3/22/73

BEFORE THE
UNITED STATES OF AMERICA
ATOMIC ENERGY COMMISSION

In the Matter of

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| CONSOLIDATED EDISON COMPANY | ) | |
| OF NEW YORK (Indian Point, | ) | Docket No. 50-247 |
| Unit No. 2) | ) | |

MOTION TO REQUIRE APPLICANT
AND STAFF TO PROVIDE SPECIFIC RESPONSE
TO THE CCPE RADIOLOGICAL CONTENTIONS

We are today filing a statement of specific contentions on outstanding radiological matters with respect to fuel densification. The purpose of both filings is to facilitate orderly resolution of the issues. Neither Applicant nor the Staff have previously filed any substantive response with respect to these issues (except fuel densification) limiting their position to alleged procedural irregularities. We believe now they must respond on the merits and do so in sufficient time to allow us to prepare our case for the proposed hearing on April 9, 1973.

We therefore move this Board to order that on or before March 30, 1973, the Applicant and the Staff deliver to CCPE and serve on the parties, their detailed response to the CCPE contentions filed today and the evidence they intend to introduce into the record with respect to those contentions and on or before April 4, 1973, Applicant and the Staff deliver to CCPE and serve on the parties with respect to the CCPE fuel densification contentions their detailed response to the CCPE contentions and the evidence (if not previously identified) that they intend to introduce into the record with respect to that issue.

It is further requested that the Board provide that failure to meet these responsibilities will result in excluding from the hearing any contentions and evidence not specifically identified on the date required or alternatively postponement of the hearing on the radiological issue to which the contention or evidence relates until two weeks after the contention is filed with proper specificity or the evidence is filed or both.

The Board's authority to take the action requested here is provided by 10 CFR Part 2, Section 2.718 (c) (e) and (1).

Respectfully submitted,

Anthony Z. Roisman
Counsel for Citizens Committee
for Protection of the Environment

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CITIZENS COMMITTEE FOR
PROTECTION OF THE ENVIRONMENT
STATEMENT OF CONTENTION WITH RESPECT
TO FURTHER RADIOLOGICAL ISSUES

## A. THIN WALLED VALVES

By letter dated June 22, 1972, the Regulatory Staff, acting through Mr. James P. O'Reilly, Director, Directorate of Regulatory Operations, Region 1, advised the Applicant that the Staff had reason to believe that Indian Point #2 may contain valves with wall thickness below minimum requirements specified in applicable codes. Because of this problem the Staff, in effect, determined that the Applicant would have to supply additional information to verify that all valves important to nuclear safety met the minimum thickness requirements of the specified codes or standards.

Applicant responded to the letter by a letter from Mr. William Caldwell dated July 21, 1972. In that letter they acceded to the Staff request and accepted the Staff position that further verification of minimum wall thicknesses for the valves in question was required. A proposed procedure was set forth and further data on the procedure was promised August 31, 1972. No further communication on this subject has been received by us from either the Applicant or the Staff.

Based on these two letters (which we will introduce into evidence) we make the following contentions:

1. Applicant has not proven that it is in compliance with the requirements of 10 CFR Part 50, Appendix A Design Criteria 1 because records of design, fabrication; erection and testing of all valves which are important to nuclear safety have not been maintained by or under Applicant's control.

2. Applicant has not proven that it is in compliance with the requirements of 10 CFR Part 50, Sections 50.55a (a) and (f)(1), and 50.57(a)(1) and Safety Guide 26 because the valves which are important to nuclear safety have not been shown to have walls of the minimum thickness required by the applicable codes and standards.

3. Applicant's proposed method to detect thin-walled valves is inadequate to determine that all valves important for nuclear safety meet applicable codes and

standards because the selection of one valve of each size from a manufacturer and examination of only six areas on each valve represents an irrational random sample which does not guarantee discovery of all defective valves. The Staff letter does not establish that thin walls have occurred in all valves of a given size from one manufacturer or that the fault is sufficiently large in surface area that six selected area UT examinations for each valve will disclose the flaw.

