L. Lazo

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
JUL 18 1984

Docket No. 50-412

MEMORANDUM FOR: Thomas M. Novak, Assistant Director
for Licensing
Division of Licensing

FROM: William V. Johnston, Assistant Director
Materials, Chemical & Environmental Technology
Division of Engineering

SUBJECT: DUQUESNE LIGHT COMPANY, BEAVER VALLEY POWER STATION UNIT 2 (BVPS-2)

Plant Name: Beaver Valley Power Station Unit 2
Supplier: Westinghouse, Stone & Webster
Docket No.: 50-412
Responsible Branch: Licensing Branch #1
Project Manager: L. Lazo
Reviewer: B. J. Elliot
Despcription of Task: Safety Evaluation Report for Sections 5.3.1, 5.3.2, and 5.3.3
Review Status: Complete

Materials Applicant Section, Materials Engineering Branch, Division of Engineering, with the assistance of Idaho National Engineering Laboratory, has completed its review of the Final Safety Analysis Report for BVPS-2. Based on our review of information in FSAR Amendments thru No. 6 and in letter from E. J. Woolever to G. W. Knighton dated April 30, 1984, we have prepared our input to Safety Evaluation Report Sections 5.3.1, 5.3.2 and 5.3.3, (Attachment 1).

William V. Johnston

William V. Johnston, Assistant Director
Materials, Chemical & Environemental Technology
Division of Engineering

Attachments: As Stated

cc: See Next Page

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ATTACHMENT 1

DUQUESNE LIGHT COMPANY

BEAVER VALLEY POWER STATION UNIT NO. 2

DOCKET NO. 50-412

REACTOR COOLANT PRESSURE BOUNDARY FRACTURE TOUGHNESS

MATERIALS ENGINEERING BRANCH

MATERIALS APPLICATION SECTION

5.3.1 Reactor Vessel Material

The fracture toughness of ferritic reactor vessel and reactor coolant pressure boundary materials, and the materials surveillance program for the reactor vessel beltline have been reviewed. The acceptance criteria and references which are the basis for this evaluation are set forth in paragraph II.5, II.6, and II.7 (Appendices G and H, 10 CFR Part 50) of SRP Section 5.3.1 in NUREG-0800 Rev. 1 dated July 1981. A discussion of this review follows.

General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," Appendix A, 10 CFR Part 50, requires, in part, that the reactor coolant pressure boundary be designed with sufficient margin to ensure that, when stressed under operating, maintenance, and test conditions, the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized. General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary," Appendix A, 10 CFR Part 50, requires, in part, that the reactor coolant pressure boundary be designed to permit an appropriate material surveillance program for the reactor pressure vessel.

The fracture toughness requirements for the ferritic materials of the reactor coolant pressure boundary are defined in Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Requirements" of 10 CFR Part 50.

Compliance to Section 50.55(a), 10 CFR Part 50

The Edition and Addenda of the ASME Code that are applicable to the design and fabrication of the reactor vessel and reactor coolant pressure boundary (RCPB) components are specified in Section 50.55(a) of 10 CFR Part 50.

The ASME Code Edition and Addenda that are required depend upon the date the construction permit was issued. The Beaver Valley Power Station, Unit 2 (BVPS-2) construction permit was issued on May 3, 1974. Based upon the construction permit date, 10 CFR Part 50 Section 50.55(a) requires that ferritic materials used for the BVPS-2 reactor vessel be designed and constructed to editions that are

no earlier than the Winter 1971 Addenda to the 1971 ASME Code (hereafter Code) and that ferritic materials used in piping, pumps, and valves be constructed to editions that are no earlier than the Winter 1972 Addenda to the Code. The BVPS-2 ferritic materials meet all the above requirements.

Branch Technical Position MTEB 5-2 requires that the fracture toughness of ferritic RCPB materials must be assessed to the requirements of the Code, as augmented by Appendix G, 10 CFR Part 50. We have assessed the ferritic RCPB materials to the above requirements in our review of the applicant's compliance to Appendix G, 10 CFR Part 50.

Compliance with Appendix G, 10 CFR Part 50

Our evaluation of the BVPS-2 FSAR to determine the degree of compliance with the fracture toughness requirements of Appendix G, 10 CFR Part 50, indicates that the applicant has met all the requirements of this Appendix.

Compliance with Appendix H, 10 CFR Part 50

The materials surveillance program at BVPS-2 will be used to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region, resulting from exposure to neutron irradiation and the thermal environment as required by General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary." The BVPS-2 surveillance program, which must be in compliance with Appendix H, 10 CFR 50, requires fracture toughness data be obtained from material specimens that are representative of the limiting base, weld, and heat-affected zone materials in the beltline region. These data will permit the determination of the conditions under which the vessel can be operated with adequate margins of safety against fracture throughout its service life. As a result of the information supplied by the applicant, we have determined that the BVPS-2 surveillance program has met all the requirements of Appendix H, 10 CFR Part 50.

