Fire and Personnel Selection Procedures" Supplied to Applicant on December 23, 1971 ("Fire and Personnel Interrogatories"), Questions 1-2, 5-7, 13-14, 16-20, CCPE Exhibit EE (introduced into evidence Tr. 4479); Tr. 3817, 4485-86, 4491-96, 4513-14, 4516-17.

structural, electrical, mechanical and functional testing of individual components as well as functional tests on individual systems. In addition, safeguards systems located in the PAB will be tested in a functional or operational manner prior to initial criticality. Following criticality and as a part of start-up testing, auxiliary systems in the PAB will be functionally tested. The procedures for strict quality control included in the program and the review of the implementation of the program by the Nuclear Facilities Safety Committee and the Division of Compliance provide reasonable assurance that the safety and reliability of the equipment contained in the Primary Auxiliary Building will be restored in accordance with the approved design as contained in the FSAR.[21b/] In its presentation the Applicant has demonstrated that it is highly improbable that a similar fire would occur either

---

21b/ Restoration Plan, Sections A (page 2), B, C, D (follows Tr. 4169); Fire and Personnel Interrogatories, Questions 1-3, 7-15, 19-21 (CCPE Exhibit EE); Letter to Dr. Morris from Mr. Caldwell, dated December 6, 1971 (follows Tr. 4169); Tr. 3825-27, 3833, 4180-4183, 4186, 4189, 4190-92, 4198, 4208-09, 4211, 4497-4500, 4505-12.

during operation of the facility or after Unit No. 2 reaches criticality. The design of the facility, however, provides for a safe shutdown of the reactor if a fire or other event led to an abnormal condition in which a safety limit were reached.[21c/]

23. The evidence concerning the status of construction adduced by the Applicant and Staff during the course of the hearing adequately demonstrates that construction of Unit No. 2 has been substantially completed in accordance with the provisions of the construction permit, the application as amended, the provisions of the Act and the rules and regulations of the Commission.[22/] Before an operating license authorized by this Initial Decision is issued, the Compliance Division of the Commission will verify the completeness of construction

---

21c/ Restoration Plan, Section A (page 1) (follows Tr. 4169); FSAR, pp. 7.2-1 to 7.2-2, 7.2-21 to 7.2-30; Tr. 3813, 3823-24, 4178, 4194.

22/ Testimony of Alex C. Husband, page 3 (follows Tr. 382); Testimony of John T. Stiefel, pp. 2-14 (follows Tr. 382); Affidavit of William J. Cahill, Jr. dated June 16, 1971, submitted with Motion of Applicant for an Order to Permit Fuel Loading and Sub-Critical Testing dated June 18, 1971, pp. 2-3; Tr. 692-94, 800, 1466-68, 1470-73, 1954-55; SSE, Supp. 1, pp. 22-23 (follows Tr. 405); SSE, Supp. 2 (follows Tr. 914); references in footnotes 19-21c.

of Unit No. 2 for safe operation at the authorized power level.[23/]

23/ SSE, page 80 (follows Tr. 405).

V. Facility Site

24. Unit No. 2 is located on a 239-acre site on the east bank of the Hudson River in the Village of Buchanan, Westchester County, New York. The Applicant is the sole owner of the entire property. Unit No. 2 is located adjacent and to the north of Applicant's Indian Point Station Unit No. 1. The proposed Unit No. 3 will be located adjacent and to the south of Unit No. 1. The site is approximately 24 miles north of the New York City boundary line. The nearest city, Peekskill, is 2.5 miles northeast of Indian Point.[24/]

25. The area surrounding the Indian Point site is generally residential with some large parks and military reservations. The majority of the area to the east of the river within fifteen miles of the site is zoned for residential usage. West of the river within a fifteen-mile radius the Palisades Interstate Park and residential areas are the dominant land usage. The only agricultural areas within fifteen miles are south or northwest of the plant on the west side of the river.[25/]

24/ Summary of Application, pp. 1, 4; SSE, page 7 (follows Tr. 405); FSAR, Sections 1.2.1, 2.2.1, 2.2.3.

25/ Summary of Application, page 4; FSAR, Section 2.4.

26. The 1960 population within a 15-mile radius of the site was approximately 327,000. The 1980 estimated population is approximately 670,000. Within a 5-mile radius of the facility approximately 55,000 people now reside. This number is expected to increase to approximately 108,000 by 1980. The largest concentration is in the City of Peekskill where the population is approximately 19,000. The estimated 1980 population for Peekskill is 30,000. Only 1,080 people live within one mile of Unit No. 2. Only 46 people reside within 1/2 mile.[26/]

27. The minimum distance from the reactor center line to the boundary of the site exclusion area and the outer boundary of the low population zone ("LPZ") as defined in 10 CFR Part 100 is 520 meters and 1100 meters respectively. The "Restricted Area" as defined in 10 CFR Part 20 is the same as the "Exclusion Area."[27/]

28. Applicant's consultants have conducted extensive studies of the meteorology, geology, hydrology and seismology

---

26/ Summary of Application, page 4; SSE, page 7; FSAR, Sections 1.2.1, 2.4.

27/ SSE, page 7-8 (follows Tr. 405); FSAR, Section 2.2; Tech. Spec. No. 5.1.

of the Indian Point site and the surrounding area.[28/] In light of these studies and recent evaluations relating to potential flooding that could occur at the site, the Applicant, the Staff and the ACRS have concluded that Unit No. 2 can be operated at the Indian Point site without endangering the health and safety of the public.[29/]

28/ Summary of Application, pp. 4-10; FSAR, Sections 2.2 to 2.9.

29/ SSE, pp. 9-11, 88-89, 93-98 (follows Tr. 405).

VI. Facility Features

29. Unit No. 2 utilizes a pressurized water reactor with an initial rating of 2758 megawatts thermal and 873 megawatts net electric. The reactor coolant system will operate at a nominal pressure of 2250 psia and an average temperature of 569.5 degrees F. The reactor core is approximately eleven feet in diameter and twelve feet in height. It will contain 193 fuel assemblies, each containing 204 fuel rods in square array and held in place by grids. These fuel rods are manufactured from zircaloy tubes and loaded with fuel pellets of slightly enriched uranium dioxide. Core reactivity will be controlled by a combination of fixed burnable neutron absorbing rods, movable absorber rods and neutron absorber (boric acid) dissolved in the coolant. Four cooling loops are used to carry the heat from the reactor.[30/] Reactors of similar design including those at Indian Point Unit No. 1, Connecticut Yankee, San Onofre, and Ginna have demonstrated successful and safe operation.[31/]

30. The containment for Unit No. 2 consists of a reinforced concrete structure lined with steel plate. The

---

30/ Summary of Application, pp. 11-14; SSE, pp. 4-5 (follows Tr. 405).

31/ Summary of Application, pp. 33-34; SSE, page 14 (follows Tr. 405).

containment completely encloses the reactor and reactor coolant system.[32/] The containment structure for Unit No. 2 is designed to withstand the combined effects of an earthquake and the most severe loss-of-coolant accident. Missiles generated by a tornado, a turbine-generator failure or a primary system failure will not penetrate the containment structure.[33/] The containment is designed to assure an essentially leak tight containment system. This is assured by the inclusion of an isolation valve seal water system and a containment penetration and weld channel pressurization system.[34/] The containment has been tested for structural integrity and leak tightness.[35/]

31. Other engineered safeguards are utilized in Unit No. 2. These include:

(a) Emergency core cooling system.

(b) Containment spray system.

32/ Summary of Application, page 15; SSE, page 5 (follows Tr. 405).

33/ Summary of Application, pp. 16-17; SSE, pp. 30-37 (follows Tr. 405).

34/ Summary of Application, pp. 15-16; SSE, page 6 (follows Tr. 405).

35/ SSE, Supp. No. 2, page 4 (follows Tr. 914).

(c) Containment air recirculation cooling and filtration system.

(d) Hydrogen control system.

Applicant has provided evidence concerning the research and development that has been carried on pertaining to the engineered safeguards and other features in Unit No. 2 and has shown that such research and development has been satisfactorily concluded.[36/] In addition to the engineered safeguards enumerated above the reactor protective system provides for the protection of the reactor core and reactor coolant systems during all phases of plant operation. The protective system shuts down the reactor power output by dropping the control rod assemblies into the core and starting any engineered safeguards that may be required to maintain a safe condition.[37/] Redundant on-site and off-site power sources are available to supply emergency power to the safeguard systems. The on-site sources consist of three emergency diesel generators for Unit No. 2, any two of which can carry all loads required for safety under normal or accident conditions. An additional emergency

---

36/ Summary of Application, pp. 48-56; Answers of Applicant to Questions Raised by ASLB on January 19, 1971 ("ASLB--Jan. 19, 1971"), Part II, Question 16 (follows Tr. 665); FSAR, Section 1.5.

37/ Summary of Application, pp. 20-22; SSE, pp. 6, 46-48 (follows Tr. 405).

power supply is available from three gas turbines located on-site and at the Buchanan substation next to the site.[38/]

38/ Summary of Application, pp. 22-24; SSE, pp. 48-50 (follows Tr. 405); Tech. Spec. No. 3.7; FSAR, Section 8.2.