4. Applicant has not proven that if faulty valves are discovered it has a method for correction of the defect which will bring its valves in compliance with the codes and standards applicable to the plant.

5. The Staff position that verification of conformance with the applicable codes and standards as required by Sections 50.55a, Design Criterion 1 and Safety Guide 26 wait until after the plant has begun to operate (up to June 22, 1975) is contrary to the requirements of the regulation, the design criterion and the safety guide. The determination apparently involves an implicit risk/benefit determination outside the confines of Commission

Regulations which is itself contrary to the Staff position in this proceeding (Staff Response to Questions dated October 21, 1971 and Tr. 905) and violative of the Atomic Energy Act of 1954. In addition, such a risk/benefit determination, if legal, must include a precise quantification of the risk and how it was determined and this has not been done.

The evidence to support these contentions consists of the two letters identified above from Mr. O'Reilly and Mr. Caldwell.

B. <u>RUPTURE OF WATER AND STEAM LINES OUTSIDE THE CONTAINMENT</u>

In a letter dated December 19, 1972, the Regulatory Staff (A. Giambusso, Deputy Director for Reactor Projects, Directorate of Licensing) advised the Applicant that here was a need to analyze and adequately document the consequences of rupture of main steam and feedwater lines outside the containment to determine if, as a result of those ruptures, safety systems could be impaired. Reference was made to 10 CFR Part 50, Appendix A, Design Criterion 4 and its requirement that all safety systems be designed to withstand accidents to other plant systems outside the nuclear power unit. The Staff all observed:

> We note however that the auxiliary feedwater pumps are located in an enclosure adjacent to the main steam and feedwater line containment penetrations. From this it appears that damage to the auxiliary feedwater system might result from a postulated pipe failure in the main steam or feedwater lines.

To date we have received no substance response fomm the Applicant with respect to this matter. Based upon the information contained in the letter from the Staff we contend:

1. Applicant has not proven that it complies with Design Criterion No. 4 of 10 CFR Part 50 Appendix A. In particular, Applicant has not proven that due to the location of the main steam and feedwater lines a pipe rupture of either of those lines can not damage the auxiliary feedwater system as a result of pipe whip, flooding, damage from pieces of metal occassioned by the break or jet forces.

C. PRESSURE VESSEL RUPTURE

In its Initial Decision on the 50% testing license the Board expressed concern with the problem of pressure vessel integrity as it relates to full power operation. See Initial Decision pp. 16-17. It also certified a question relating to the scope of the proper inquiry by the Board into reactor vessel integrity. By Memorandum and Order of October 26, 1972, the Commission responded to the certified question (p. 4):

> We adopt the view expressed by the staff as consistent with the language of the regulation and the underlying Statement of Considerations. Where there are matters raised in a case that are "special safety significance", supplementary measures in respect to the facility under review are an appropriate subject of hearing exploration. The certified question, insofar as it deals with the admission of evidence pertaining to pressure vessel integrity in licensing proceedings, is therefore, answered in the negative. */

*/ To warrant inquiry, the evidence must be directed to the existence of special considerations involving a particular facility in issue. Licensing Boards, in their discretion, are empowered to exclude contentions or challenges which have no substantial or prima facie basis, or which merely amount to generalized attacks upon the standards presently required by the regulations.

Consistent with this decision, this Board requested Applicant to provide certain documents and data relevant to the reactor vessel. (Tr. 7431-7433)

At the outset the Board here will face the allegation by the Applicant and the Staff that there are no special considerations involving Indian Point #2 which warrant further inquiry. */ We believe Indian Point #2 does present special considerations.

