8-21-72

BEFORE THE UNITED STATES

ATOMIC ENERGY COMMISSION

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

BEFORE THE ATOMIC SAFETY
AND LICENSING APPEAL BOARD

APPLICANT'S BRIEF RELATING TO THE QUESTION CONCERNING REACTOR VESSEL INTEGRITY CERTIFIED BY THE ATOMIC SAFETY AND LICENSING BOARD

I.
INTRODUCTION

In its Initial Decision dated July 14, 1972, the Atomic Safety and Licensing Board ("Licensing Board") certified a question [1/] to the Atomic Safety and Licensing

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1/ The Licensing Board, on page 16 of its Initial Decision authorizing the issuance of an operating license for Unit No. 2 for testing purposes at power levels up to 20 percent of full power, certified the following question:

> "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?"

Rhearing

Appeal Board for determination and guidance in the pending proceeding for a full-term, full-power operating license for Indian Point Unit No. 2 ("Unit No. 2"). The question certified encompasses two points: (1) Whether Unit No. 2 need be designed against the consequences of failure (i.e., rupture) of the reactor vessel; and (2) Whether additional evidence concerning the integrity of the reactor vessel for Unit No. 2 should be adduced in the pending proceeding.[2/]

Applicant maintains that Unit No. 2 need not be designed against the consequences of a postulated rupture of the reactor vessel because of the measures which have been and will be taken, as required by the Commission's regulations, to assure that the reactor vessel will not fail. The regulations of the Commission, the positions of the Regulatory Staff and the Advisory Committee on Reactor Safeguards and the consistent regulatory practice of the Atomic Energy Commission support this position. Applicant submits that it is unnecessary to

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2/ Applicant does not consider that the Licensing Board has questioned the need for the detailed evidence on this subject already adduced by Applicant and the Staff, primarily in response to the Licensing Board's questions.

adduce additional evidence concerning the integrity of the reactor vessel for Unit No. 2 in the pending proceeding since extensive and adequate evidence on this matter has already been presented. Applicant respectfully requests that the Atomic Safety and Licensing Appeal Board review Applicant's statement of position set forth in this brief and the evidence relating to this certified question adduced during the course of the proceeding which is referenced herein and rule in Applicant's favor with respect to the certified question referred to above.

II.

Because of the Extensive Requirements Imposed to Assure That the Reactor Vessel Will Not Fail, Unit No. 2 Need Not Be Designed Against the Consequences of Failure of the Reactor Vessel

The regulations of the Atomic Energy Commission do not require that a nuclear power reactor be designed against the anticipated consequences of a postulated rupture of the reactor vessel and the Atomic Energy Commission has never required that such a facility be so designed.[3/] The General Design Criteria for Nuclear Power

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3/ The fact that the design of Unit No. 2 and certain other facilities includes some protection against the mechanical effects of a reactor vessel failure does not alter the basic position of the Staff on this matter. See Responses of the DRL to the Questions of the ASLB at the Hearing Session dated March 24, 1971, Question (Tr. 683-684) (introduced into evidence Tr. 917).

Plants set forth in Appendix A to Part 50 of the regulations of the Commission establish the parameters of "design, fabrication, construction, testing and performance requirements for structures, systems, and components important to safety; that is, structures, systems and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.[4/] By setting forth the design requirements for both normal operation and postulated accident conditions, Appendix A describes not only the requisite design for particular components of nuclear power facilities but also sets forth the consequences against which the facility must be designed.

Appendix A requires that a nuclear facility be designed against the consequences of a "loss-of-coolant accident." Such accidents are defined as "those postulated accidents that result from the loss of coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the

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4/ 10 CFR, Part 50, App. A, "Introduction."

double-ended rupture of the largest pipe of the reactor coolant system."[5/] This definition says nothing about a postulated rupture of the reactor vessel.

