JUN 20 1975

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Docket Nos. 50-508/509
MS 24-12

R. C. DeYoung, Assistant Director
for Light Water Reactors, Group 1
Division of Reactor Licensing

WASHINGTON PUBLIC POWER SUPPLY SYSTEM, WPPSS NUCLEAR PROJECTS NO. 3 AND 5 (CP), DOCKET NUMBERS 50-508/509

Plant Name: WPPSS Nuclear Projects No. 3 and 5 (CP)
Supplier: Combustion Engineering
Licensing Stage: CP
Docket Numbers and Date: 50-508/509, August 2, 1974
Responsible Branch and Project Manager: LWR 1-3; P. D. O'Reilly
Requested Completion Date: June 20, 1975
Description of Task: Safety Evaluation
Review Status: Complete

Information submitted by the applicant in the PSAR through Amendment No. 14 has been reviewed by the Materials Performance Section of the Materials Engineering Branch, Office of Nuclear Reactor Regulation. Our sections of the Safety Evaluation are enclosed.

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

Enclosure:
As stated

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COMBUSTION ENGINEERING
WPPSS NUCLEAR PROJECTS NUMBERS 3 AND 5 (CP)
DOCKET NUMBERS 50-508/509
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 proposed 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. The ferritic materials are specified to meet the toughness requirements of the ASME Code, Section III, including Summer 1972 Addedna. In addition, materials for the reactor vessel are specified to meet the additional test requirements and acceptance criteria of Appendix G, 10 CFR 50.

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

2. Operating Limitations

The safety evaluation for operating limitations is presented in the Safety Evaluation for CESSAR.

3. Reactor Vessel Material Surveillance Program

The safety evaluation for reactor vessel material surveillance program is presented in the Safety Evaluation for CESSAR.

4. Reactor Vessel Closure Studs

The safety evaluation for reactor vessel closure studs, is presented in the Safety Evaluation for CESSAR.

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 direct or remote access for inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, and that suitable equipment will be developed to facilitate the remote inspection of 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.

To ensure that no deleterious defects develop during service in ASME Code Class 2 system components, selected welds and weld heat-affected zones will be inspected prior to reactor startup and periodically throughout the life of the plant. In addition Code Class 2 systems and Code Class 3 systems will receive visual inspections while the systems are pressurized in order to detect leakage, signs of mechanical or structural distress, and corrosion.

Examples of Code Class 2 systems are: (1) Residual heat removal systems, (2) Portions of chemical and volume control systems, and (3) Engineered safety features not part of Code Class 1 systems. Examples of Code Class 3 systems are: (1) Component cooling water systems and (2) Portions of radwaste systems. All of these systems transport fluids. The applicant has stated that the design of Code Class 2 systems meets the requirements of ASME Section XI. Also, the Code Class 2 systems and Code Class 3 systems are in conformance with the recommendations of Regulatory Guide 1.51. Compliance with the inservice inspections required by this Code and Regulatory Guide constitutes an acceptable bases for satisfying NRC General Design Criterion 36, 39, 42, and 45, Appendix A of 10 CFR Part 50.

Pump Flywheel

The safety evaluation for pump flywhel is presented in the Safety Evaluation for CESSAR.

RCPB Leakage Detection System

Coolant leakage within the primary 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 will include diverse leak detection methods, will have sufficient senstivity to measure small leaks, will identify the leakage source to the extent practical, and will be provided with suitable control room alarms and readouts. The leakage detection systems proposed to detect leakage from components and piping of the reactor coolant pressure boundary follow the recommendations of NRC Regulatory Guide 1.45 as far as practical and thus provide reasonable assurance that any structural degradation resulting in leakage during service will be detected in time to permit corrective actions. This constitutes an acceptable basis for satisfying the requirements of NRC General Design Criterion 30, Appendix A of 10 CFR Part 50.

Reactor Vessel 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 WPPSS-Nuclear Projects 3 and 5.