VII. Normal Operation

32. Consolidated Edison has designed and will operate Unit No. 2 including the waste disposal system in such a manner that all releases of radioactive effluents will be kept as low as practicable within the limits specified in Part 20 of the Commission's regulations for normal operations and for any transient situation that might reasonably be anticipated.[39/] In this regard the Board notes the fact that for more than eight years Unit No. 1 has operated in such a manner that all radioactive releases from the site have been within the requirements of Part 20.[40/] Applicant will operate Units No. 1 and 2 so that the combined releases from these plants will also be within the limits defined by Part 20.[41/]

39/ Summary of Application, pp. 24-28; SSE, pp. 53-55; ASLB--Jan. 19, 1971, Part I, Questions 15, 21, 36, 37, 38 (follows Tr. 665), ASLB--March 24, 1971, Part II, Question 7 (follows Tr. 888); FSAR, Section 11; Tech. Spec. No. 3.9.

40/ ASLB--Jan. 19, 1971, Part I, Questions 1, 11, 28 (follows Tr. 665); ASLB--March 24, 1971, Part I, Question 1 (follows Tr. 728); Tr. 2228-29.

41/ Summary of Application, page 7; SSE, page 53 (follows Tr. 405); ASLB--Jan. 19, 1971, Part I, Question 21 (follows Tr. 665); Tech. Spec. No. 3.9.

33. In order to provide additional assurance that the design objectives for the waste disposal system are met and to maintain all radioactive releases to unrestricted areas at a level which is as low as practicable, the Applicant has provided several design feature modifications for Unit No. 2. The leakage from the reciprocating charging pumps will now be collected and returned to the chemical and volume control system and thus will reduce the load on the evaporator. Applicant is providing a modified bellows seal assembly on the pressurizer spray valves in order to eliminate leakage from these valves and, therefore, reduce the load on the evaporator. The waste disposal evaporator is being modified to improve the capacity and decontamination factor and a polishing demineralizer/filter installation is being provided as a backup for condensate cleanup. Charcoal filters are being provided for the plant vent to remove radioactive gaseous iodine from containment atmosphere purge in the event of radioactive iodine in the primary coolant concurrent with leakage of radioactive primary coolant to the containment atmosphere. An intertie is being provided between the Unit No. 2 steam generator blowdown lines and the Unit No. 1 blowdown purification system to reduce significantly liquid

radwaste releases from Unit No. 2 in the event of steam generator leakage.42/

34. The Applicant will conduct a comprehensive environmental monitoring program with respect to the operation of Unit No. 2.43/ This program of surveys of radioactivity has continued since 1958 and will continue throughout the operation of Units No. 1 and 2. The program includes continuous sampling of atmospheric dust, Hudson River water, and measurement of the gross gamma background on the Indian Point site. Surface water from a small lake on the site and drinking water from the Indian Point tap and nearby reservoirs are sampled weekly. The program also includes samples of milk, Hudson River water upstream and downstream from the site, vegetation on the site, marine life from the river and water from the Indian Point well. Vegetation and soil samples are taken regularly during the growing season and gamma

---

42/ ASLB March 24, 1971, Part II, Question 7 (follows Tr. 888).

43/ Summary of Application, pp. 8-10; SSE, pp. 11-13 (follows Tr. 405); ASLB--Jan. 19, 1971, Part I, Questions 1, 5, 6, 7, 23 (follows Tr. 665); ASLB--March 24, 1971, Part I, Questions 1, 5, 23 (follows Tr. 728); Tech. Spec. No. 4.10.

radiation surveys on roads in the station vicinity are made. In addition, survey programs undertaken by New York State and New York University provide additional assurance that Unit No. 2 will comply with 10 CFR Part 20.[44/]

44/ Summary of Application, page 10; SSE, page 12 (follows Tr. 405).

tank, a fuel handling accident and a rod ejection accident. In these assumed accidents as well, conservatively calculated off-site exposure is below the 10 CFR Part 100 guidelines.[45/]

35a. The consequences of serious postulated accidents which might occur during testing of Unit No. 2 at up to 50 percent of full power have been analyzed by the Applicant in terms of calculated doses at the site boundary. These calculations show that doses would be far below the guidelines of 10 CFR Part 100, and even well within 10 CFR Part 20 levels. These calculations were made by the Applicant using more realistic assumptions concerning the consequences of accidents than are presented in the Staff Safety Evaluation. However, utilizing the conservative assumptions applied in the Staff Safety Evaluation, calculated doses as the result of postulated accidents which might occur during the testing activities would be lower than the values presented in the Staff Safety Evaluation for equilibrium operation at full power, since the maximum fission

---

45/ Summary of Application, pp. 43-47; SSE, pp. 59-63; FSAR, Section 14.0, Questions 14.1, 14.4.3, 14.6.

product inventory in the reactor during the testing activities will at most be 50 percent of that assumed in the Staff's analyses. The actual inventory of most fission products would be even less than 50 percent of that assumed in the Staff's analyses due to the limited time at power during the test program.[45a/]

36. During the course of the hearing the Citizens Committee has contended that rupture of the reactor vessel was an accident against which the facility must be designed. The Applicant has responded to this contention and the inquiries of the Board with testimony from a highly qualified panel of seven expert witnesses in the field of pressure vessel technology.[46/] This

---

45a/ Applicant's Oct. 19, 1971 Testimony, pp. 1-2, 16-20 (follows Tr. 4013).

46/ Tr. 1933-1936.

evidence and evidence produced by the Staff provide adequate assurance that failure of the reactor vessel will not occur, and therefore that it need not be considered a design basis accident for Unit No. 2.[47/]

47/ ASLB--Jan. 19, 1971, Part I, Questions 2, 3, 10, 13, 34 (follows Tr. 665); ASLB--March 24, 1971, Part I, Question 2 (follows Tr. 728); ASLB--March 24, 1971, Part II, Questions 3, 11 (follows Tr. 888); Responses of the DRL to the Questions of the ASLB at the Hearing Session dated January 19, 1971 ("ASLB--Jan. 19, 1971--Staff"), pp. 4-6 (introduced into evidence Tr. 917); Answers of Applicant to Questions Raised by ASLB on May 13, 1971, dated July 6, 1971 ("ASLB--May 13, 1971"), Questions 9, 10 (follows Tr. 890); Set H Responses, Part I, Questions H-1, H-11, H-15; Set H Responses, Part II, dated March 31, 1971, Question H-43 (CCPE Exhibit H); Additional Testimony of Applicant Concerning Reactor Vessel Integrity, dated September 17, 1971 ("Reactor Vessel Integrity Testimony") (follows Tr. 1932); Report by the AEC Regulatory Staff in Response to ASLB Questions Concerning Reactor Vessel Integrity and "Additional Testimony of Applicant Concerning Reactor Vessel Integrity (September 17, 1971)," dated October 26, 1971 ("Staff Report on Pressure Vessel"), pp. 1-3 (introduced into evidence Tr. 2715); Tr. 2032-2058, 3934-3946, 3964-68, 3948-3950, 3952, 3954-64; 3968-77, 3979-81.

VIII. Accident Analyses

35. Having analyzed the consequences of a variety of assumed accidents and equipment failures, Applicant has demonstrated that in the event of any of those postulated accidents public exposure would be well within the guidelines set forth in Part 100 of the Commission's regulations. The most severe accident analyzed and designed against 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 under extremely improbable conditions, and the calculated exposures remain within the guidelines of 10 CFR Part 100. Analyses were also performed which included lesser accidents such as the steam generator tube rupture, the secondary system steam line break, a failure in the gaseous waste disposal system, rupture of the volume control

37. Applicant has demonstrated that the reactor vessel for Unit No. 2 has been designed in accordance with the applicable general design criteria for this plant and the requirements of 10 CFR 50.55a.[48/]

38. The reactor vessel is designed in accordance with the ASME Code and the Westinghouse equipment specification. As a consequence, because of the extensive and technically sound requirements imposed on the design of the vessel in accordance with the Code, because the significance of these requirements was known and understood so that they could be implemented properly in the design, and because evidence of compliance with these requirements was obtained, there is assurance that the Indian Point Unit No. 2 reactor vessel will not fail by overstress, creep rupture, or in fatigue.[49/]

39. The reactor vessel is in compliance with ASME Code and equipment specification material, fabrication, and inspection requirements. In many cases, the

---

48/ FSAR, Section 4.1; Reactor Vessel Integrity Testimony, Sections 1-7 (follows Tr. 1932).

