BEFORE THE
UNITED STATES OF AMERICA
ATOMIC ENERGY COMMISSION

In the Matter of CONSOLIDATED EDISON COMPANY OF NEW YORK (Indian Point, Unit No. 2)	))))))	Docket No. 50-247

STATEMENT OF PROPOSED FACTUAL FINDINGS
WITH REFERENCES TO SUPPORTING DATA
SUBMITTED BY THE CITIZENS COMMITTEE
FOR THE PROTECTION OF THE ENVIRONMENT

1. It is possible that the engineered safety features will fail to operate after a loss of coolant accident (hereinafter LOCA). (A-3; D-vii; D-1, 7; E-8-10; M-3; Staff answer to Q. H-56(d))

2. If the engineered safety features failed to operate after a LOCA the damage to people and property would be unacceptable and would violate the standards in 10 CFR Part 100.

   a. In the Matter of Consolidated Edison (Indian Point, Unit No. 3) (Docket No. 50-286) (hereinafter IP #3) Staff submittal dated June 20, 1969 responding to Board questions of June 12, 1969.

   b. FSAR Q. 14.1 and 14.2.

   c. WASH-740

   d. G - Q. 27 1/

   e. T

1/ See Appendix A for the key to citations.

3. The Applicant has not proven that the occurrence of a LOCA followed by a failure of engineered safety features is impossible nor sufficiently improbable to be disregarded.

a. Applicant has not proven that the ECCS will function as predicted in a LOCA.

1. There is inadequate data on rod bursting and swelling from reflooding of the core.

a) Rods will begin to fail at 1400°F (A-15) and all will fail when temperature reaches 2000°F. if internal pressure is 100 psi (Q-5, 11).

b) design pressure limits for this plant's fuel rods are 2250 psi. (FSAR 3.1.2-2)

c) fuel rod pressures of 50-250 psi are normal. (R-267)

d) rod burst causes rod to swell (Q-11, 5) and swelling can block flow. (Y-59, 92; P-24; M-15, 25, 27, 148-149)

e) Quenching can cause failure. (A-52; M-178) and tests to contrary are inadequate because

1) test temperature increase rate was well below actual temperature increase rates in a LOCA. (R-267; Q-6)

2) test did not include any allowance for zirconium-water reaction. (Q-6; M-15, 25, 27, 148-149).

3) disassembly of core affects cooling. (M-15)

f) Tests of rods did not show a good correlation with predicted results (A-49) and the tests excluded several factors which would make the results more severe.

1) single rod tests were conducted without simulation of core conditions such as stresses on the rods and variable heat changes. (P-2)

2) rod diameters differed substantially (15%) from those used in this reactor. (Q-5; FSAR Table 3.2.3-1)

3) observed and predicted results were not close particularly in the range where the probability of rod bursts increases substantially with the temperature. (Q-11)

4) to extent unirradiated tubes were tested the results showed that unirradiated tubes are not similar in performance to irradiated tubes. (Q-5, 8; R-268 (fig. III-66))

g) fuel rod swelling and bursting is not considered in the codes which analyze the LOCA. (M-140, 175-6)

h) early in the life of the fuel a LOCA will result in substantial swelling of the rods without bursting and thus substantial flow blockage of the ECCS water. (Q-5, 8; R-268 (fig. III-66).

2. There is inadequate data on the strength of the reactor vessel in a LOCA.

a) Ultrasonic tests (UT) only find flaws perpendicular to the wave and Radiographic tests (RT) only find flaws parallel to the wave (H-10(d), H-7, H-8; A-35)

b) vessel was tested with UT or RT but not both. (L-Q. 2 (p. 3); H-6; FSAR Table 4.5.1)

c) irradiation of the vessel wall can effect the growth of flaws and strength of material. (V-III-2-4; U-1-1; A-34-35; R-214-221)

d) test samples placed in the reactor to test for irradiation do not have any specific flaws in them to match the flaws in the reactor vessel. (Tech Specs. - 4.2-6 (Item 7.2); (H-12)

e) tests run on irradiated materials (irradiation was only for a short time compared to the 40 year life of the vessel (I-8; U-2-7)), included 1 and 2 inch materials and not the thicker plates used on the plant's reactor (U-3-1; FSAR Table 4.1-2), but even these tests showed adverse results in ultimate tensile strength and due to irradiation yield strength (U-3-3).

f) in service tests of the reactor vessel are not required for this plant (Tech Specs, Items 1.1, 1.2, 1.3, 1.4, 1.7).