The regulations of the Commission define "reactor coolant pressure boundary" as "those pressure-containing components of boiling and pressurized water-cooled nuclear power reactors, such as pressure vessels, piping, pumps, and valves . . . ."[6/] The definition of loss-of-coolant accidents, however, specifically limits the breaks of concern in the reactor coolant pressure boundary to those "up to and including" breaks equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system. The Commission does not require an Applicant to analyze the consequences of a postulated reactor vessel rupture in order to satisfy the Commission's regulations regarding a "loss-of-coolant accident." The interpretation that an agency places upon its regulations in practice is a clear indication of the meaning of the regulation.[7/]

5/ 10 CFR 50, App. A, Definitions and Explanations.

6/ 10 CFR § 50.2(v).

7/ Bowles v. Seminole Rock & Sand Co., 325 U.S. 410, 413-14 (1945); Udall v. Tallman, 380 U.S. 1, 16-17 (1965); Thorpe v. Housing Auth., 393 U.S. 268, 276 (1969).

If the Commission intended to include a reactor vessel rupture within the definition of a loss-of-coolant accident the limiting phrase referred to above would not only be unnecessary but also incorrect. When it promulgated Appendix A the Commission had considered the theoretical possibility that a reactor vessel rupture could occur. The regulation is not a model of clarity, but to assume such an accident is encompassed by a definition which is purposely framed in terms of a double-ended rupture of the largest reactor coolant pipe is patently illogical.

The footnote contained within the definition of loss-of-coolant accidents in no way broadens the definition. The footnote simply states that "[f]urther details relating to the type, size and orientation of postulated breaks in specific components of the reactor coolant pressure boundary are under development." This statement is in accord with the statements contained in the Introduction to Appendix A that "some of the definitions need further amplification" and that "[i]t is expected that the criteria will be augmented and changed from time to time as important new requirements for these and other features are developed."

The footnote simply illustrates that the intent of the definition was not to include all breaks of all components in the reactor coolant pressure boundary but rather to set forth the ultimate limits of concern and that further clarification within those bounds is to be developed.

The omission of a reactor vessel rupture from the definition of a loss-of-coolant accident, however, does not compromise in any manner the health and safety of the public. To the contrary the fact that the regulations set forth extensive Criteria[8/] to assure the integrity of the reactor vessel buttresses the position of the Applicant,

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8/ See, e.g., 10 CFR Part 50, App. A, Criteria 13, 14, 15, 30, 31 and 32. See also Answers of Applicant to Questions Raised by ASLB on March 24, 1971, Pt. II, dated July 6, 1971, Question 3, page 1 (follows Tr. 888); Applicant's Responses to Round Two Questions Submitted by CCPE, Pt. I, dated March 29, 1971, Question H-1 (CCPE Exh. H).

the Staff and the Advisory Committee on Reactor Safeguards that Unit No. 2 need not be designed against the consequences of a rupture of the reactor vessel.[9/]

By promulgating and proceeding in accordance with Appendix A the Commission has delineated its position that a nuclear facility designed, fabricated, erected and tested in accordance with the Criteria relating to the integrity of the reactor coolant pressure boundary can be operated without undue risk to the health and safety

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9/ It should be noted that both the ACRS and the Staff have concluded that Unit No. 2 can be operated at power levels up to 2758 Mwt without undue risk to the health and safety of the public. See Safety Evaluation by the DRL, dated Nov. 16, 1970, pp. 3, 20-29, 60, 88-91, 111 (follows Tr. 405). See 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, dated Jan. 28, 1972, Finding Nos. 36-42 and references contained therein; 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; Response of Staff to Proposed Findings of Fact of CCPE, dated March 10, 1972, Response 9.(a.)(b.)(c.)(d.) through 9.(i.); Letter to the Board from Mr. Karman dated April 13, 1972.

of the public.[10/] In addition, by requiring that an applicant follow specific Criteria relating to the integrity of the reactor coolant pressure boundary, including the reactor vessel, the Commission has determined that an applicant