49/ Reactor Vessel Integrity Testimony, Section 2.0, App. A and D (follows Tr. 1932); Staff Report on Pressure Vessel, pp. 4-14 (introduced into evidence Tr. 2715); FSAR, Sections 4.2.2, 4.3.1, 4.4.1, 4.4.3, App. B, Questions 4.6, 4.8.1, 4.8.2, 4.8.4, 4.10; Tr. 2032-33, 2049-2050, 3944-3946, 3952, 3966-69.

equipment specification requirements are more stringent than the requirements of the ASME Code.50/ By virtue of its compliance with the Code and the equipment specification, the Unit No. 2 reactor vessel was, therefore, fabricated with materials and by techniques, and inspected, in accordance with extensive and technically sound requirements. Evidence of compliance with requirements was obtained. Thus, there is assurance that the materials employed are well known and there is extensive experience in their use; they have the properties assumed by the designer; they are free of injurious defects; and good workmanship was employed and fabrication was properly carried out. Hence, there is assurance that the Unit No. 2 reactor vessel will not fail because of material or fabrication deficiencies.51/

40. The reactor vessel will be subject to operation in accordance with the technical specifications for Unit No. 2. The technical specifications provide adequately conservative

---

50/ Reactor Vessel Integrity Testimony, Section 3.0 (follows Tr. 1932); ASLB--March 24, 1971, Part I, Question 2, pp. 4-6 (follows Tr. 728).

51/ Reactor Vessel Integrity Testimony, Section 3.0, App. A and D (follows Tr. 1932); Staff Report on Pressure Vessel, pp. 14-24 (introduced into evidence Tr. 2715); FSAR, Section 4.2.5, App. B, Questions 4.8.6, 4.8.7; Tr. 2035, 2036, 2039-46, 3934-40, 3943-44, 3948-50, 3955-56, 3963-66, 3970-71, 3976, 3989-91.

operating limits on reactor coolant system temperature, pressure, heatup and cooldown rates, and chemical environment, and specify equipment availability, and operational procedures. As a consequence, operation in accordance with the technical specifications provides assurance that the reactor vessel will not fail due to brittle failure, ductile yielding, or any of the postulated operational transients including accident conditions, and the integrity of the reactor vessel will not deteriorate in the environment in which it is to operate.52/

41. Applicant has presented detailed evaluations of safety margins using the latest methods of failure analysis which demonstrate that the Unit No. 2 reactor vessel will not fail by brittle failure.53/

52/ Summary of Application, pp. 13, 57-59; Reactor Vessel Integrity Testimony, Sections 4.0, 5.0, App. B and C (follows Tr. 1932); Staff Report on Pressure Vessel, pp. 24-32 (introduced into evidence Tr. 2715); FSAR, Sections 4.2.2, 4.2.3, 4.2.6, 4.2.8, 4.5, App. 4A, Question 4.8.5; Tech. Specs. No. 2.1, 2.2, 2.3, 3.1A, B, C, E, F. 3.2, 3.3, 3.5, 3.7, 3.10, 4.1, 4.2, 4.3, 4.5, 4.6, 6.3, 6.7; Tr. 2037-2039, 2046-56, 3940-43, 3948-50, 3953-63, 3969-76.

53/ Reactor Vessel Integrity Testimony, Section 5.0 (follows Tr. 1932); Staff Report on Reactor Vessel, pp. 24-28 (introduced into evidence Tr. 2715).

42. Applicant has further demonstrated that the Board's questions relating to the pressure vessel which included design, fabrication and inspection techniques, methods of primary system leakage detection and studies of the control rod ejection accident have been satisfactorily investigated and resolved.54/

43. The Citizens Committee also contended that two other accidents should be considered in the design of the plant: a major meltdown of the core following a loss-of-coolant accident and the crash of an airplane into the reactor building.

44. Core meltdown can only be postulated in the event of a major failure of the primary coolant system and subsequent failure of the emergency core cooling system to perform adequately. Applicant has shown that the emergency core cooling system will limit the cladding temperature

---

54/ ASLB--Jan. 19, 1971, Part I, Questions 2, 3 (follows Tr. 665); ASLB--Jan. 19, 1971, Part II, Question 16, pp. 13-15 (follows Tr. 665); ASLB--Jan. 19, 1971--Staff, pp. 4-6 (follows Tr. 728); ASLB--May 13, 1971, Questions 9, 10 (follows Tr. 890); Reactor Vessel Integrity Testimony, Sections 3, 5, App. A, C, D (follows Tr. 1932).

to below the melting temperature of zircaloy - 4 and below the temperature at which gross core geometry distortion including clad fragmentation may be expected.[55/] For this reason major meltdown of the core is not a postulated accident against which Unit No. 2 has been designed.[56/] Applicant determined and the ACRS and the Regulatory Staff agreed that a crucible which had previously been proposed to protect against the possible consequence of a core meltdown need not be included in the final design of the Plant. The Staff further testified that it did not consider that core meltdown need be considered in the plant's design.[57/] On the basis of its findings with respect to the reactor vessel integrity and the performance of the emergency core cooling system, together with its other findings herein, the Board concludes that this Plant need not be designed against the meltdown of the reactor core.

55/ See Paragraphs 71-74.

56/ ASLB--Jan. 19, 1971, Part I, Questions 10, 13 (follows Tr. 665); ASLB--March 24, Part II, Question 3 (follows Tr. 888); FSAR, Section 1.5.

57/ SSE, pp. 40, 90 (follows Tr. 405); Tr. 1148-49, 4025-26, 4029; FSAR, p. 1.1-3 to 1.1-4.

45. With respect to the hypothetical airplane crash, the Board concludes that the probability of a plane striking the facility is so remote that this contingency need not be considered in the design of Indian Point Unit No. 2.[58/]

58/ Additional Testimony of Applicant, Part I, dated July 6, 1971, Response to Item 6.C (follows Tr. 892); Set H Responses, Part II, Question H-29 (CCPE Exhibit H).

IX. Engineered Safeguards

46. The hydrogen control system provided by Applicant will maintain the measured hydrogen concentration in the containment atmosphere below 2 percent by volume thereby precluding hydrogen ignition as an energy source subsequent to a loss-of-coolant accident ("LOCA"), and assuring that containment integrity will be maintained.[59/]

47. The redundant hydrogen recombiner units installed within the containment have been designed, fabricated, and tested in such manner to demonstrate that they will perform their intended function.[60/]

48. The containment air recirculation cooling and filtration system will recirculate and cool the containment atmosphere in the event of a LOCA and thereby assures that the containment pressure and temperature will not exceed the design value of 47 psig and 271°F

59/ Summary of Application, page 19; SSE, pp. 44-45 (follows Tr. 405); FSAR, Question 6.8(a).

60/ SSE, pp. 44-45 (follows Tr. 405); ASLB--Jan. 19, 1971, Part II, Question 16, pp. 21-22 (follows Tr. 665); Set H Responses, Part I, Question H-21 (CCPE Exhibit H); FSAR, Questions 6.8, 6.9, 6.10, 6.11, 14.8, 14.9, 14.19.

(100% relative humidity) respectively.[61]

49. Assuming the containment design leak rate, the filtration capacity of the containment air recirculation cooling and filtration system will when operated in combination with the containment spray system reduce the concentration of particulate and organic iodine in the containment atmosphere following a LOCA to levels which assure that the guidelines of 10 CFR Part 100 are met.[62]

50. Applicant and Staff have presented evidence of analytical and experimental work which fully supports the conclusion of the adequacy of the iodine removal capabilities of the containment air recirculation cooling and filtration system.

61/ Summary of Application, p. 18, SSE, pp. 41-43 (follows Tr. 405); FSAR, Sections 6.4, 14.3.4 to 14.3.10, 14.3.23, Questions 6.6, 7.8, 7.9; Tech. Specs. No. 3.3.B, 4.1, 4.5.C.

62/ SSE, pp. 41-43, 60-62; FSAR, Sections 6.4, pp. 14.3.5-3 to 14.3.5-4, Questions 14.1, 14.10; Tech. Specs. No. 3.3.B, 4.1, 4.5.C.

(a) This testimony demonstrates that the equipment associated with this system will work as required in the post LOCA environment.63/

(b) The charcoal used will have adequate iodine removal capability even under 100 percent humidity conditions to satisfy 10 CFR Part 100.64/

51. The containment spray system will spray cool alkaline sodium borate solution into the containment atmosphere in the event of a LOCA and thereby assures that the containment pressure will not exceed the design value of 47 psig.65/

52. The containment spray system will remove elemental iodine from the containment atmosphere should it be released in the event of a LOCA and thereby assures

---

63/ ASLB--Jan. 19, 1971, Part II, Question 16, pp. 18-20, 23-28 (follows Tr. 665); FSAR, Questions 6.6, 7.8, 7.9; Tr. 1300-12.

64/ FSAR, Question 14.10; Tr. 1300-1312.

65/ Summary of Application, page 18; SSE, pp. 41-43 (follows Tr. 405); FSAR, Sections 6.3, pp. 14.3.4-10 to 14.3.4-13, Questions 6.5, 7.8, 7.9; Tech. Specs. No. 3.3.B, 3.5, 4.1, 4.5.B; Tr. 1101.

that the off-site thyroid dose will be within 10 CFR Part 100.[66/]

53. Applicant and Staff have presented evidence of analytical and experimental work which fully supports the conclusion of the adequacy of the iodine removal capabilities of the containment spray system. This testimony demonstrates that the use of the single drop model is conservative and that the mechanisms associated with spray removal are adequately understood.[67/]

(a) The average drop diameter assumed for the application of the single drop model is conservative when considering the effects of drop size distribution, coalescence, nozzle pressure and condensation.[68/]

66/ SSE, pp. 41-43, 60-62 (follows Tr. 405); FSAR, Section 6.3, page 14.3.5-3, App. 6A, Question 14.1; Tech. Specs. No. 3.3.B, 3.5, 4.1, 4.5.B.