10 CFR Part 100 and T.I.D. 14844 (CCPE ex. D) are premised on the concept that engineered safety systems are to be used to compensate for the safety afforded to the public by distance from a reactor. As the reactor moves closer to a population center there is need for more engineered safety systems. In 1965, the ACRS in writing about reactor pressure vessels specifically referred to the special circumstances created by the proximity of a reactor,

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*/ Applicant and Staff will also contend that our contentions are attacks on the standards presently required by the regulations. But the Commission Memorandum only precludes "generalized attacks" on standards and that is consistent with the view that generalized (i.e. non-specific) contentions are not allowed. If our contentions are attacks on the standards they are specific attacks and therefore allowable. See Section 2.758. However, our contentions are not attacks upon the validity of a standard adopted by regulation. We contend that the special circumstances present here warrant imposing standards more stringent that the "minimum" standards set by the regulation.

particularly a large reactor to large population centers. CCPE Ex. C. In the most recent issue of Nuclear Technology (March, 1973) it is reported that (p. 282):

> Sweden is considering using pipe rupture as the maximum credible accident for accident analysis in non-urban sites and vessel rupture for urban site analysis, a departure from present United States of America Atomic Energy Commission policy.

As we indicated in our Proposed Findings of Fact 1.d. this reactor is located in close proximity to the largest city in the United States and if allowed to operate will be one of the largest operating reactors in the United States. At least at one time the ACRS and the Staff considered Indian Point #2 to be a unique plant because this reactor originally included a crucible designed to catch fuel in the event pressure vessel rupture - a condition never imposed on any other plant. */ Finally, this Board has, in effect, ruled that Indian Point #2 does involve special considerations with respect to the vessel in its Initial Decision on 50% Testing (pp. 13-17, 29-30) and in questions put to the Applicant subsequently (Tr. 7431-33).

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*/ The basis for removal of the crucible was the installation of the accumulators for ECCS performance. This of course did not increase pressure vessel integrity but only, at best, reduced the chance of a failure of ECCS and a core meltdown from that cause.

In light of these special circumstances we believe it is appropriate to explore at this hearing the risk of a pressure vessel rupture at Indian Point #2 to determine whether the probablity of that event and the potential consequences if it should occur warrants the conclusion that no further engineered safeguards are required for Indian Point #2.

A risk assessment involves a determination of probablity of occurrence of the event and analysis of the consequences. In our Proposed Findings of Fact we relied upon some of the available data which demonstrates the totally unacceptable consequences of pressure vessel rupture and uncontrolled core meltdown. CCPE Proposed Findings of Fact 1.a-c, e. In a recent AEC Memorandum from Dr. Steven Hannauer to Dr. Hendrie and Mr. Giambusso (Regulatory Staff Internal Memoranda No. 2 AM-83 dated June 1, 1972) it is stated that:

> ...a probability of an uncontained accident of $10^{-6}$ per reactor years from a specific identifiable cause realistically evaluated is too high for me.

This has been adopted as a Regulatory Staff rule of thumb for a risk assessment cut-off point.

In The Safety of Nuclear Power Reactors (Drafts) WASH-1250 p. 6-39 the probablity of a catastrophic failure of a reactor pressure vessel is given as $10^{-6}$ /year.*/ Thus the risk of the accident falls within the limits used by the Staff to determine which events are unacceptably likely to occur. In reaching the conclusion on the likelihood of vessel rupture WASH-1250 states (p. 6-38):

> It will be noted that the numbers given in Table 6-9 for primary coolant pipe rupture and reactor pressure vessel (catastrophic failure) are not based on "reality", since no such random or spontaneous failure in nuclear grade vessels and primary piping have occurred. These are guesses which generally have to be based on two factors: a) records of failures from the past experience of non-nuclear industries with respect to pressure vessels and piping built to different, and generally less stringent, standards than those used in nuclear plants, and frequently used under harsher circumstances (e.g., corrosive chemicals), and b) the exercise of engineering judgments to decide how these recorded failures can be translated to random failure rate numbers for components built to more stringent standards.

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*/ The data relates to a "typical" 1000 mw reactor but the difference between that reactor and Indian Point #2 is irrelevant insofar as vessel size and pressures on it are concerned.

