UNITED STATES OF AMERICA
ATOMIC ENERGY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

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

AEC REGULATORY STAFF'S BRIEF ON CERTIFICATION OF JULY 14, 1972

I. INTRODUCTION

This brief is submitted pursuant to the Order of August 1, 1972, by the Atomic Safety and Licensing Appeal Board, directing the filing of briefs by the parties, to assist the Appeal Board in reviewing the question certified by theAtomic Safety and Licensing Board (ASLB) in its Initial Decision of July 14, 1972.

The certification by the ASL Board asked the following:

> "Is it the position of the Commission that the measures taken to assure the integrity of the pressure vessels for light water reactors have been demonstrated and documented sufficiently that protection against the consequences of failure of the reactor vessel need not be included in the design of the plant and evidence concerning the integrity of the pressure vessel should not be adduced in the licensing proceedings?"

For the reasons set forth hereinafter, the Staff would urge the Appeal Board to answer the certified question in the negative, _i.e._, pressure vessel integrity is a proper area of inquiry during a licensing proceeding.

II. BACKGROUND TO CERTIFIED QUESTION

In the Indian Point Unit No. 2 proceeding, the intervenor contended that rupture of the reactor vessel is an accident against which Unit No. 2 must be designed.

In its Initial Decision, the ASLB noted that the paragraph on loss-of-coolant accidents and footnote 1 of the section "Definitions and Explanations" of Appendix A of 10 CFR Part 50 includes a break in the reactor vessel in the category of accidents defined as loss-of-coolant accidents. Accordingly, the Board requested that evidence be adduced concerning integrity of the reactor vessel.1/

The applicant submitted approximately 320 pages of testimony2/ in response to the inquiry by the ASLB. Thereafter, the Staff responded to both the inquiry of the ASLB and the Applicant's submission.3/

1/ Tr. 1658, July 16, 1971.

2/ Additional Testimony of Applicant Concerning Reactor Vessel Integrity, dated September 17, 1971 (introduced in evidence at Tr. 1932).

For earlier submissions by the Applicant concerning pressure vessel integrity, see:

1. Final Safety Analysis Report, Section 4.1.
2. Applicant's Responses to Round Two Questions Submitted by the Citizens Committee for the Protection of the Environment (CCPE) on March 9, 1971, Pt. I, dated March 29, 1971, and Pt. II, dated March 31, 1971, Questions H-1, H-6 through H-16, H-28 and H-43 (CCPE Exh. H).
3. Answers of Applicant to Questions Raised by ASLB on January 19, 1971, Pt. I, March 11, 1971, Questions 2, 3 and 34 (follows Tr. 665).
4. Answers of Applicant to Questions Raised by ASLB on March 24, 1971, Pt. I, dated May 8, 1971, Question 2 (follows Tr. 728).
5. Answers of Applicant to Questions Raised by ASLB on March 24, 1971, Pt. II, dated July 6, 1971, Questions 3 and 11 (follows Tr. 888).
6. Answers of Applicant to Questions Raised by ASLB on May 13, 1971, dated July 6, 1971, Questions 9 and 10 (follows Tr. 890).

(continued)

In its 32-page Report, the Staff provided testimony in support of its conclusion that the reactor vessel of Unit No. 2 can be operated over its service lifetime with a negligible risk of failure. "Failure" is defined in the Report as "a vessel rupture of such an extent that the capability of emergency core cooling systems to adequately cool the core may be impaired."4/

See also:

1. 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, Pt. I, dated January 28, 1972, Findings No. 36-42 and references contained therein.

2. Applicant's Reply to Proposed Findings of Fact and Conclusions of Law Submitted by CCPE, dated March 10, 1972, Responses 9 through 9.i.4.

3/ 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 (introduced in evidence at Tr. 2715).

For earlier submissions by the Staff concerning pressure vessel integrity, see:

1. Staff Safety Evaluation (follows Tr. 405).

2. AEC Regulatory Staff Responses to Second Round Questions of the CCPE, Answer to Question 8, Set I (CCPE Exh. I); CCPE Exh. I, Comments on Set H Responses, Answer Q. 55 (refers to Staff Responses A-44--contained in letter to Mr. Roisman from Mr. Karman, dated January 11, 1971).

3. Responses of the DRL to the Questions of the ASLB at the Hearing Session dated January 19, 1971, Question (Tr. 491) (introduced into evidence at Tr. 917).

4. Responses of the DRL to the Questions of the ASLB at the Hearing Session dated March 24, 1971, Question (Tr. 682-683), Question (Tr. 683-684) and Question (Tr. 690-691) (introduced into evidence at Tr. 917).

5. Responses of the DRL to the Questions of the ASLB at the Hearing Session dated May 13, 1971, Question 6 (introduced into evidence at Tr. 917).

See also:

1. Responses of Staff to Proposed Findings of Fact of CCPE, dated March 10, 1972, Responses 9.(a.), (b.), (c.), (d.), through 9.(i.).