g) applicant has not proven that techniques to conduct the in service test will be available within ten years. (H-10(i))

h) there is a need for more up to date codes than present ASME codes and Applicant doesn't even meet the present ASME Codes. (H-10(i) H-10(g); ORNL-NSIC-21 (p. 150))A-36-37)

i) reactor vessel inspectors were not fully qualified (H-15; H-16; A-35; ORNL-NSIC-21, (pp. 410, 51))

j) there is a need for general improvements in testing and inspectiny primary coolant materials. (Letter of November 24, 1965 from ACRS to Chairman Seaborg)

3. There is inadequate data on the effect of a metal water reactions and chemical explosions in a LOCA.

a) metal water reactions with steel and zirconium occur at 2000°F. - 2100°F. (A-47; Y-9, 55).

b) there is little known about metal water reactions and more data is needed (Y-55-59; A-51-52; M-172) through the use of an integral model and not isolated tests. (M-10, 171)

c) there is a need for more data on steam explosions. (A-53)

d) chemical reactions can be very serious. (M-3, M-6, M-8, 9, M-12).

e) metal water reaction can rupture the reactor vessel. (M-12, 13, 29, 120, 160-161; Y-88)

f) metal water reaction causes hydrogen build-up (M-12, 13, 29, 120, 160-161, 167, 168-170, 177) which can be reduced by use of nitrogen in the core. (M-13, 30; N-4-7 et seq.)

g) relatively small releases of hydrogen can build up in pockets in the reactor vessel and then explode. (M-1-6-7)

h) steam directly affects metal water reaction and Con Ed assumes none is left in their FLASH-R Code. (M-142; N-Fig. 3-10, N-4-3, 4-4 4-11, 4-12)

i) the stainless-steel water reaction is not considered by Con Ed although the inside liner of the reactor vessel is stainless steel (FSAR Table 4.2-1) and there is stainless-steel cladding on the burnable poison rods (FSAR 3.2.3-20). (M-63-72)

j) there is a substantial danger of loss of fuel integrity if clad reaches 2100°F. particularly if there has been substantial bursting of rods (Y-55-57)

b. Applicant has not proven that the containment spray will operate at a level of efficiency which will hold post-LOCA radioactive dangers to or below 10 CFR Part 100 standards.

1. all of the problems with respect to this issue raised by Board in IP #3 still remain. (T)

2. tests of the containment spray have not been run with simulation of containment atmosphere or with same pressure, temperature, wind velocities, etc. (W-438; T-56,61, 73; A-57-58) 2/ as post-LOCA conditions.

2/ These page references are to the pages of Initial Decision as issued by the ASLB.

3. the tests of the spray nozzles have not been run with randomly selected nozzles actually manufactured for this plant. (T-17,703-9)[3/]

4. there is no data to establish that plateout will occur when the spray is used. (T-71)

5. there is no data to establish that the variables in 2) above will not greatly affect the reliability of the spray system.

6. proprietary data before the Board and available to Intervenor substantiates the fact that many of the uncertainties associated with IP #3 in December, 1969 have not been resolved by that data. (WCAP 7499-L (dated April, 1970), pp. 3-1, 3-2, 5-1, 3-20, App. B-1)

7. sprays don't affect noble gas releases. (A-58, V-III-86-87)

8. test models used for the spray were only 1/3 as high as the IP #2 containment. (K-30-31)

9. with these uncertainties unresolved dose levels are too high. (T-72)

10. enviornment in which a spray operates and is needed is the environment following a LOCA and

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3/ These page references are to the pages of the CCH Atomic Energy Reporter.

failure of ECCS and this will be very volatile (FSAR, Section 14.3.4)

11. the spray effectiveness depends upon physical contact between the spray drops and iodine and it is critical to know precise operation of the system in the containment environment with its baffles, walls, corners, etc (FSAR, Figures 5.1-2, 5.1-3, 5.1-4, 5.1-5, 5.1-6) (T-17,703-9)

c. Applicant has not proven that the containment air recirculation and filtration system will operate at a level of efficiency which will hold post-LOCA radioactive dosages to or below 10 CFR, Part 100 standards.

1. the filter system is virtually inoperative in first 2 hours. (T-54)

2. filter system effectiveness is greatly affected by flooding (K-27) and Applicant merely assumes a 70% effective rate without justifying the failure to use a lower rate. (K-27-28; T-63-66)

3. staff safety analysis in this case assumes effectiveness of filters is substantially higher than the staff assumed for IP #3 (Staff Safety Evaluation - 42) without any justification for this optimism. (T-64)

d. Applicant has not proven that the weld channel penetration pressurization system and isolation valve seal water system will operate in a manner to keep post-LOCA radioactive dosages to or below 10 CFR Part 100 standards.

