Docket Files
(50-436)

JUL 2 1975

Docket No. 50-346
MS 24-12

Voss A. Moore, Assistant Director
for Light Water Reactors, Group 2
Division of Reactor Licensing

TOLEDO EDISON COMPANY DAVIS-BESSE NUCLEAR STATION UNIT 1 (OL) DOCKET NUMBER 50-346 OF MARCH 31, 1973

Plant Name: Davis-Besse Nuclear Station - Unit 1
Suppliers: Babcock and Wilcox; Bechtel
Licensing Stage: OL
Docket Number: 50-346; March 31, 1973
Project Branch and Manager: LWR 2-3; L. Engle
Requested Completion Date: June 30, 1975
Task: SER
Review Status: Complete

The information submitted by the applicant, including Amendment No. 29 has been reviewed by the Materials Performance Section, Materials Engineering Branch, Office of Nuclear Reactor Regulation. Our sections of the Safety Evaluation are enclosed.

The Technical Specifications have not been submitted. They will be reviewed at a later date.

The applicant also has not submitted the results of fracture toughness tests on the reactor vessel materials. On the basis of information available, we expected that they will be acceptable. We will require documentation of these test results prior to the ACRS meeting.

R. R. Maccary, Assistant Director
for Engineering
Division of Technical Review
Office of Nuclear Reactor Regulation

For enclosure and cc's
see attached sheet

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Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

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David-Besse 1

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TOLEDO EDISON COMPANY
DAVIS-BESSE NUCLEAR STATION UNIT 1 (OL)
DOCKET NUMBER 50-346
SAFETY EVALUATION

MATERIALS ENGINEERING BRANCH
MATERIALS PERFORMANCE SECTION

REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Integrity of Reactor Coolant Pressure Boundary

Fracture Toughness

1. Compliance with Code Requirements

We have reviewed the materials selection, toughness requirements, and extent of materials testing accomplished by the applicant to provide assurance that the ferritic materials used for pressure retaining components of the reactor coolant boundary will have adequate toughness under test, normal operation, and transient conditions. All ferritic materials met the toughness requirements of the ASME Boiler and Pressure Vessel Code, Section III, 1968 Edition and Addenda through Summer 1968. Although all the tests required by Appendix G, 10 CFR Part 50 were not conducted, we conclude from the tests that were conducted that ferritic materials used for the reactor pressure vessel would meet the requirements of Appendix G.

The fracture toughness tests and procedures required by Section III of the ASME Code, and Appendix G, 10 CFR 50, for the reactor vessel, provide reasonable assurance that adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture can be established for the pressure-retaining components of the reactor coolant boundary.

2. Operating Limitations

The reactor will be operated in a manner that will minimize the possibility of rapidly propagating failure, in accordance with Appendix G to Section III of the ASME Boiler and Pressure Vessel Code and Appendix G, 10 CFR 50. Additional conservatism in the pressure-temperature limits used for heatup, cooldown, testing, and core operation will be provided because these will be determined assuming that the beltline region of the reactor vessel has already been irradiated.

The use of Appendix G of the Code as a guide in establishing safe operating limitations, using results of the fracture toughness tests performed in accordance with the Code and NRC Regulations, will ensure adequate safety margins during operating, testing, maintenance, and postulated accident conditions. Compliance with these Code provisions and NRC regulations constitute an acceptable basis for satisfying the requirements of NRC General Design Criterion 31, Appendix A of 10 CFR Part 50.

3. Reactor Vessel Materials Surveillance Program

The toughness properties of the reactor vessel beltline material will be monitored throughout service life with a material surveillance program. This program is described in Topical Report BAW-10100, "Reactor Vessel Material Surveillance Program," which we have reviewed and found acceptable. The program meets the requirements of ASTM E 185-73 and Appendix H, 10 CFR 50 (July 17, 1973), as far as practical.

Changes in the fracture toughness of material in the reactor vessel beltline caused by exposure to neutron radiation will be assessed properly, and adequate safety margins against the possibility of vessel failure will be provided since the essential material surveillance requirements of ASTM E 185-73 and Appendix H, 10 CFR Part 50, are met. The surveillance program constitutes an acceptable basis for monitoring radiation induced changes in the fracture toughness of the reactor vessel material, and will satisfy the requirements of NRC General Design Criterion 31, Appendix A, of 10 CFR Part 50.

Pump Flywheel

The probability of a loss of pump flywheel integrity can be minimized by the use of suitable material, adequate design, and inservice inspection.

The applicant has stated that the integrity of the reactor coolant pump flywheel is provided by having it designed and tested for a 125% overspeed condition. In addition, a 100% ultrasonic volumetric inspection of the flywheel plate before machining using ASME Section III acceptance criteria, was performed. Inservice inspections of the flywheel will be performed in accordance with the provisions of NRC Regulatory Guide 1.14.

We conclude that the provisions for material selection and flywheel design, and inservice inspections in accordance with Regulatory Guide 1.14 ensure adequate flywheel integrity and constitutes an acceptable basis for satisfying the requirements of NRC General Design Criterion 4, Appendix A, 10 CFR Part 50.

REACTOR VESSEL AND APPURTENANCES

Reactor Vessel Integrity

We have reviewed all factors contributing to the structural integrity of the reactor vessel and we conclude there are no special considerations that make it necessary to consider potential vessel failure for the Davis-Besse Plant, Unit 1.