For transcript pages where pressure vessel integrity was discussed, see Tr. 976-82, 1002-04, 1054-67, 1106-46, 1167, 1247, 1652-67, 1893-98, 1906-07, 1914-15, 2032-58, 3386-91, 3934-81, 3985-91.

4/ Report by the AEC Regulatory Staff, dated October 26, 1971, p.1.

In its Initial Decision (pp. 14-15), the ASLB summarized the salient portions of the testimony presented:

> "The essence of all this testimony is that the reactor vessel for Unit No. 2:
>
> "1. Was designed and fabricated to the high standards of quality required by the 1965 Edition of Section III of the ASME Code, the 1965 Summer Addenda and Code Cases and the Westinghouse equipment specification.
>
> "2. Was made from materials of controlled and demonstrated high quality.
>
> "3. Underwent extensive inspection and testing to provide substantial assurance that the vessel will not fail because of material or fabrication deficiencies.
>
> "4. Will be operated under conditions and procedures and with protective devices that provide assurance that the reactor vessel design conditions will not be exceeded during normal reactor operation or during most upsets in operation and that the vessel will not fail under the conditions of any of the postulated accidents.
>
> "5. Will be subjected to monitoring and periodic inspection in order to demonstrate that the high initial quality of the reactor vessel has not deteriorated significantly under the service conditions."

According to the Initial Decision, neither the Applicant nor the Staff produced to the satisfaction of the ASLB any documentary evidence showing any official declaration by the Commission, the Advisory Committee on Reactor Safeguards (ACRS), or the Staff that a rupture of the reactor vessel is not an accident which must be designed against in a pressurized water reactor plant. Hence, the ASLB made its certification to this Appeal Board.

III. PROTECTION AGAINST THE CONSEQUENCES OF PRESSURE VESSEL FAILURE IS A PROPER AREA OF INQUIRY IN A LICENSING PROCEEDING

The Staff position is that measures taken to assure the integrity of pressure vessels in light water reactor plants may be properly considered in licensing proceedings. The purpose of such consideration would be to determine whether the measures taken or proposed to be taken are sufficient to demonstrate that the pressure vessels can be operated over their service lifetimes with a negligible risk of failure.5/ In addition, it is the Staff position that the inherent or specific design features provided in a facility design that provide protection against the consequence of pressure vessel failure are also a proper subject of inquiry in a license proceeding.

Specific aspects of nuclear facility design which must be considered in assessing the acceptability of facility design for the purposes of Commission licensing are set forth in the General Design Criteria, Appendix A to Part 50. No provision of the General Design Criteria requires protection against the consequences of a failure of a reactor vessel. Rather, a number of the key provisions of the Criteria require designs of such character as to minimize the likelihood of such a failure. See particularly Criteria 14, 15, 30, 31, and 32. On the

5/ "Failure" is defined by the Staff as pressure vessel rupture of such an extent that the capability of emergency core cooling systems adequately to cool the core may be impaired.

other hand, there has been no official declaration by the Commission that pressure vessel failure is not an accident which must be considered in the design of light water reactor plants, and the definition of a loss-of-coolant accident in Appendix A to 10 CFR Part 50 does not exclude the hypothetical failure of a pressure vessel within the reactor coolant pressure boundary of light water reactor plants.

In addition to the specific design aspects which must be addressed in assessment of a particular facility, as provided in Part 50 of the Commission's Regulations, Part 100 provides a list of the factors that must be considered in evaluating the suitability of a proposed reactor site. These factors include: (1) characteristics of reactor design and proposed operation; (2) population density and use characteristics of the site environs, and (3) physical characteristics of the site. Part 100 further provides that a proposed site may be found to be acceptable if the design of the facility includes appropriate and adequate engineering safeguards that compensate for any unfavorable site characteristics. In addition, this Part requires that a site be assessed on the basis of an assumed maximum credible fission product release. The exposures associated therewith would not be exceeded by those from any accident considered credible for that facility.

Accordingly, in assessing a particular facility, it is necessary to determine whether the design of the facility includes appropriate and adequate design features to compensate for any unfavorable site characteristics and whether the consequences associated with the accident assumed for determining the siting characteristics described in Part 100 are exceeded by those from any other credible accident. This, then, imposes more general requirements, in addition to the specific requirements of Part 50, concerning aspects of facility design which must be considered.

In the Indian Point Unit No. 2 proceeding, the Applicant considered a range of credible accidents as design basis accidents, of which the event with the greatest computed exposures was the loss-of-coolant accident. This consideration by the Applicant corresponds to the practice with all large water-cooled nuclear reactor facilities licensed to date. This consideration further corresponds to the specific requirement of the General Design Criteria that a loss-of-coolant accident must be considered in the design of the facility.

The question as certified thus comes to bear directly upon whether reactor pressure vessel failure is a credible event which must be considered in assessing the adequacy of design and location of any or all nuclear reactor facilities.