Conclusions for Compliance with Appendices G and H, 10 CFR Part 50

Based on our evaluation of compliance with Appendices G and H, 10 CFR Part 50, we conclude that the applicant has met all the fracture toughness requirements of these Appendices.

Appendix G, "Protection Against Non-Ductile Failures," Section III of the ASME Code, was used, together with the fracture toughness test results required by Appendices G and H, 10 CFR Part 50, to calculate the pressure-temperature limitations for the BVPS-2 reactor vessel.

The fracture toughness tests required by the ASME Code and by Appendix G of 10 CFR Part 50 provide reasonable assurance that adequate safety margins against the possibility of non-ductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the reactor coolant boundary. The use of Appendix G, Section III of the ASME Code, as a guide in establishing safe operating procedures, and use of the results of the fracture toughness tests performed in accordance with the ASME Code and NRC regulations, will provide adequate safety margins during operating, testing, maintenance, and anticipated transient conditions. Compliance with these Code provisions and NRC regulations constitutes an acceptable basis for satisfying the requirements for General Design Criterion 31.

The materials surveillance program, required by Appendix H, 10 CFR Part 50, will provide information on the effects of irradiation on material properties so that changes in the fracture toughness of the material in the BVPS-2 reactor vessel beltline can be properly assessed, and adequate safety margins against the possibility of vessel failure can be provided.

Compliance with Appendix H, 10 CFR Part 50 assures that the surveillance program will be capable of monitoring radiation induced changes in the fracture toughness of the reactor vessel material and satisfies the requirements of General Design Criterion 32.

5.3.2 Pressure Temperature Limits

The applicant's pressure temperature limits for operation of the reactor vessel have been reviewed. The acceptance criteria and list of references which are the basis for this evaluation are set forth in the Standard Review Plan (SRP) Section 5.3.2 of NUREG-0800 Rev. 1 dated July 1981. A discussion of this review follows.

Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," 10 CFR Part 50, describe the conditions that require pressure-temperature limits and provide the general bases for these limits. These appendices specifically require that pressure-temperature limits must provide safety margins at least as great as those recommended in the ASME Code, Section III, Appendix G, "Protection Against Non-Ductile Failures." Appendix G, 10 CFR Part 50, requires additional safety margins for the closure flange region materials and beltline materials whenever the reactor core is critical, except for low-level physics tests.

The following pressure-temperature limits imposed on the reactor coolant pressure boundary during operation and tests are reviewed to ensure that they provide adequate safety margins against nonductile behavior or rapidly propagating failure of ferritic components as required by General Design Criterion 31:

a. Preservice hydrostatic tests
b. Inservice leak and hydrostatic tests
c. Heatup and cooldown operations, and
d. Core operation.

The pressure-temperature limit curves, which were submitted for review are in compliance with the requirements of Appendix G, 10 CFR 50.

The pressure-temperature limits to be imposed on the reactor coolant system for all operating and testing conditions must have adequate safety margins against non-ductile or rapidly propagating failure, must be in conformance with established criteria, codes, and standards. The use of operating limits based on these criteria, as defined by applicable regulations, codes, and standards, will

provide reasonable assurance that non-ductile or rapidly propagating failure will not occur, and will constitute an acceptable basis for satisfying the applicable requirements of General Design Criterion 31.

5.3.3 Reactor Vessel Integrity

The staff has reviewed the FSAR sections related to the reactor vessel integrity of BVPS-2. Although most areas are reviewed separately in accordance with other review plans, reactor vessel integrity is of such importance that a special summary review of all factors relating to reactor vessel integrity is warranted. In this section, we have reviewed the fracture toughness of ferritic reactor vessel and reactor coolant pressure boundary materials, the pressure temperature limits for operation of the reactor vessel, and the materials surveillance program for the reactor vessel beltline. The acceptance criteria and references which are the basis for the evaluation are set forth in paragraphs II.2, II.6, and II.7 (Appendices G and H, 10 CFR Part 50) of Standard Review Plan (SRP) Section 5.3.3 in NUREG 0800 Rev. 1 dated July 1981.

We have reviewed the above factors contributing to the structural integrity of the BVPS-2 reactor vessel and conclude that the applicant has fully complied with the requirements of Appendices G and H, 10 CFR 50.

We have reviewed all factors contributing to the structural integrity of the reactor vessel and conclude there are no special considerations that make it necessary to consider potential reactor vessel failure for BVPS-2.