The bases for our conclusion are that the design, material, fabrication, inspection, and quality assurance requirements conformed to the rules of the ASME Boiler and Pressure Vessel Code, Section III, 1968 Edition, all Addenda through Summer 1968, and all applicable Code Cases.

The fracture toughness requirements of the ASME Code, Section III, 1968 Edition and the 1968 Summer Addenda have been met. Also, operating limitations on temperature and pressure are established for this plant in accordance with Appendix G, "Protection Against Nonductile Failure," of the 1972 Summer Addenda of the ASME Boiler and Pressure Vessel Code, Section III, and Appendix G, 10 CFR 50.

The integrity of the reactor vessel is assured because the vessel:

1. Was designed and fabricated to the high standards of quality required by the ASME Boiler and Pressure Vessel Code and pertinent Code Cases listed above.

2. Was made from materials of controlled and demonstrated high quality.

3. Was extensively inspected and tested 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 to demonstrate that the high initial quality of the reactor vessel has not deteriorated significantly under the service conditions.

Inservice Inspection Program

To ensure that no deleterious defects develop during service, selected welds and weld heat-affected zones will be inspected periodically. The applicant has stated that the design of the reactor coolant system incorporates provisions for access for inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, and that a remote inspection tool has been developed and will be used to inspect those areas of the reactor vessel not readily accessible to inspection personnel.

The conduct of periodic inspections and hydrostatic testing of pressure-retaining components in the reactor coolant pressure boundary in accordance with the requirements of ASME Section XI Code provides reasonable assurance that evidence of structural degradation or loss of leaktight-integrity occurring during service will be detected in time to permit corrective action before the safety function of a component is compromised. Compliance with the inservice inspections required by this Code constitutes an acceptable basis for satisfying the requirements of NRC General Design Criterion 32, Appendix A of 10 CFR Part 50.

RCPB Leakage Detection System

Coolant leakage within the containment may be an indication of a small through-wall flaw in the reactor coolant pressure boundary.

The leakage detection system proposed for leakage to the containment includes diverse leak detection methods, has sufficient sensitivity to measure small leaks and can identify the leakage source to the extent practical. The major components of the system are the containment vessel sump and radiogas and air particulate radioactivity monitors. Intersystem leakage will be detected by abnormal readings from radiactivity monitors in the secondary system. The applicant has complied with the requirements of Regulatory Guide 1.45, except that systems are not specifically seismically qualified and all systems do not have a control room readout. Both of these requirements were established subsequent to the SAR application date. The leakage detection systems proposed to detect leakage from components and piping of the reactor coolant pressure boundary are generally in accordance with NRC Regulatory Guide 1.45 and provide reasonable assurance that any structural degradation resulting in leakage during service will be detected in time to permit corrective actions. This degree of conformance with the recommendations of NRC Regulatory Guide 1.45 constitutes an acceptable basis for satisfying the requirements of NRC General Design Criterion 30, Appendix A of 10 CFR Part 50.

COMPONENT AND SUBSYSTEM DESIGN

Steam Generator Tube Integrity

We have evaluated the factors that affect the integrity of the steam generator tubes for Davis-Besse 1. We conclude that all reasonable measures have been taken to ensure that the tubing will not be subjected to conditions that will cause deleterious wastage or cracking. Our conclusion is based on the following:

1. There have been no instances of tube degradation in steam generators of the once through design that will be used in the Davis-Besse 1 plant.

2. The secondary water chemistry control used will be all volatile, thereby minimizing the probability of deleterious local high concentrations of caustic or phosphate on the tubing.

3. To further control impurities in the secondary water to very low levels Davis-Besse 1 will use full flow demineralization of the condensate.

4. We will require that periodic inservice inspections of the steam generator tubes be performed in accordance with the recommendations of the revision to Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes."

TECHNICAL SPECIFICATIONS

The Technical Specifications have not been submitted for review. They will be reviewed when available.

MATERIALS ENGINEERING BRANCH
MATERIALS PERFORMANCE SECTION

REFERENCES

General

Federal Register 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Plants," July 7, 1971.

Federal Register 10 CFR Part 50, § 50.55a, "NRC Codes and Standards Rules - Applicable Codes, Addenda, and Code Cases 'In Effect' for Components That Are Part of the Reactor Cooland Pressure Boundary," February 15, 1974.

"Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 1, October 1972.

List of NRC Approved Code Cases, February 22, 1973

Fracture Toughness

10 CFR 50 - Appendix G, "Fracture Toughness Requirements," July 17, 1973.

ASME Boiler and Pressure Vessel Code, Section III, 1972 Summer Addenda, including Appendix G, "Protection Against Non-Ductile Failure."

Materials Surveillance Programs

10 CFR 50 - Appendix H, "Reactor Vessel Material Surveillance Program Requirements," July 17, 1973.

ASTM Specification E 185-73, "Surveillance Tests on Structural Materials in Nuclear Reactors," Annual Book of ASTM Standards, Part 30, July 1973.

Pump Flywheels

1. NRC Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," October 27, 1971.

RCPB Leakage Detection Systems

1. NRC Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

Inservice Inspection Program

1. NRC Guideline Document, "Inservice Inspection Requirements for Nuclear Power Plants Constructed with Limited Accessibility for Inservice Inspections," January 31, 1969.

2. ASME Boiler and Pressure Vessel Code, Section XI, 1971 and 1974 Editions.

Reactor Vessel Integrity

1. ASME Boiler and Pressure Vessel Code, Section III, 1968 Edition plus Addenda through Summer 1968.

Steam Generator Tube Integrity

1. NRC Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," June 1974.