67/ FSAR, Question 6.2, App. 6A; Tr. 1315, 1322.

68/ ASLB--Jan. 19, 1971, Part II, Question 16, pp. 29-31 (follows Tr. 665); ASLB--May 13, 1971, Question 3 (follows Tr. 890); FSAR, Questions 6.2, App. 6A; Tr. 1314-1318, 1325-1327, 1329-1333, 1377, 1478-1482, 1485-1486, 1490-1492, 1494-1498, 1503-1510, 1513, 1523-1525, 1527, 1528.

(b) The model used to predict the elemental iodine removal coefficient is supported by substantial experimental evidence.[69/]

(c) The assumption of 95 feet as the lower limit of spray fall height is conservative in view of considerably longer fall heights in some regions of the containment such as the refueling pit and the region between the crane support wall and the containment liner.[70/]

(d) Effective mixing within the containment is provided by the ventilation system as well as by the vertical temperature gradient within containment as a result of spray cooling action.[71/]

(e) The assumed drop fall height (121.1 ft.) is conservative and results in a shorter drop residence time (11.9 sec.) than would actually be expected since

---

69/ FSAR, Question 6.2, App. 6A; Tr. 1314-1318, 1322.

70/ FSAR, Question 6.2, App. 6A; Tr. 1314, 1335, 1346-1347 1487-1490.

71/ ASLB--May 13, 1971, Question 1 (follows Tr. 890); FSAR, Question 6.2, App. 6A; Tr. 1488-1489, 1499-1504, 1510-1513, 1516, 1534, 1549-1550, 1598-1600.

the drops leave the nozzle with a horizontal as well as the assumed vertical velocity component.72/

(f) A value of the iodine removal constant of 32.6 hrs.$^{-1}$ based on the conservative model summarized above was used in Applicant's accident analysis. Compliance with 10 CFR Part 100 is demonstrated by Applicant's analysis as well as by the Regulatory Staff's analysis utilizing an even more conservative value (4.5 hrs.$^{-1}$).73/

(g) The spray additive, sodium hydroxide, has been demonstrated to be adequate in the removal of iodine from the containment atmosphere following the LOCA.74/

54. The isolation valve seal water system provides additional assurance of the effectiveness of those containment isolation valves which are located in lines connected to the reactor coolant system or which could be exposed to the containment atmosphere. This is accomplished by providing a

---

72/ FSAR, Question 6.2, App. 6A; Tr. 1335, 1347, 1487-1490.

73/ SSE, pp. 61-63 (follows Tr. 405); FSAR, Sections 6.3, 14.3.5, App. 6A, Questions 6.2, 14.1; Tr. 1317-1318, 1337-1346, 1557-1561, 1564-1571.

74/ ASLB--Jan. 19, 1971, Part II, Question 16, pp. 29-31 (follows Tr. 665); ASLB--March 24, 1971, Questions 6, 10 (Tr. 730-745); ASLB--May 13, 1971, Questions 4, 5 (follows Tr. 890); FSAR, Questions 6.2, 6.5, 14.1, App. 6A, page 14.3.5-3, Tr. 730-733, 735-745, 1322-1333, 1350-1351, 1518-1525, 1539-1544.

water seal and in a few cases a gas seal at the valves.[75/] Although no credit is taken for operation of this system in the calculation of off-site accident doses, it does provide assurance that the containment leak rate is lower than that assumed in the accident analysis.[76/]

55. The containment penetration and weld channel pressurization system provides means for continuously pressurizing the positive pressure zones incorporated into the containment penetrations and the channels over the welds in the steel inner liner in the event of a loss-of-coolant accident.[77/] No credit is taken for the operation of this system in calculating off-site accident

---

75/ Summary of Application, pp. 15-16; SSE, p. 44 (follows Tr. 405); FSAR, Section 6.5; Tech. Spec. No. 3.3.C.

76/ SSE, pp. 44, 62 (follows Tr. 405); FSAR, Sections 6.5, 14.3.5, pp. 14.3.5-2, 14.3.5-13 to 14.3.5-14, 14.3.5-18.

77/ Summary of Application, page 16, SSE, page 43 (follows Tr. 405); FSAR, Section 6.6, Tech. Specs. No. 3.3.D, 4.4.IV.

doses, but it is designed as an engineered safety feature and, as the isolation valve seal water system, does provide assurance that the containment leak rate would be lower than that assumed in the accident analysis.[78/]

56. On June 29, 1971, the Commission published in the Federal Register (36 Fed. Reg. 12,247-50) Interim Acceptance Criteria for Emergency Core Cooling Systems for Light Water Power Reactors (sometimes hereinafter referred to as the "Criteria"). The Criteria establish the performance requirements which the emergency core cooling system for Unit No. 2 must meet (Section IV.A.) and further establish the evaluation model which the Applicant should follow in analyzing the performance of this system (Appendix A, Part 3). Subsequent to the issuance of the Criteria, Applicant undertook additional extensive analysis which is reflected in the evidence in this proceeding and which

---

78/ SSE, page 62 (follows Tr. 405); FSAR, Sections 6.6, 14.3.5 pp. 14.3.5-2, 14.3.5-13 to 14.3.5-14, 14.3.5-18.

demonstrates that the emergency core cooling system for Unit No. 2 ("ECCS") satisfies the Criteria. The Staff agrees with this conclusion.[79/] In demonstrating this, Applicant has provided adequate assurance of the effectiveness of its system for purposes of the Board's Initial Decision in this proceeding. However, because the intervenor Citizens Committee has attempted to challenge the validity of the Criteria the Board's findings herein deal with the effectiveness of the ECCS for Unit No. 2 in terms more general than are necessary to determine simply whether the Criteria have been satisfied.

57. Applicant has provided an ECCS which will deliver cooling water to the reactor core in the event of a LOCA in order to assure that the core will remain intact and in place, with its essential heat transfer geometry preserved. The ECCS consists of accumulators, safety injection (high head) pumps, residual heat removal (low head) pumps and associated equipment. The components of the emergency core cooling system operate in three modes: passive accumulator injection, active safety injection, and residual heat removal recirculation.[80/]

79/ Staff Safety Evaluation, Supplement No. 3 ("SSE, Supp. 3"), pp. 13-14 (introduced into evidence Tr. 2715).

80/ FSAR, Section 6.2.

58. A LOCA may result from a rupture of the reactor coolant system piping or of any line connected to that system up to the first closed valve.[81] The reactor core transients resulting from a LOCA have been analyzed for a spectrum of breaks up to and including the double-ended rupture of the largest pipe in the reactor coolant system.[82] For ruptures of a very small cross-section, the charging pumps will maintain an operating water level in the pressurizer permitting an orderly shutdown without need for the emergency core cooling system.[83]

59. A break too large for the charging pumps alone would result in loss of pressurizer liquid level and pressure which would in turn cause reactor trip and initiation of the emergency core cooling system.[84] Reactor trip and emergency core cooling would also be initiated by high containment

---

81/ FSAR, page 14.3.1-1.

82/ FSAR, Section 14.3, Additional Testimony of Applicant Concerning Emergency Core Cooling System Performance dated July 13, 1971 ("Additional ECCS Testimony"), pp. 1-72 (follows Tr. 1931).

83/ FSAR, pp. 14.3.1-1, 14.3.1-25 to 14.3.1-26; Additional ECCS Testimony, page 1 (follows Tr. 1931).

84/ FSAR, pp. 6.2-6, 7.2-1 to 7.2-3, 7.2-27, 14.3.1-7.

pressure.[85] The LOCA has been analyzed to show that timely initiation of emergency core cooling occurs for all break sizes.[86] For breaks up to and including those 4" in diameter, the water level in the reactor coolant system would drop but the ECCS would replace water fast enough that the core hot spot would never be uncovered.[87] For breaks larger than 4" in diameter the ECCS would function to reflood the core.[88]

60. Postulated breaks up to 6" in diameter were analyzed using the SLAP computer code.[89] The results of this conservative analysis demonstrated that the ECCS, assuming only partial effectiveness, limited the core hot spot cladding temperature to 1550°F for the 6" diameter break and to lower temperatures for smaller breaks.[90]

85/ FSAR, page 6.2-6.

86/ FSAR, page 14.3.1-28.

87/ FSAR, pp. 14.3.1-26 to 14.3.1-27.

88/ FSAR, pp. 6.2-6 to 6.2-11, 14.3.1-27.

89/ FSAR, pp. 14.3.1-11 to 14.3.1-12.

90/ FSAR, pp. 14.3.1-25 to 14.3.1-28.