In all cases evaluated to date, the Staff has concluded that the probability of failure of the pressure vessel is so low as to be incredible and has not required consideration of the consequences of such failure in the assumptions employed in determining site suitability.

As stated in its Report, the basis for the Staff's confidence concerning the low probability of pressure vessel failure includes the fact that design, material, fabrication, inspection, and quality assurance requirements for nuclear pressure vessels are specified in Section 50.55a of the Commission's Regulations, in Appendix A to Part 50, "General Design Criteria for Nuclear Power Plants," in Appendix B to Part 50, "Quality Assurance Criteria for Nuclear Power Plants," and in proposed Part 50, Appendices G, "Fracture Toughness Requirements," and H, "Reactor Vessel Material Surveillance Program Requirements," which are now used as interim guidance in the licensing process. Such Commission requirements reflect the Staff's experience, gained from review of all individual licensed facilities, as well as from extensive participation by the Staff in the development of industry codes.

Actual operation of licensed facilities offers empirical evidence supporting the Staff position. As the Staff pointed out in its Report of October 26, 1971, (p. 3):6/

> "Service and operator experience to date has provided confirmation of the quality and reliability expected of nuclear reactor pressure vessels. From data available to date, 95 pressure vessels of commercial pressurized and boiling water reactor plants have successfully completed over 3,500,000 [vessel-] operating hours without any structural failure and without evidence of any unanticipated problems which could be related to potential vessel failure. This experience, which represents 400 vessel-years of reliable and safe operation, includes nuclear pressure vessels which have seen as much as 10 years of operating service."

6/ See note 3, supra; see also Chart, "Nuclear Pressure Vessels - Operational Service, Statistics" (follows Tr. 3948).

During the course of the Indian Point Unit No. 2 hearing, both the Applicant and Staff produced expert witnesses who testified that the probability of pressure vessel failure is so low as to be incredible.

Applicant's witness, Mr. Robert A. Wiesemann, testified:

> "Q.***Do you mean to say that an explosive rupture is possible?
>
> "A. In my professional judgment it is incredible that such an event should occur." (Tr. 976)

The witness later defined "incredible" as "something...sufficiently remote that it does not have to be considered." (Tr. 985)

Staff witness, Karl Kniel, who was Senior Project Leader in the Division of Reactor Licensing (now Directorate of Licensing) explained the meaning of "incredible," as used by the Staff:

> "A credible event is something we design against to prevent a public hazard. <u>An incredible event is something we do not design against.</u>" (Tr. 1112 - emphasis added.)

Witness Kniel also stated that the Commission's General Design Criteria for nuclear power plants, Appendix A to Part 50, reflect the judgment of the Staff, and not necessarily an applicant's views.

> "...[T]he Atomic Energy Commission conducts its review independent of what the Applicant does. The Atomic Energy Commission has published a series of criteria, the General Design Criteria, which reflect the judgment of the technical staff as to what considerations should be given to design of nuclear power plants.

* * *

"***We certainly use the applicant's own efforts in trying to decide whether an event should be considered as credible or incredible and should be designed against or not designed against. In the final analysis, it is our own judgment as to which procedure to follow." (Tr. 1107-08, 1111).

Nonetheless, evaluation of each application requires assessment that the facility characteristics are in conformity with Parts 50 and 100. Such assessment requires determination that the design characteristics of the facility are such that consequences of accidents assumed for purposes of determining siting suitability are not exceeded by those of any accident considered credible. Each such assessment requires careful consideration of the particular aspects and features of the specific facility and site under consideration, to assure that for such facility all credible events are considered.

For the foregoing reasons the Staff believes that pressure vessel integrity in a light water reactor plant is a proper area of inquiry during a licensing proceeding. However, there is no requirement under the Commission's Regulations, nor need there be, for protection against the consequences of pressure vessel failure in all facilities, and protection against the consequences of pressure vessel failure need not be required for any facility, unless it has been determined that for such facility there are special considerations that make it necessary that potential pressure vessel failure be considered.

Respectfully submitted,

Winston M. Haythe
Counsel for AEC Regulatory Staff

Dated at Bethesda, Maryland,
this 21st day of August, 1972.

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NOTICE OF APPEARANCE

Notice is hereby given that the undersigned attorney herewith enters an appearance in the captioned matter. In accordance with Section 2.713, 10 CFR Part 2, the following information is provided:

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Winston M. Haythe

Winston M. Haythe
Counsel for AEC Regulatory Staff

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CERTIFICATE OF SERVICE

I hereby certify that copies of "AEC Regulatory Staff's Brief on Certification of July 14, 1972," dated August 21, 1972, in the captioned matter, and copies of "Notice of Appearance" for Winston M. Haythe, Esq., dated August 21, 1972, have been served on the following by deposit in the United States mail, first class or air mail, this 21st day of August, 1972:

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