1. there is no frequent testing of the containment at design basis conditions after operation but tests are only twice in 10 years. (Tech. Specs. 4.4 -2, 4.4-3)

2. design criteria for tests would allow a 0.1% leak rate without correction (Tech. Specs. 4.4.-3), even though Applicant assumes no leak from operation of these systems (FSAR, 14.3.5-2) or leakage for only one minute at 0.1% (FSAR 14.3.5-14).

3. weld channel pressurization tests allow 0.2% leak rate (Tech Specs. 4.4 -4; FSAR 6.6-2, 6.6-8)

e. Applicant has not proven that the hydrogen-oxygen recombiner can prevent dangerous build-up of hydrogen in the containment.

1. hydrogen oxygen recombiner cannot operate if containment pressure is more than 5 psig. (H-22)

2. applicant has not demonstrated that in the event of a failure of the ECCS the build-up of hydrogen and pressure would not require use of the hydrogen oxygen recombiner before pressure levels were below 5 psig.

3. applicant has not proven that use of the containment vent in such case would not violate 10 CFR Part 100 standards particularly if the iodine removal effectiveness of filters were as conservative as postulated by the Staff or was ineffective due to flooding.

4. Analysis of the reliability of engineered safety features and their predicted performances are based upon inherently inadequate mathematical models and test procedures.

a. codes

1. failure to consider all relevant factors in codes. (G-Q-13; M-97, M-136)

a) effect on blow-down forces of proximity of break to the reactor is not considered. (A-12)

b) codes don't consider fuel rod swelling (M-140, M-175-6)

c) SATAN-V Code only can analyze two loops. (N-2-6, 2-7)

2. SATAN-V Code has built-in design limitations such as the fact that only 96 elements can be analyzed (N-2-7) and has failed to adequately simulate all aspects of pressure in core after LOCA (N-3-2).

3. FLECHT Code does not consider temperatures in excess of 2000°F. (N-3-8, 3-9) even though such temperatures are possible in LOCA (FSAR Figure 14.3.2-23) and it does not consider different temperatures in different channels. (Y-91)

4. codes are simplified in order to conserve the use of computer time. (M-136)

5. there is a need to verify the Code results before they are really acceptable. (I-Q1)

6. there is no evidence that code analysis and experimental results are sufficiently identical to warrant reliance on codes without experimental verification. (Q-8, Q-11; N-Figures 3-8, 3-9, 3-10)

7. there is a tendency of some of the operators of the codes (those who stand to gain economically if tests succeed) to find excuses for adverse test results based upon the unreliability of the model without a concommitant skeptisism about favorable test results. (N-3-3; Nucleonics Week, May 13, 1971).

b. <u>tests</u>

1. often do not include all of the relevant variables which will occur when the tested system will be actually needed. (See previous facts relating to rod quench test, rod burst tests irradiated steel specimens and containment spray)

2. often occur with tests of specific conditions but not all conditions simultaneously. (T-17,703-10)

5. Applicant has failed to prove that adequate emergency procedures have been established for the plant and is in violation of Appendix E of 10 CFR Part 50.

a. the applicant does not assume primary responsibility for advising the public of steps to be taken in the event of a radiation emergency. (H-32, 33, 36; G-Q. 14)

b. the alleged emergency plan of the State of New York is deficient in several critical respects.

1. there is no discussion of precise safety measures to be taken by the public or a program of training for the public in the use of these methods. (S-A5)

2. there is no evacuation plan and no public information with regard to use of such a plan.

3. there is no system which will guarantee warnings to <u>all</u> members of the public within a short time after the emergency and no program to train those responsible for giving warnings in order to prevent a panic from such announcements. (S-5, A4, B7)

4. the plan fails to reveal the precise conditions under which it will go into effect. (S-2)

5. there are no provisions for supplemental food or water supplies or control of shipment of contaminated products.

6. Applicant has not proven that occurrences which could result in releases of radioactivity to the environment are limited to those analysed in the FSAR.

a. A major melt-down of the core is not so improbable that it should be ignored.

1. only the ECCS can prevent a major melt-down in the event of a design basis accident. (A-47, A-56)

2. core disassembly could lead to a major melt-down (A-9) and this disassembly could be caused by excessive bursting of irradiated fuel rods or shattering of such rods from quenching if there has been zirconium-water reactions. (See previous facts demonstrating that the applicant has failed to prove that it has fully analyzed these factors.)

3. a rupture of the pressure vessel could cause a melt-down of the core (ACRS letter to Seaborg of November 24, 1965; A-156-157) and this is not incredible in todays large reactors. (See previous

facts demonstrating inadequacy of Applicants data on reliability of vessel, build-up hydrogen in core and reliability of computer codes upon which projected LOCA blow-down forces are based.)