61. Analyses of a spectrum of breaks larger than .5 square feet were performed in accordance with the Commission's Criteria.[91/] The model used was as specified in Appendix A, Part 3 of these Criteria.[92/] The model utilizes the computer codes SATAN-V and LOCTA-R2, as well as a reflood analysis.[93/]

62. The SATAN-V code contains appropriate simulations of blowdown hydraulics and reactor kinetics.[94/] The ability of SATAN-V to predict the hydraulic behavior of the reactor coolant during blowdown has been verified by comparing the results of the code with the experimental results at LOFT semiscale, Illinois Institute of Technology and Containment Systems Experiment (CSE).[95/]

91/ SSE, Supp. 3, page 7 (introduced into evidence Tr. 2715); Additional ECCS Testimony, page i (follows Tr. 1931).

92/ 36 F.R. 12,249; Additional ECCS Testimony, pp. i-iii (follows Tr. 1931).

93/ Additional ECCS Testimony, pp. 4-17 (follows Tr. 1931).

94/ Additional ECCS Testimony, pp. 4-9, 18 (follows Tr. 1931); FSAR, pp. 14.3.1-5 to 14.3.1-7, App. 14B.

95/ Additional ECCS Testimony, page 5 (follows Tr. 1931); ASLB--Jan. 19, 1971, Part II, Question 16, page 5 (follows Tr. 665); FSAR, App. 14B, Tr. 3355-3356, 2850-2858.

63. The effects of flow redistribution (radial flow) during blowdown were conservatively accounted for by reducing the core flows calculated by SATAN-V by twenty percent although the calculated flow redistribution is only ten to fifteen percent.96/ The ten to fifteen percent flow redistribution during blowdown was calculated using the THINC code which has been verified experimentally.97/

64. Following the termination of the SATAN-V blowdown transient calculation, an appropriate reflood and heat transfer analysis was performed incorporating a conservative calculation of flooding rate and application of the FLECHT heat transfer data.98/

65. The cladding temperature and zircaloy-water reaction throughout the accident, during both blowdown and reflood, were calculated using the LOCTA-R2 computer code.99/ To calculate the extent of zircaloy-water reaction, LOCTA-R2 uses the parabolic rate equation derived from experimental

---

96/ Tr. 2799-2800, 2840-42.

97/ Tr. 2129-30, 2740, 2777-79, 2801.

98/ Additional ECCS Testimony, pp. 9-14 (follows Tr. 1931); Tr. 3054-3055, 3134-35, 3139-42, 3296, 4110-14.

99/ Additional ECCS Testimony pp. 4, 14-17 (follows Tr. 1931); FSAR, pp. 14.3.1-7 to 14.3.1-10.

data taking no credit for steam limited reaction rate.[100/] Calculations using the parabolic rate equation have been performed and compared to experimental results. These comparisons have demonstrated that the calculation conservatively overpredicts the extent of zircaloy-water reaction.[101/]

66. In addition to zircaloy-water reaction, other potential chemical reactions such as zircaloy-inconel eutectic formation, UO_2-steam reaction, zircaloy-hydrogen reaction, and hydrogen explosion have been investigated and adequately taken into account.[102/]

67. Both the blowdown and the reflood calculations were made assuming no distortion of the core. The effects of fuel rod distortion resulting from a LOCA were adequately considered, both in extensive experimental and related analytical programs.[103/] These programs included:

100/ Additional ECCS Testimony, page 15 (follows Tr. 1931); FSAR, page 14.3.1-8; Tr. 2109, 2133-34, 2305, 2542, 2637.

101/ Tr. 2297-99, 2310, 2314, 2317-18, 2591-92, 2599, 2620-21, 2637.

102/ Additional Testimony of Applicant, Part I, dated July 6, 1971, Response to Items 3.a.3g and 3.a.3(i) (follows Tr. 892); Tr. 2109, 2110, 2170-71, 2458-59, 2383, 2469, 2472, 2557-59, 2572, 3868-73.

103/ Additional ECCS Testimony, page 3 (follows Tr. 1931); ASLB--Jan. 19, 1971, Part II, Question 16, pp. 10-12 (follows Tr. 665); Tr. 13-14 (Nov. 10, 1971, *in camera*).

a. The single rod burst tests which provided appropriate scoping data for the multirod burst tests.[104] These tests also demonstrated that no observable difference existed between irradiated and non-irradiated cladding.[105]

b. Multirod burst tests which determined that fuel rods burst in randomly distributed locations and that extensive flow blockage did not occur.[106]

c. Transient heat transfer tests (FLECHT) which determined that flow blockage greater than the maximum blockage observed in the multirod burst tests did not adversely effect core cooling heat transfer and that, in fact, the heat transfer was enhanced.[107]

d. Analyses using limiting geometries from the multirod burst program, which determined that

---

104/ Tr. 11-12 (Nov. 10, 1971, in camera); Tr. 45 (Nov. 8, 1971, in camera).

105/ Tr. 13, 75 (Nov. 10, 1971, in camera); Tr. 2115, 2149-2150.

106/ ASLB--Jan. 19, 1971, Part II, Question 16, page 11 (follows Tr. 665); Tr. 2111-2114, 2125-27.

107/ ASLB--Jan. 19, 1971, Part II, Question 16, pp. 3-4 (follows Tr. 665); Additional ECCS Testimony, page 30 (follows Tr. 1931); Tr. 2119, 2128, 2237, 2240-41, 3044-45, 4114-15.

peak clad temperatures calculated during a LOCA increase 70° F due to geometric distortion. These analyses were performed with no credit taken for the beneficial effects of flow blockage on heat transfer which were observed in the FLECHT tests.[108/]

68. Section IV.A.1. of the Criteria requires that the calculated maximum fuel element cladding temperature not exceed 2300°F. Applicant's analyses demonstrate that this Criterion is met by showing that the ECCS with partial effectiveness limits maximum cladding temperature to 2300°F for the largest break and lower maximum cladding temperatures for smaller breaks.[109/]

69. Applicant has demonstrated that for operation of Unit No. 2 at fifty percent of full power (1379 Mwt) the maximum fuel element cladding temperature for the largest break would be less than 1200°F and that no swelling or bursting of fuel rods would occur.[110/]

108/ Additional ECCS Testimony, page 3 (follows Tr. 1931); FSAR, page 14B-11; Tr. 2117-19, 2138-39, 2148-49, 2157-62, 2733-39, 3050, 3054-60, 3846, 3884; Tr. 13-14, 55-59 (Nov. 10, 1971, in camera).

109/ Additional ECCS Testimony, page 20 (follows Tr. 1931).

110/ Tr. 2125, 4033, 4166. In response to an inquiry by the Board (see Tr. 4163) Applicant introduced into evidence a graph illustrating that in fact the peak clad temperature subsequent to the double-ended break at 1400 Mwt thermal assuming infinite irradiation is 1140°F (introduced into evidence Tr. 4166).

70. Analyses performed demonstrate that Applicant's ECCS meets Section IV.A.2. of the Criteria. This Criterion requires that the amount of fuel element cladding that reacts chemically with water or steam not exceed one percent of the total amount of cladding in the reactor. The core wide extent of zircaloy-water reaction for the largest break was determined to be less than .07 percent of the total amount of cladding in the reactor.[111] Potential chemical reactions other than zircaloy-water reaction are not significant.[112]

71. Section IV.A.3. of the Criteria requires that the clad temperature transient be terminated at a time when the core geometry is still amenable to cooling and before the cladding is so embrittled as to fail during and after quenching. Applicant has demonstrated that its ECCS meets this Criterion. The potential effects of clad shattering, clad swelling and resulting flow blockage, and blowdown forces have been analyzed and shown to be limited so that the core geometry is still amenable to cooling.[113]

111/ Tr. 2104, 2278.

112/ Tr. 2109-10.

113/ See Paragraphs 72-74 and references cited therein.

72. Quench tests, conducted as part of the single rod burst test program, demonstrate that cladding at 2700°F with ten percent local metal water reaction or at 2300°F with sixteen percent local metal water reaction will retain its integrity when subjected to more severe thermal transients than would occur during an accident.[114/] Applicant calculated that its ECCS would limit conditions for the hottest spot in the core to 2300°F with 7.5 percent local metal water reaction.[115/] Since this is less than the limits determined by the quench tests, no clad shattering would be expected. The condition of 2300°F and 7.5 percent local metal water reaction applies only to a very small fraction of the core. Most of the core is much cooler and is therefore subjected to essentially no metal water reaction.[116/]

73. Applicant has shown that clad swelling and resulting flow blockage will not result in the core being uncoolable.[117/]

114/ Additional ECCS Testimony, page 3 (follows Tr. 1931); Tr. 2115-17, 2364, 2369, 2395; Tr. 24-25 (Nov. 8, 1971, *in camera*).

115/ Tr. 1464 Additional ECCS Testimony, pp. 20-21 (follows Tr. 1931).

116/ Tr. 2277-79.

117/ Additional ECCS Testimony, pp. 3, 30 (follows Tr. 1931); ASLB--Jan. 19, 1971, Part II, Question 16, pp. 3-4, 10-12 (follows Tr. 665); FSAR, page 14B-11; Tr. 2111-15, 2117-19, 2125-28, 2138-39, 2148-50, 2237, 2240-41, 2157-62, 2733-39, 3044-45, 3054-60, 3846, 3884, 4114-15; Tr. 45 (Nov. 8, 1971, *in camera*), Tr. 11-12, 13-14, 55-59, 75 (Nov. 10, 1971, *in camera*). See discussion in Paragraph 67.