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10/ The mandate of the regulations cannot be obfuscated on the basis of a letter dated November 24, 1965 (prior to the promulgation of Appendix A) from the Advisory Committee on Reactor Safeguards ("the ACRS") to the Commission recommending that "means be developed to ameliorate the consequences of a major pressure vessel rupture." On July 11, 1967, when the proposed amendments were published in the Federal Register (32 Fed. Reg. 10,213), the Commission stated that in developing these amendments it had taken into consideration the comments and suggestions from the ACRS, members of industry and the public. Preliminary proposed criteria for the design of nuclear power plants reflecting recommendations made by a seven-member Regulatory Review Panel appointed by the Commission were discussed with the ACRS and were informally distributed for public comment in Commission Press Release H-252 dated November 22, 1965. Appendix A, published in the Federal Register on February 20, 1971, reflected the additional comments and suggestions received in response to the notice of proposed rulemaking and subsequent developments in nuclear technology and the licensing process. Again, on July 7, 1971, the Commission published amendments correcting and clarifying the intent of the Commission with respect to several of the criteria in Appendix A to Part 50. These developments subsequent to the 1965 ACRS letter adequately took into consideration the recommendations and concerns of the ACRS. See also Responses of the DRL to the Questions of the ASLB at the Hearing Session dated Jan. 19, 1971, Question (Tr. 491) pp. 4-6 (introduced into evidence Tr. 917); Responses of the DRL to the Questions of the ASLB at the Hearing Session dated March 24, 1971, Question (690-691) (introduced into evidence Tr. 917); Answers of Applicant to Questions Raised by ASLB on March 24, 1971, Pt. II, dated July 6, 1971, Question 11, pp. 3-9 (follows Tr. 888).

need not design a nuclear power reactor against the consequences of a rupture of the reactor vessel.

III.
Additional Evidence Concerning the Integrity of the Reactor Vessel Should Not Be Adduced in the Further Proceedings for a Full-Term, Full-Power Operating License for Unit No. 2

Part I above demonstrates that the regulations of the Commission do not require that Unit No. 2 be designed against the consequences of a postulated failure of the reactor vessel, because of the extensive requirements imposed to assure that the vessel will not fail. The regulations do require, however, that the reactor vessel be designed, fabricated, erected and tested in accordance with the General Design Criteria. Applicant and Staff have introduced sufficient evidence in the proceeding to demonstrate that the applicable requirements of the General Design Criteria of 10 CFR 50.55a have been satisfied with respect to Unit No. 2.[11/] In addition, in response to the allegations of the intervenor and the inquiries of the Board, the Applicant and the Staff have introduced overwhelming and

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11/ Final Facility Description and Safety Analysis Report ("FSAR"), Section 4.1; Safety Evaluation By the DRL, dated Nov. 16, 1970 (follows Tr. 405).

uncontradicted evidence to demonstrate specifically that the reactor vessel for Unit No. 2 will not fail.[12/] Such testimony was offered by a panel of seven eminently qualified expert witnesses in the field of pressure vessel technology. Applicant's evidence was probably the most extensive presentation on this subject ever made before an Atomic Safety and Licensing Board regarding a nuclear power reactor.

Further evidence concerning the integrity of the reactor vessel for Unit No. 2 is not required in the pending proceeding for a full-term, full-power operating license for Unit No. 2. Applicant and the Staff have already provided assurance that the regulations of the Commission have been satisfied, both with respect to the initial design,

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12/ Additional Testimony of Applicant Concerning Reactor Vessel Integrity, dated Sept. 17, 1971 ("Reactor Vessel 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 Oct. 26, 1971 ("Staff Report on Pressure Vessel") (introduced into evidence Tr. 2715).

fabrication, erection and testing of the reactor vessel[13/] and with respect to its operating conditions,[14/] and that, therefore, the health and safety of the public during the intended period of operation of Unit No. 2 will be adequately protected.