4. Applicant has presented no reliable evidence to demonstrate that in a core melt-down there would be adequate protection of the public. (A-6, 47, 49, 56)

a) there has been no consideration of the affect of steam explosions following a core melt-down. (A-48, A-156-157)

b) effect of a melt-through of molten fuel is not considered and in fact a previously designed crucible for this reactor has been eliminated. (A-52, A-53, G-Q. 10)

c) none of the analyses of the effectiveness of the other safety systems including sprays, containment, etc. include consideration of the physical disruption which would be caused by a major melt-down of the core. (Answers to Questions by Citizens Committee for the Protection of the Environment - Set A-8, 9, 22; Set D-69, 88)

b. Sabotage by persons friendly to the United States but unfriendly to Applicant or the AEC, or merely deranged could occur and there is no evidence of adequate protection. (H-30, H-31)

a) there has been no analysis of a shaped charge fired against the containment. (H-31)

b) points of entry to the plant area are not guarded continuously but only by scheduled tours. (H-30)

c. The plant is located in an area where there is heavy airplane traffic and has failed to present data to prove that the crash of a plane into the reactor building is so remote that it need not be considered. (H-29)

1) one holding pattern for two-engine commercial jet aircraft virtually overlaps the plant site. (H-29 (figures 29-4, 29-5))

2) one approach route to Kennedy Airport goes directly over the plant site. (H-29 (figures 29-1, 29-5))

3) substantial numbers of commercial planes come within a 10 mile horizontal distance of the plant. (H-29(b))

7. Applicant and the Staff have presented conflicting and confusing analyses of postulated accident conditions and there is a lack of clear evidence that the requirements of 10 CFR Part 100 have been met or that the guidelines in

TID 14844 are being followed. (FSAR - Q. 14.1, Q 14.13, Section 14.3; H-45, 46; K-27-28; G - Q. 27; staff answer to Q. H-39 compared to Staff assumptions used in IP #3 Safety Evaluation (see Staff Response (dated June 20, 1969) to Board's Letter dated June 12, 1969, p. 7 in particular); Staff answer to Q. H-42, Q. H-45; T-53,61-66; Initial Decision in the Matter of Trustees of Columbia University (Docket No. 50-208) pp. 48-50)

8. A substantial portion of the basis for the assessment of the safety of this plant is based upon unsupported favorable assumptions and the exclusion of equally valid unfavorable assumptions.

a. Performance of the ECCS is based upon conclusions, the support for which are still in the experimental stage. (A-5-8; V-II-10 to II-16)

b. the design basis accident of a double ended pipe break is arbitrarily selected and may not be the worst possible occurrence. (A-42)

c. the use of redundant safety features may increase risks rather than reduce them. (A-42)

d. the probabilities of a major accident are impossible to determine (D-2, 3, 5; E-17)

e. where uncertainties of analysis exist conservative values are used but it is not possible to know if the value is sufficiently conservative unless you know the margin of error. (K-7)

f. assumptions in TID 14844 upon which the Staff relies in applying 10 CFR 100 standards exclude certain potential factors which could substantially alter those standards. (E-17; F-1 to 2)

9. The Citizens Committee for the Protection of the Environment wishes to cross examine the following individuals:

a. Robert A. Wieseman - Cross-examination will relate to the witness explanation (H-1) of the meaning of the term "probability". The purpose will be to have the witness provide a sufficiently quantitative definition to permit an evaluation of the use of this term by the Applicant in precluding consideration of certain events as improbable.

b. The individual from the Staff who reviewed the answer to question H-1 and concluded there was nothing to add. The cross-examination will be essentially identical to that of Mr. Wieseman.

c. At least one witness from the Staff and one witness from the Applicant who can discuss the criteria used in determining that the particular engineered safety features on this plant provide adequate and appropriate compensation for the deviation from the standards in 10 CFR Part 100, Section 100.10(b) and (c). The cross-examination will not be directed to the details of specific tests or analyses of particular safety features but rather to the use of such qualitative and underfined terms as "design margin", "conservative assumptions" and "reliable" which form the judgmental basis for the Staff and the Applicant relying upon engineered safety features.

d. The individual from the Staff responsible for the answer to the question (Tr. 487) relating to risk-benefit considerations. (F-3) Cross-examination will focus on the general criteria used by the Staff in determining the measure of the risk involved and the factors which enter into the conclusions about benefit.

e. Any witnesses offered by the Staff or Applicant with respect to any of the issues in this proceeding which have been raised by the Citizens Committee for the Protection of the Environment may be cross-examined.