The bases for our conclusion are that the design, material, fabrication, inspection, and quality assurance requirements for the reactor vessels of WPPSS-Nuclear Projects 3 and 5 will conform to the rules of the ASME Code, Section III, 1971 Edition, and the 1972 Summer Addenda. Also, operating limitations on temperature and pressure will be established for this plant in accordance with Appendix G, "Protection Against Non-Ductile 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:

a. Will be designed and fabricated to the high standards of quality required by the ASME Code Section III and pertinent Code Cases listed above.

b. Will be made from materials of controlled and demonstrated high quality.

c. Will be inspected and tested to provide substantial assurance that the vessel will not fail because of material or fabrication deficiencies.

d. 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.

e. 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.

f. May be annealed to restore the material toughness properties if this becomes necessary.

COMPONENT AND SUBSYSTEM DESIGN

Steam Generator Tube Integrity

We have evaluated the factors that affect the integrity of the steam generator tubes for WPPSS Units 3 and 5. We conclude that 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. The steam generators will be of advanced design with improved secondary water flow characteristics. This will provide more tolerance for occasional lack of control of the secondary water chemistry.

2. All volatile treatment is planned for secondary water chemistry control, thereby minimizing the probability of deleterious local high concentrations of caustic or phosphate on the tubing.

3. The condenser will be made with Ni-Cr-Fe alloy tubing, thus minimizing the probability of condenser leakage contributing to contamination of the secondary water.

4. We will require inservice inspections that reflect the criteria of Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes."

COMBUSTION ENGINEERING, INCORPORATED
COMBUSTION ENGINEERING STANDARD SAFETY ANALYSIS REPORT (CESSAR)
DOCKET NO. 50-470
SAFETY EVALUATION

MATERIALS ENGINEERING BRANCH, L
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 proposed 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. The ferritic materials are specified to meet the toughness requirements of the ASME Code, Section III, including Summer 1972 Addenda. In addition, materials for the reactor vessel are specified to meet the additional test requirements and acceptance criteria of Appendix G, 10 CFR 50.

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

2. Operating Limitations

The reactor will be operated in accordance with the ASME Code, Section III, including Summer 1972 Addenda, and Appendix G, 10 CFR 50. This will minimize the possibility of failure due to a rapidly propagating crack. Additional conservatism in the pressure-temperature limits used for heatup, cooldown, testing, and core operation will be provided because the pressure-temperature limits 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 AEC Regulations, will ensure adequate safety margins during operating, testing, maintenance, and postulated accident conditions. Compliance with these Code provisions and AEC regulations constitute an acceptable basis for satisfying the requirements of AEC General Design Criterion 31, Appendix A of 10 CFR Part 50.

3. Reactor Vessel Material Surveillance Program

The toughness properties of the reactor vessel beltline material will be monitored throughout service life with a material surveillance program that will meet the requirements of ASTM Specification E 185-73. This program also complies with Appendix H, 10 CFR Part 50 except that specimen holders are attached to the vessel cladding. Combustion Engineering has submitted a Topical Report CENPD-155P, "CE Procedure for Design, Fabrication, Installation and Inspection of Surveillance Specimen Holder Assemblies." We have evaluated this report and conclude that their method of attaching capsule holders to the vessel clad is acceptable and results in no degradation of the vessel base material.

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 can be provided if the material requirements of the above documents are met. Compliance with these documents will ensure that the surveillance program consistutes an acceptable basis for monitoring radiation induced changes in the fracture toughness of the reactor vessel material, and will satisfy the requirements of AEC General Design Criterion 31, Appendix A of 10 CFR Part 50.

Although the use of controlled composition material for the reactor vessel beltline will minimize the possibility that radiation will cause serious degradation of the toughness properties, the applicant has stated that should results of tests indicate that the toughness is not adequate, the reactor vessel can be annealed to restore the toughness to acceptable levels.

4. Reactor Vessel Closure Studs

The reactor vessel closure studs will be procured and initially inspected in conformance with the mechanical and toughness property requirements and inspection requirements recommended in Regulatory Guide 1.65, "Materials and Inspections for Reactor Vessel Closure Studs." However the inservice inspection requirements are not satisfactory. Inservice inspection requirements of reactor vessel studs will be covered in Applicant's SAR and will be reviewed on a case-by-case basis.

5. 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 suitable equipment will be developed to facilitate the remote inspection of 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 AEC General Design Criterion 32, Appendix A of 10 CFR Part 50.