74. The reactor core and internals together with the ECCS are designed so that in the event of a LOCA the reactor can be safely shut down and essential heat transfer geometry of the core can be preserved. An analysis has been made of the effects of blowdown forces on the core and internals, and these forces have been shown to result in no alterations of core geometry.[118/] The BLODWN code which is used for these calculations has been adequately confirmed with experimental results.[119/]

75. Section IV.A.4. of the Criteria requires that the core temperature be reduced and decay heat be removed for an extended period of time, as required by the long-lived radioactivity remaining in the core. Applicant's emergency core cooling system is designed to continue removing decay heat from the core for an extended period of time[120/] and therefore, Applicant has demonstrated compliance with this Criterion.

76. The Board takes note that in Applicant's analyses of the ECCS for Unit No. 2 in accordance with the Interim Criteria, Applicant has demonstrated not only that

---

118/ FSAR, Section 14.3.3.

119/ Tr. 2747, 2750, 2752.

120/ FSAR, pp. 6.2-8 to 6.2-11.

the system will limit core damage to a level below that which might result in the core becoming uncoolable but also that the Criteria themselves (i.e., Appendix A, Part 3) contain a number of conservative assumptions which give additional assurance of the effectiveness of the system and expanded confidence that the public health and safety will be protected. These conservatisms result in the calculated peak clad temperature of 2300°F, referred to in Paragraph 68. More realistic assumptions would result in a calculated peak clad temperature of 1500°F.[121/]

The following assumptions used in the analysis were conservative:

(1) Departure from nucleate boiling was assumed in 0.1 seconds followed by transition boiling heat transfer lower than that expected.

(2) All accumulator water injected during blowdown was assumed to bypass the core and not contribute to reflood of the core.

(3) Break discharge coefficient larger than expected was assumed for all break sizes.

121/ Tr. 3879-80.

(4) Reactor coolant pumps were assumed to trip at the time of the break and develop head during cavitation, thus minimizing core flow and core heat transfer during blowdown.

(5) From the end of blowdown until core reflood the core was assumed to be adiabatic with no credit for heat transfer to steam.

(6) Decay heat twenty percent greater than expected was assumed during blowdown and reflood.

(7) The intact reactor coolant piping was assumed to be plugged and unavailable for steam venting during the accumulator injection period.

(8) For purposes of computing the amount of entrained water during reflood it was assumed that the entire core was composed of hot assemblies, thus maximizing loop pressure drop.

(9) Conservative loop resistances were assumed including locked rotor resistance for reactor coolant pumps.

(10) FLECHT test results used for the reflood heat transfer did not include beneficial effects of mixing vane grids.

(11) Effects of flow blockage on peak clad temperature were conservatively calculated as a 70°F increase.[122/]

77. Further, the Board notes that the analyses of the system in accordance with the Criteria have been performed assuming loss of all off-site power and failure of one of the three high head safety injection pumps and one of the two low head pumps.[123/] Therefore the system will perform its function even in degraded conditions.

78. Applicant has presented evidence that for the operation of Unit No. 2 at fifty percent of full power the high head safety injection system is not needed for large breaks. The operation of the low head safety injection system could be delayed approximately five minutes without temperatures exceeding 2300°F.[124/]

122/ Additional ECCS Testimony, pp. 8, 18 (follows Tr. 1931); Tr. 3879-85.

123/ FSAR, pp. 6.2-2, 6.2-7.

124/ Responses of Applicant to Questions of the ASLB at the November 17, 1971 Hearing Session, Item 3, page 2 (follows Tr. 4162).

79. On the basis of its findings numbered 57 through 78 the Board concludes that the Applicant has adequately considered and complied with all elements of the Criteria and that the ECCS for Unit No. 2 will function in such a manner that the public health and safety will not be endangered in the event of a LOCA.[125/]

125/ SSE, Supp. 3, pp. 13-14 (introduced into evidence Tr. 2715).

X. Industrial Security

80. The Atomic Energy Commission has stated that protection against possible industrial sabotage is a matter to be dealt with at the operating license stage and that, at that stage, the Commission would expect assurance that an applicant would provide for "appropriate industrial security measures."[126/] The Board has considered the evidence in this proceeding in light of the Commission's directive and related guidance provided by the Atomic Safety and Licensing Appeal Board.[127/]

81. As contained within the record of this

126/ Fla. Power & Light Co., 3 A.E.C. 173 (1967).

127/ In its Memorandum of May 26, 1970, in Trustees of Columbia Univ., Docket No. 50-208, the Atomic Safety and Licensing Appeal Board noted the Fla. Power & Light decision of the Commission and stated that in its view "the matter of possible civil disturbance is also within the ambit of the foregoing approach.... Thus, as respects the possibility of industrial sabotage or civil disturbance, it will properly be the role of the Board to determine, on the basis of the record, whether the applicant's proposed industrial security measures for this particular facility are adequate. [footnote omitted] In evaluating the adequacy of these security measures, their effectiveness in preventing any credible hazards to the public should be examined, as should be the inherent and engineered safety characteristics of the facility which bear on the matter." At 10-11.

proceeding[128/] the Applicant has shown that it has provided "appropriate industrial security measures" for this facility.

82. The Security Plan describes the administrative and design features to guard against and detect unauthorized access to various portions of the plant. The plan provides for a minimum of a double physical barrier against entry into the plant.[129/] Applicant will provide adequate surveillance of these barriers, including use of a guard patrol, an electronic detection system and perimeter fence lighting.[130/] Locks

---

128/ The Board convened in camera hearing sessions (Tr. 1281) on plant security measures on July 15, July 21, November 12, December 14, 1971 and January 11, 1972. Applicant testified that disclosure of the details of the security measures would be inimical to the public health and safety in that it would result in a diminution of plant security. (Tr. 4-5, 8-9, July 15, 1971, in camera). At the November 12, 1971 hearing session the Indian Point Security Plan ("Security Plan") and Additional In Camera Testimony of Applicant Concerning Station Security ("Applicant's Additional In Camera Testimony") were received into evidence at transcript pages 6 and 8 respectively.

129/ November 12, 1971, in camera, Tr. 9-16; Security Plan Sections 2.0, 2.2, 2.4 (follows Tr. 6, November 12, 1971, in camera); SSE, pp. 68-69 (follows Tr. 405).

130/ Tr. 18-19, 21, 41-44 (Nov. 12, 1971, in camera); Security Plan, Sections 2.1, 2.3, 2.8, 3.0, 3.1; Applicant's Additional In Camera Testimony, page 1 (follows Tr. 8, Nov. 12, 1971); Additional Testimony of Applicant, Part 1, dated July 6, 1971, Response to Item 6.b. (follows Tr. 892); Set H Responses, Question H-30 (CCPE Exhibit H); FSAR, Questions 12.6, 12.7.

and protective devices are provided.[131/] Areas of particular safety importance have been provided with additional security features.[132/] Appropriate procedures have been developed for monitoring of security areas.[133/] Adequate and detailed procedures for the monitoring and the controlling of access to and from the plant and movement within the plant have been developed.[134/] Periodic monitoring by plant personnel and continuous operational surveillance by mechanical means are provided for equipment and facility structures.[134a/] Procedures for dealing with particular security hazards such as bomb threats, unauthorized entries and door monitoring system activation have been developed.[135/] Discussions with appropriate

---

131/ Tr. 23-28 (Nov. 12, 1971, in camera); Security Plan, Sections 2.4, 2.6, 2.7 (follows Tr. 6, Nov. 12, 1971, in camera).

132/ Tr. 1707-09 (July 21, 1971, in camera); Tr. 26-27 (Nov. 12, 1971, in camera); Security Plan, Sections 2.5, 2.6 (follows Tr. 6, Nov. 12, 1971, in camera); FSAR, Question 12.6.

133/ Security Plan, Section 3.2 (follows Tr. 6, Nov. 12, 1971, in camera).

134/ Tr. 29-31 (Nov. 12, 1971, in camera); Security Plan, Sections 3.0, 3.1, 3.2, 4.1 to 4.6 (follows Tr. 6, Nov. 12, 1971, in camera); SSE, pp. 68-69 (follows Tr. 405); FSAR, Question 12.7.

134a/ Testimony of Applicant Concerning Personnel Selection and Instructions on Indian Point Plant Security, December 10, 1971 ("Applicant's Dec. 10, 1971 Personnel Selection Testimony"), pp. 7-8 (follows Tr. 4340).

135/ Security Plan, Sections 4.6, 4.9, 4.10 (follows Tr. 6, Nov. 12, 1971, in camera).

officials and police have been held and procedures for assuring adequate communications, timely notification, availability of sufficient law enforcement personnel, and prompt response times have been factored into the plan.[136/]

82a. Having reviewed the Applicant's Security Plan for Unit No. 2 the Staff has required that particular elements of that plan be implemented prior to initial criticality. The Staff has also testified that the Applicant must comply with a required schedule for the completion of the supplemental, more sophisticated portions of the plan prior to completion of power ascension testing.[136a/] The Staff has testified and this Board finds that the Security Plan as reviewed and approved by the Staff together with the schedule for completion provides the necessary protection

---

136/ Tr. 54-55 (July 15, 1971, in camera); Tr. 1687-88, 1697-98 (July 21, 1971, in camera); Security Plan, Sections 2.9, 4.7, 4.8 (follows Tr. 6, Nov. 12, 1971, in camera); FSAR, Question 7.15.