As set forth in the testimony already received in this proceeding, Applicant has updated its inspection program

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13/ Reactor Vessel Testimony, pp. 2-1 through 2-14, 3-1 through 3-5, Apps. A and D (follows Tr. 1932); Staff Report on Pressure Vessel, pp. 4-24 (introduced into evidence Tr. 2715); FSAR, Sections 4.2.2, 4.2.5, 4.3.1, 4.4.1, 4.4.3, App. B, Questions 4.6, 4.8.1, 4.8.2, 4.8.4, 4.8.6, 4.8.7, 4.10; Tr. 2032-33, 2035-36, 2039-46, 2049-50, 3934-40, 3943-46, 3948-50, 3952, 3955-56, 3963-71, 3976, 3989-91; Applicant's Responses to Round Two Questions Submitted by CCPE, Part I, dated March 29, 1971, Questions H-6, pp. 1-3, H-9, H-11, H-13, H-14, and H-15 (CCPE Exh. H); Responses of the DRL to the Questions of the ASLB at the Hearing Session dated March 24, 1971, Question (Tr. 682-683) (follows Tr. 917); Answers of Applicant to Questions Raised by ASLB on March 24, 1971, Pt. I, dated May 8, 1971, Question 2, pp. 2-6 (follows Tr. 728). For further discussion see Applicant's Proposed Findings Nos. 36-42 attached as Appendix hereto.

14/ Summary of Application, Applicant's Exhibit No. 1C (introduced into evidence Tr. 377), pp. 13, 57-59 and references contained therein; Reactor Vessel Testimony, pp. 4.1 through 4.4, Section 5, Apps. 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, Questions 4.8.5; Tech. Spec. Nos. 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. 2037-39, 2046-56, 3940-43, 3948-50, 3953-63, 3969-76. For further discussion see Appendix attached hereto.

to include the inservice inspection requirements of Section XI of the ASME Code.[15/] This program provides verification of the initial quality of the primary system by base line inspection as well as additional assurance that the integrity of the primary coolant system will be maintained during full operating life of the facility.[16/]

15/ 10 CFR 50.55a (g) provides that for construction permits issued on or after January 1, 1971, systems and components shall meet the requirements of Section XI of the ASME Code. A construction permit for Unit No. 2 was issued on October 17, 1966. Prior to the amendments to the regulations of the Commission incorporating the inservice inspection requirements of Section XI, Unit No. 2 had been designed to facilitate inservice inspections from the reactor interior. In addition, pre-service mapping had been included in Applicant's quality control program. See Answers of Applicant to Questions Raised by ASLB on March 24, 1971, Pt. I, dated May 8, 1971, Question 2, pp. 5-6 (follows Tr. 728).

16/ Applicant's Responses to Round Two Questions Submitted by CCPE, Pt. I, dated March 29, 1971, Questions H-10, pp. 2-9 and H-12, pp. 1-2 (CCPE Exh. H); Answers of Applicant to Questions Raised by ASLB on Jan. 19, 1971, Pt. I, dated March 11, 1971, Questions 2, pp. 1-3 and 3, pp. 1-2 (follows Tr. 665); Answers of Applicant to Questions Raised by ASLB on March 24, 1971, Pt. I, dated May 8, 1971, Questions 2, pp. 5-6 and 11, p. 3 (follows Tr. 888); Answers of Applicant to Questions Raised by ASLB on May 13, 1971, dated July 6, 1971, Question 10, pp. 1-2 (follows Tr. 890); Responses of the DRL to the Questions of the ASLB at the Hearing Session Dated May 13, 1971, Questions 5 and 6 (follows Tr. 917).