APPENDIX A

The Citizens Committee for the Protection of the Environment intends to offer into evidence the following documents:[1/]

- Exhibit A - Emergency Core Cooling - Report of Advisory Task Force on Power Reactor Emergency Cooling
- Exhibit B - In the Matter of Trustees of Columbia University (Docket No. 50-208) Initial Decision (April 6, 1971)
- Exhibit C - Letter from ACRS to Chairman Seaborg dated November 24, 1965.
- Exhibit D - WASH-740 - Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants.
- Exhibit E - TID 14844 - Calculations of Distance Factors for Power and Test Reactor Sites.
- Exhibit F - Responses of the Division of Reactor Licensing to the Questions of the Atomic Safety & Licensing Board at the Hearing Session dated January 19, 1971.
- Exhibit G - Answers of Applicant to Questions Raised by Atomic Safety & Licensing Board on January 19, 1971 (Part 1) - ["Q." References are to the question numbers]
- Exhibit H - Answers to Set H of Questions by the Citizens Committee for the Protection of the Environment to Applicant - [Number references are to the question number]
- Exhibit I - Staff Answers to Set H and I of Questions By Citizens Committee for the Protection of the Environment to Applicant.
- Exhibit J - June 20, 1969 Response of Staff to June 12, 1969 Questions of Board in Indian Point, No. 3 Hearing (Docket No. 50-286).

1/ The Exhibits are cited in the statement of proposed facts by letter only but occasionally the full title of the document is given.

Exhibit K - Answers of Applicant to Questions Raised by Atomic Safety & Licensing Board on January 19, 1971 (Part II). ["Q." references are to the question numbers]

Exhibit L - Answers of Applicant to Questions Raised by Atomic Safety & Licensing Board on March 24, 1971 (Part I) - ["Q." references are to the question numbers.]

Exhibit M - Potential Metal-Water Reactions in Light-Water-Cooled Power Reactors - ORNL-NSIC-23.

Exhibit N - WCAP-7422 - Topical Report - Westinghouse PWR Core Behavior Following a Loss of Coolant Accident.

Exhibit P - WCAP - 7379 - Topical Report - Performance of Zircaloy Clad Fuel Rods During a Simulated Loss-of-Coolant Accident - Single Rod Tests (Vol. I)

Exhibit Q - WCAP-7379 - Topical Report - Performance of Zircoloy Clad Fuel Rods During a Simulated Loss-of-Coolant Accident - Single Rod Tests (Vol. II).

Exhibit R - Fundamental Nuclear Energy Research (1969) A Supplemental Report to the Annual Report to Congress for 1969 of the U.S. AEC - Pages 214-221, 265-269.

Exhibit S - Document Entitled New York State Emergency Plan For Major Radiation Accidents Involving Nuclear Facilities - [The document is offered as evidence of what the Applicant claims it will rely upon for emergency procedures and not as a genuine copy of the actual New York State Plan.]

Exhibit T - In the Matter of Consolidated Edison Company (Indian Point Unit No. 3) (Docket No. 50-286) Initial Decision with Separate Opinions.

Exhibit U - WCAP 7561 - Heavy Section Steel Technology Program Technical Report No. 9 (August, 1970).

Exhibit V - WASH-1146 - Water Reactor Safety Program Plan.

Exhibit W - Extract from Nuclear Technology, A Journal of the American Nuclear Society (Volume 10, Number 4, April, 1971) pages 436-443.

Exhibit X - Responses of Applicant to the following Questions of Citizens Committee for the Protection of the Environment; Set A-8, 9, 22; Set D-69, 88.

Exhibit Y - Emergency Core-Cooling Systems for Light-Water-Cooled Power Reactors-ORNL-NSIC-24.

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UNITED STATES OF AMERICA
ATOMIC ENERGY COMMISSION

In the Matter of CONSOLIDATED EDISON COMPANY OF NEW YORK (Indian Point Unit No. 2)	Docket No. 50-247

CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing Summary of Position of the Citizens Committee for the Protection of the Environment with Respect to the Application for an Operating License for Indian Point, Unit No. 2; and Statement of Proposed Factual Findings with References to Supporting Data Submitted by the Citizens Committee for the Protection of the Environment were hand-delivered the 3rd day of June, 1971 to Arvin E. Upton, Esq., LeBoeuf, Lamb, Leiby & MacRae, 1821 Jefferson Place, N. W., Washington, D. C. 20036, and that copies were mailed, postage prepaid, on the 4th day of June, 1971, to the following:

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