136a/ Supplemental Staff Testimony, Indian Point Hearing, pp. 1-2 (follows Tr. 6, Dec. 14, 1971, in camera); Staff Testimony in Response to Intervenor's Interrogatories Dated December 22, 1971, Indian Point - 2 Hearings ("Staff Responses to Interrogatories"), Response 6 (follows Tr. 29, Jan. 11, 1972, in camera); Tr. 21-22 (Dec. 14, 1971, in camera), Tr. 34 (Jan. 11, 1972, in camera).

for the health and safety of the public.[136b/]

83. The design and arrangement of the plant provides additional assurance that the public health and safety will not be impaired.[137/] The reactor protection system has been designed to shut down the plant automatically and put it in a safe shutdown condition if a safety limit is exceeded.[138/] This means that actions taken in the control room by an intruder would not cause the plant to go into an unsafe condition. In addition, the plant may be maintained in a safe condition by operation at local remote control panels.[139/]

83a. Applicant's program for selection and review of personnel utilizes a well-developed defense in depth

---

136b/ Supplemental Staff Testimony, Indian Point Hearing, page 2 (follows Tr. 6, Dec. 14, 1971, in camera); Staff Response to Interrogatories, Responses 1, 2, 5 (follows Tr. 29, Jan. 11, 1972, in camera); Tr. 13 (Dec. 14, 1971, in camera); Tr. 33, 36-41 (Jan. 11, 1972, in camera).

137/ Tr. 1676-78, 1691-92 (July 21, 1971, in camera); Set H Responses, Question H-31 (CCPE Exhibit H).

138/ Set H Responses, Part II, Question H-31 (CCPE Exhibit H); Additional Testimony of Applicant, dated July 8, 1971 (follows Tr. 894); FSAR, Section 7.2.

139/ FSAR, pp. 7.7-4 to 7.7-11.

concept including interviews, checks of background, pre-employment and annual medical examinations, and observations and periodic reports by trained supervisors.[139a/] In addition to these multiple administrative techniques the facility is designed to protect itself against accidental or deliberate acts or malfunctions causing a safety limit to be exceeded.[139b/] Applicant's program coupled with the design of the facility gives reasonable assurance that the facility is adequately protected against an unstable individual intent on causing harm to the facility.

84. The Citizens Committee for the Protection of the Environment has contended that plant security is inadequate. In support of this position the intervenor has postulated the theoretical hazard of ten men armed with weapons and possessing sophisticated knowledge relating to

---

139a/ Applicant's Dec. 10, 1971 Personnel Selection Testimony, pp. 1-6 (follows Tr. 4340); Fire and Personnel Interrogatories, Questions 22-24, 26-31 (CCPE Exhibit EE); Tr. 4586, Tr. 29-34 (Dec. 14, 1971, in camera), Tr. 6-12, 17-18, 21-23 (Jan. 11, 1971, in camera).

139b/ Tr. 30-31 (Dec. 14, 1971, in camera); FSAR, Section 7.2.

nuclear reactors. The intervenor has also postulated attacks on the plant with various weapons.[140/] The intervenor introduced no evidence in support of the probability of these theoretical occurrences. The Board concludes that the Applicant's security measures and the inherent safety characteristics of the facility are adequate to prevent credible hazards to the public. Moreover, the Commission's regulations do not require the Applicant to provide for design features or other measures for the specific purpose of protection against the effects of "...attacks and destructive acts, including sabotage, directed against the facility by an enemy of the United States ..."[141/] In summary, reasonable and prudent security measures coupled with inherent and engineered safety characteristics of the facility provide adequate protection against credible acts of possible industrial sabotage or civil disturbance which might endanger the health and safety of the public.

140/ Tr. 55 (July 15, 1971, in camera); Set H Responses, Part II, Question H-31 (CCPE Exhibit H).

141/ 10 CFR 50.13.

XI. Emergency Plans

85. In accordance with Appendix E to 10 CFR Part 50, Applicant has developed a "Radiation Contingency Plan."[142/] This plan describes the emergency organization and procedures to be effected in the unlikely event of an accident which might endanger plant personnel or the general public. In the event of an accident defined as a "Site Contingency," the plan provides for immediate notification of a number of off-site persons and organizations including representatives of the New York State Department of Health.[143/]

86. Applicant has held numerous meetings with representatives of State and Federal agencies concerning the off-site protective actions which might be required in the event of an accident.[144/] These discussions have

---

142/ ASLB--Jan. 19, 1971, Part I, Question 14 (follows Tr. 665); FSAR, Question 12.5.

143/ ASLB--Jan. 19, 1971, Part I, Question 14 (follows Tr. 665); ASLB--March 24, 1971, Part II, Question 9 (follows Tr. 888); Set H Responses, Questions 32, 35 (CCPE's Exhibit H); Radiation Contingency Plan ("Plan"), page 26.

144/ ASLB--Jan. 19, 1971, Part I, Question 14 (follows Tr. 665).

resulted in agreement as to the nature of the procedures to be followed and information to be provided by Applicant in the event of a "Site Contingency." Applicant has estimated that in the event of the hypothetical occurrence of a double-ended rupture of the largest primary coolant pipe and consequent release of radioactivity into the containment, Applicant could provide the requisite information to the State, including type of accident that has occurred, gross activity levels inside containment, wind speed, wind direction, and calculated thyroid doses downwind within one-half hour after the onset of an accident.[145/]

87. The State of New York has developed an emergency plan for major radiation accidents involving nuclear facilities, as well as specific operating procedures for the Indian Point Station.[146/] The plan provides an overall framework for providing emergency response to major radiation accidents by the New York State Department of Health, Department of Transportation, State Police and

---

145/ ASLB--Jan. 19, 1971, Part I, Question 14 (follows Tr. 665); ASLB--March 24, 1971, Part II, Question 5 (follows Tr. 888); Supplementary Testimony of Sherwood Davies dated July 7, 1971 ("Testimony of Mr. Davies"), pp. 2-3 (follows Tr. 1754); Tr. 1868-69.

146/ New York State Emergency Plan for Major Radiation Accidents Involving Nuclear Facilities ("State Plan"), State of New York's Exhibit No. 2 (introduced into evidence Tr. 1748); Specific Operating Procedures ("Procedures"), State of New York's Exhibit No. 5 (introduced into evidence Tr. 1797).

other State and local agencies in cooperation with the Applicant. The responsive actions include notification of appropriate officials, evaluation of the particular accident, determination of the appropriate protective measures (including notification of the public) and the carrying out of such measures.147/

88. In accordance with the State's plan, the State Department of Health, upon notification by Applicant that a "Site Contingency" exists, would determine the necessity for protective actions off-site and would direct the various actions required. The State Department of Transportation is responsible for coordinating the operation of the plan upon receipt of instructions from the Department of Health.148/

147/ ASLB--Jan. 19, 1971, Part I, Question 6 (follows Tr. 665); Testimony of Mr. Davies, pp. 11-12 (follows Tr. 1754); Supplementary Testimony of Edward H. L. Smith dated September 15, 1971 ("Testimony of Mr. Smith"), pp. 3-4 (follows Tr. 1996 - the document is incorporated into the uncorrected transcript at Tr. 1549); State Plan, pp. 4-5 (State of New York's Exhibit No. 2).

148/ ASLB--Jan. 19, 1971, Part I, Question 14 (follows Tr. 665); Testimony of Mr. Davies, pp. 2-10, 18-19 (follows Tr. 1754); Supplementary Testimony of Sherwood Davies dated September 15, 1971 ("Sept. Testimony of Mr. Davies") page 2 (follows Tr. 1996 - the document is incorporated into the uncorrected transcript at Tr. 1549); Testimony of Mr. Smith, pp. 1-3 (follows Tr. 1996); State Plan, pp. 3-4, B1-B2 (State of New York's Exhibit No. 2).

89. Under the State's plan, protective action to minimize radiation exposure shall be undertaken if the projected dose to the thyroid is likely to exceed 30 rem and may be undertaken for lower projected doses.[149/] The State's plan and the implementing procedures for the Indian Point Station encompass pre-planned protective measures for limiting doses to 30 rem to the thyroid or less for "base case" major accidents having off-site consequences of up to 10 percent of the theoretical consequences of the design basis loss-of-coolant accident.[150/] For accidents involving off-site doses greater than 10 percent of the theoretical consequences of the design basis accident, the State has not pre-planned its emergency response.[151/] However, the State has a general emergency

---

149/ Testimony of Mr. Davies, pp. 7-8a (follows Tr. 1754); State Plan, page 2 (State of New York's Exhibit No. 2); Tr. 1846-47, 3754.