The approach suggested in WASH-1250 was in part utilized by Dr. Wechsler in his draft report The Radiation Embrittlement of Pressure Vessel Steels and the Safety of Nuclear Reactor Pressure Vessels (March, 1970). His analyses of the non-nuclear pressure vessel rupture revealed that in some respects nuclear vessels are less likely to rupture than non-nuclear and in some instances they are more likely to rupture. His conclusion, there is that assuming that pressure vessels will not rupture and thus providing no engineered safety feature to ameliorate the consequences is questionable in light of our present lack of knowledge about the causes and likelihood of vessel rupture.*/

Part of Dr. Wechsler's analysis involves conversion of the non-nuclear data into relevant information for nuclear vessels. The higher standards and higher quality metals used by nuclear vessel make it more reliable. But the greater difficulty in providing homogeneous structure within the thick metal used in nuclear pressure vessels makes it problematical that the high standards are in fact achieved. Moreover, the welding of thick materials is substantially more difficult. Also as this Board has observed, in-service inspection is critical to vessel integrity

*/ The final version of the report is still not ready. We understand that Dr. Wechsler still adheres to this position and still relies upon the data contained in the report, among other data, to substantiate it. In this event we feel it appropriate to rely on the document and to request the Staff to make Dr. Wechsler available to expand upon his ideas in this proceeding.

but is difficult for the nuclear vessel and as to some aspects of inservice inspection Indian Point #2 will not guarantee that the methods available to conduct the required inspection will be developed in time. Finally, the effects of radiation-embrittlement on NDTT causes the nuclear vessel to have an ever more easily attained brittle fracture temperature. All of these factors lead Dr. Wechsler to his cautious conclusions.

Professor Norman Rasmussen is currently under contract with the AEC to do a study of the risk of accidents in nuclear plants. He is essentially following the WASH-1250/Wechsler approach and has statisticians compiling and analyzing data on non-nuclear vessel failures. Eventually metallurgists, presumably at Oak Ridge, will apply their expertise to explain the relevance of the non-nuclear vessel analysis to nuclear vessels.

In the AEC Water Reactor Safety Program Augmentation Plan dated November, 1971 (pp. 3-7) the lack of any present data on primary system integrity, the need for such data and the manner in which lack of funds, inattention to the problem and unanticipated complexities caused this technological gap is discussed. Presumably the unfinished Rasmussen study is in part responsive to this concern.

This data, contained in the documents identified above and in this testimony of Drs. Wechsler and Rasmussen whom we request the Staff to make available for this hearing on April 9, 1973, are the bases for the following contentions:

1. Indian Point #2 because of its promixity to large population concentrations, its size, and its prior history (including the requirement to use a crucible to ameliorate the consequences of vessel rupture) presents a special case in which imposition of requirements with respect to the pressure vessel which are more stringent than the minimum requirements provided by AEC regulations is warranted.

2. The probablity of a rupture of the Indian Point #2 pressure vessel is no lower than $10^{-6}$ which in light of the proximity of the plant to large population centers and in light of the Staff standards requires inclusion of additional safety system to ameliorate the consequences of pressure vessel rupture.

3. Information is not available to demonstrate that reactor pressure vessel failure can be disregarded in safety analyses, even if the vessel meets presently applicable codes and standards. The areas of uncertainty in predicting the probability of pressure vessel rupture include:

a. Lack of knowledge sufficient to confirm that metallurgical properties meant to be included in the pressure vessel were in fact achieved for a vessel fo this thickness;

b. Difficulty in properly welding materials of the thicknesses involved;

c. Inability to conduct proper inservice inspections.

d. Lack of full understanding of the consequences of radiation-embrittlement on the pressure vessel of this reactor during its 40 year life.

e. Lack of any analysis in this case of the possibility of a rupture or crack at or near the vessel nozzle propagating into the vessel itself.

## CONCLUSION

With respect to all of these contentions, we would anticipate needing no more than one day to present our direct case (document and letters) and questioning Drs. Wechsler and Rasmussen. If subsequent submittals other than this data are made by any other party we reserve the right to expand our contentions, our direct case, our cross-examination, and our hearing time and to obtain discovery and recess and postponement of the conclusion of the radiological hearing if warranted.

Respectfully submitted,

Anthony Z. Roisman
Counsel for Citizens Committee
for Protection of the Environment

Dated: March 22, 1973