Technical Specification No. 4.2[17/] sets forth the inservice inspection program developed by the Staff and the Applicant. Although the basic inspection interval defined in the Code and in the Technical Specifications is ten years, particular examinations will be completed or partially completed during this ten-year interval. "Those tests specified to be conducted within the ten-year interval are intended to provide an assessment of the general overall condition of the coolant system, to evaluate the vessel material in high service-factor areas and to determine the effect of significant neutron fluence on the physical properties of the materials."[18/] Those tests as set forth in the technical specification can be conducted with the equipment and methods available at this time.[19/] The technical

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17/ Appendix A to Proposed Facility Operating License - Technical Specifications and Bases, Supp. No. 1 to Staff's Exh. No. 1 (introduced into evidence Tr. 678).

18/ Responses of the DRL to the Questions of the ASLB at the Hearing Session Dated March 24, 1971, Question (Tr. 682-683) (follows Tr. 917).

19/ Answers of Applicant to Questions Raised by ASLB on March 24, 1971, Pt. I, Dated May 8, 1971, Question 2, pp. 6-8 (follows Tr. 728).

specifications also set forth additional volumetric examinations for which necessary equipment is now under development.[20/] In this regard Technical Specification No. 4.2 is consistent with the wording and intent of Section XI. Although the Code recognizes that appropriate equipment for such volumetric examinations must still be developed, it also reflects the philosophy that during the ten-year interval the development of the appropriate equipment can be completed or the program can be reassessed. The technical specifications for Unit No. 2 provide that after only five years of operation of Unit No. 2 the Staff will review the results of the tests conducted up to that time and will assess the situation regarding volumetric examinations. If, as a result of such an assessment the Staff believed that additional inspection requirements should be imposed, it could require changes to be made in the technical specifications. The Staff has testified that the time periods set forth by the Applicant for

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20/ Answers of Applicant to Questions Raised by ASLB on Jan. 19, 1971, Pt. I, Dated March 11, 1971, Question 2, p. 3 (follows Tr. 665); Answers of Applicant to Questions Raised by ASLB on March 24, 1971, Pt. I, Dated May 8, 1971, Question 2, pp. 8-10 (follows Tr. 728); Answer of Applicant to Questions Raised by ASLB on May 13, 1971, Dated July 6, 1971, Question 9, pp. 1-6 (follows Tr. 890).

inservice inspections are proper and that the entire program "will not only assure timely detection of any unanticipated structural degradation in the vessel, but also provide confidence that the probability of any flaw growing unknowingly during the service lifetime to a critical size and resulting in sudden failure is negligible."[21/] Of course, the decision whether "appropriate equipment" has been developed to perform certain inspections called for by the technical specifications is subject to approval by the Regulatory Staff.

IV.
Conclusion

The Applicant has established a program regarding the integrity of the reactor vessel which not only includes provisions which are not required by the regulations of the Commission but also provisions which surpass requisite regulations. By recognizing that remote, volumetric inspection devices must be developed for expeditious

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21/ Staff Report on Pressure Vessel, pp. 31-32 (introduced into evidence Tr. 2715).

examination of certain areas of the interior of the vessel, Applicant has approached the subject in a manner consistent with the regulations of the Commission and Section XI of the ASME Code. Applicant has even gone further by obligating itself to review its program with the Commission in five years and by pursuing actively the development of equipment necessary for these examinations.

Applicant requests that the Atomic Safety and Licensing Appeal Board conclude that the evidence presented in this proceeding adequately demonstrates the continuing assurance that the integrity of the reactor vessel will be maintained over its service lifetime.[22/] Applicant further requests that the Appeal Board determine, in response to

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22/ The Licensing Board specifically concluded that " ... there is reasonable assurance that the reactor vessel will not fail during the testing and initial operation of Unit No. 2." (Page 15 of Initial Decision.)

the Board's certified question, that Unit No. 2 need not be designed against the consequences of failure of the reactor vessel and that it is unnecessary to adduce further evidence in this proceeding concerning the integrity of the Unit No. 2 reactor vessel.

Respectfully submitted,

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_____
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Dated: August 21, 1972

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

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

46/ Tr. 1933-1936.

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

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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

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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

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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).