150/ The theoretical consequences of the design basis loss-of-coolant accident for the Indian Point 2 facility result in a two-hour dose to the thyroid approaching 300 rem, using the very conservative assumptions applied in the Regulatory Staff's Safety Analysis. Testimony of Mr. Davies, pp. 7-8a (follows Tr. 1754); Sept. Testimony of Mr. Davies, page 2 (follows Tr. 1996); Procedures, Tab 1: Definitions (State of New York's Exhibit No. 5); Tr. 3729.

151/ Testimony of Mr. Davies, pp. 8a, 16 (follows Tr. 1754); Sept. Testimony of Mr. Davies, page 2 (follows Tr. 1996); Testimony of Mr. Smith, page 6 (follows Tr. 1996).

response capability and procedure which would be brought into operation under those circumstances.[152] The extent of pre-planned emergency actions by New York State is based upon the advice of the Commission's Regulatory Staff and the Staff has concurred that protective measures should be considered in appropriate circumstances when projected dose levels are in the range of 30 rem to the thyroid.[153]

90. Representatives of the State testified as to the nature and timing of emergency actions which would be taken in the event of a serious accident.[154] Procedures for the "base case" accident include direction to residents of the low population zone surrounding the facility to stay indoors.[155] Evacuation is not necessary to protect residents of the low population zone from receiving doses in excess of 30 rem to the thyroid in "base case" situations.[156]

152/ Testimony of Mr. Davies, pp. 8a, 16 (follows Tr. 1754); Sept. Testimony of Mr. Davies, page 2 (follows Tr. 1996); Testimony of Mr. Smith, pp. 6-8 (follows Tr. 1996).

153/ Testimony of Mr. Davies, pp. 8-8a (follows Tr. 1754).

154/ Testimony of Mr. Davies, pp. 8a-10, 11-12 (follows Tr. 1754); Testimony of Mr. Smith, pp. 1-2 (follows Tr. 1996).

155/ Testimony of Mr. Davies, page 4 (follows Tr. 1754); Sept. Testimony of Mr. Davies, page 1 (follows Tr. 1996).

156/ Sept. Testimony of Mr. Davies, page 1 (follows Tr. 1996).

State witnesses testified that, based upon experience in responding to other local emergencies, it is reasonable to assume that it would take from one to two hours after verification of an accident to carry out essential notification and commence preliminary response procedures in "base case" situations.[157/]

91. Evacuation of residents of the low population zone would be undertaken by the State in "base case" situations, but only if substantial dose savings could be effected thereby, and evacuation could be carried out without resulting in greater risk than if the residents remained indoors. Factors to be considered by the State in determining whether to order evacuation include wind direction, time of day, weather and traffic.[158/] It is estimated that if evacuation of the residents of the low population zone should prove to be necessary because of accidents exceeding the "base case," this could be accomplished within a relatively short period of time.[159/]

157/ Testimony of Mr. Smith, page 1 (follows Tr. 1996).

158/ Testimony of Mr. Davies, pp. 15-16 (follows Tr. 1754); Sept. Testimony of Mr. Davies, pp. 1-2 (follows Tr. 1996); Tr. 3754

159/ Testimony of Mr. Smith, pp. 5-6 (follows Tr. 1996); Tr. 3778-79.

Experience in other emergency situations supports this conclusion.[160/]

92. The State of New York has prepared its radiation emergency plan based upon the underlying theory that firm procedures can be implemented only upon a determination of all factors at the time of the accident.[161/] This is the basis of all emergency planning by the State. Experience supports the validity of this approach.[162/]

93. The State has determined that it is not necessary or desirable to provide the public with advance instructions or information about the radiation emergency plan for it to be effective.[163/] The reasons for the State's conclusion include the many variables involved and the possibility of confusion if the actual events should

---

160/ Testimony of Mr. Smith, pp. 1-3 (follows Tr. 1996); Tr. 3735-36, 3766, 3771-75.

161/ Testimony of Mr. Davies, pp. 13-15 (follows Tr. 1754); Tr. 1847-48, 3730, 3742, 3767, 3786-87.

162/ Tr. 3742, 3766-67.

163/ Testimony of Mr. Davies, pp. 13-15 (follows Tr. 1754); Testimony of Mr. Smith, page 5 (follows Tr. 1996); Set H Responses, Question H-33 (CCPE Exhibit H); Tr. 1823, 3763-64, 3767.

prove to be different from those postulated.164/ The public will be provided with specific notification as to the necessary protective actions at the time of an accident.165/ The methods and timing of carrying out specific public notification have been described. The Board concludes that the State's plan provides for adequate warning to the public in the event protective actions are required.

94. The Staff has testified that the Applicant has described adequate plans for coping with emergencies in accordance with the Commission's regulations.166/ The Board concludes that Applicant's radiation contingency plan in conjunction with the plans of the State of New York provide reasonable assurance that appropriate measures can and will be taken in the event of an emergency to protect the public health and safety.

164/ Testimony of Mr. Smith, page 5 (follows Tr. 1996); Tr. 3767.

165/ Testimony of Mr. Davies, pp. 5, 9-10, 17-18, 20 (follows Tr. 1754); Testimony of Mr. Smith, pp. 3-4 (follows Tr. 1996); Set H Responses, Question H-34 (CCPE Exhibit H).

166/ SSE, pp. 67-68 (follows Tr. 405).

XII. Risk-Benefit Determination

95. The Citizens Committee has asserted that the Board should balance the benefits to be derived from Indian Point 2 against the risks of operation of this facility in reaching the decision on the issues reflected in 10 CFR 50.57(a). The determination of the Board is that the issuance of an operating license for Unit No. 2 will not be inimical to the health and safety of the public. In reaching this conclusion the Board has evaluated the risks involved in the operation of this individual facility without balancing the benefits derived from such operation. Such balancing is not within the province of this Board in reaching the decision on the issues set forth in 10 CFR 50.57(a). Congress has determined the benefits accruing from the private development of nuclear power and this determination is reflected in the Atomic Energy Act of 1954, as amended.[167/]

167/ Tr. 905-906; Joint Motion, submitted by Citizen's Committee for the Protection of the Environment and Applicant dated March 16, 1971, Attachment B.

XIII. Conclusions

96. Upon the basis of consideration of the entire record in this proceeding, and in the light of the foregoing which constitute Findings of Fact and Conclusions of Law, the Board finds and concludes, with respect to the issuance of a license authorizing limited operation of Unit No. 2 for testing purposes at power levels up to 1379 Mwt (50 percent of full power), as follows:

(a) Construction of Unit No. 2 has been substantially completed, in conformity with Construction Permit No. CPPR-21, the application as amended, the provisions of the Atomic Energy Act of 1954 as amended ("the Act") and the rules and regulations of the Commission;

(b) Unit No. 2 will operate in conformity with the application as amended, the provisions of the Act, and the rules and regulations of the Commission;

(c) There is reasonable assurance (i) that the activities authorized by the operating license

can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

(d) Consolidated Edison is technically and financially qualified to engage in the activities authorized by the operating license in accordance with the rules and regulations of the Commission;

(e) The applicable provisions of 10 CFR Part 140 have been satisfied; and

(f) The issuance of the license will not be inimical to the common defense and security or to the health and safety of the public.

Respectfully submitted,

LEBOEUF, LAMB, LEIBY & MACRAE

By _____
Leonard M. Trosten
Partner

Dated: January 28, 1972

BEFORE THE UNITED STATES

ATOMIC ENERGY COMMISSION

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

CERTIFICATE OF SERVICE

I hereby certify that I have served a document entitled "Applicant's Proposed Findings of Fact and Conclusions of Law in the Form of a Proposed Initial Decision with Respect to Motion for 50 Percent Testing License - Part I" by mailing copies thereof first class and postage prepaid, to each of the following persons this 28th day of January 1972:

Samuel W. Jensch, Esq.
Chairman
Atomic Safety and Licensing Board
U.S. Atomic Energy Commission
Washington, D.C. 20545

Dr. John C. Geyer
Chairman, Department of Geography
and Environmental Engineering
The Johns Hopkins University
513 Ames Hall
Baltimore, Maryland 21218

Anthony Z. Roisman, Esq.
Berlin, Roisman & Kessler
1712 N Street, N.W.
4th Floor
Washington, D.C. 20036

J. Bruce MacDonald, Esq.
Counsel
New York State Department
of Commerce
112 State Street
Albany, New York 12207

Mr. R. B. Briggs
Molten Salt Reactor Program
Oak Ridge National Laboratory
P. O. Box Y
Oak Ridge, Tennessee 37830

Algie A. Wells, Esq.
Chairman
Atomic Safety and Licensing
Board Panel
U.S. Atomic Energy Commission
Washington, D.C. 20545

Honorable Louis J. Lefkowitz
Attorney General of the
State of New York
80 Centre Street
New York, New York 10013

Myron Karman, Esq.
Counsel, Regulatory Staff
U.S. Atomic Energy Commission
Washington, D.C. 20545

Angus Macbeth, Esq.
Natural Resources Defense
Council, Inc.
36 West 44th Street
New York, New York 10036

Leonard M. Trosten, Partner

LeBoeuf, Lamb, Leiby & MacRae
Attorneys for Applicant

13-3 ①