To ensure that no deleterious defects develop during service in ASME Code Class 2 system components, selected welds and weld heat-affected zones are inspected prior to reactor startup and periodically throughout the life of the plant. Code Class 2 systems and Code Class 3 systems receive visual inspections while the systems are pressurized in order to detect leakage, signs of mechanical or structural distress, and corrosion.

Examples of Code Class 2 systems are: (1) Residual heat removal systems, (2) Portions of chemical and volume control systems, and (3) Engineered safety features not part of Code Class 1 systems. Examples of Code Class 3 systems are: (1) Component cooling water systems and (2) Portions of radwaste systems. All of these systems transport fluids. The applicant has stated that the Code Class 2 systems meet the requirements of ASME Section XI. The Code Class 2 systems and Code Class 3 systems are in conformance with the recommendations of Regulatory Guide 1.51. Compliance with the inservice inspections required by this Code and Regulatory Guide constitutes an acceptable bases for satisfying AEC General Design Criteria 36, 39, 42, and 45, Appendix A of 10 CFR Part 50.

6. 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 compliance with the AEC Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity."

The use of suitable material, and adequate design and inservice inspection for the flywheels of reactor coolant pump motors as specified in the SAR provides reasonable assurance (a) that the structural integrity of the flywheels is adequate to withstand the forces imposed in the event of pump design overspeed transient without loss of function, and (b) that their integrity will be verified periodically in service to assure that the required level of soundness of the flywheel material is adequate to preclude failure. Compliance with the recommendations of AEC Regulatory Guide 1.14 constitutes an acceptable basis for satisfying the requirements of AEC General Design Criterion 4, Appendix A of 10 CFR Part 50.

7. RCPB Leakage Detection System

Coolant leakage within the primary 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 will include diverse leak detection methods, will have sufficient sensitivity to measure small leaks, will identify the leakage source to the extent practical, and will be provided with suitable control room alarms and readouts. The leakage detection systems proposed to detect leakage from components and piping of the reactor coolant pressure boundary are in accordance with AEC Regulatory Guide 1.45 insofar as it is applicable to CESSAR and thus provide reasonable assurance that any structural degradation resulting in leakage during service will be detected in time to permit corrective actions. Compliance with the recommendations of AEC Regulatory Guide 1.45 constitutes an acceptable basis for satisfying the requirements of AEC General Design Criterion 30, Appendix A of 10 CFR Part 50.

8. 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 CESSAR.

The bases for our conclusion are that the design, material, fabrication, inspection, and quality assurance requirements for the reactor vessels of Units 1 and 2 will conform to the rules of the ASME Code, Section III, 1971 Edition, and the 1972 Summer Addenda. Also, operating limitations on temperature and pressure will be established for this plant in accordance with Appendix G, "Protection Against Non-Ductile 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:

a. Will be designed and fabricated to the high standards of quality required by the ASME Code Section III and pertinent Code Cases listed above.

b. Will be made from materials of controlled and demonstrated high quality.

c. Will be inspected and tested to provide substantial assurance that the vessel will not fail because of material or fabrication deficiencies.

d. 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.

e. 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.

f. May be annealed to restore the material toughness properties if this becomes necessary.

MATERIALS PERFORMANCE SECTION, MATERIALS ENGINEERING BRANCH

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, "AEC Codes and Standard Rules - Applicable Codes, Addenda, and Code Cases "In Effect" for Components that are part of the Reactor Coolant Pressure Boundary," June 12, 1971.

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

Fracture Toughness

10 CFR 50 - Appendix G, "Fracture Toughness Requirements," June 1, 1973.

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

ASME Specification, SA-370-71b, "Methods and Definitions for Mechanical Testing of Steel Products," ASME Boiler and Pressure Vessel Code, Section II, Part A - Ferrous, 1971 Edition, Summer and Winter, 1972 Addenda.

ASTM Specification E-208-69, "Standard Method for Conducting Dropweight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels," Annual Book of ASTM Standards, Part 31, July 1973.

ASTM Specification E 23-72, "Notched Bar Impact Testing of Metallic Materials," Annual Book of ASTM Standards, Part 31, July 1973.

Material Surveillance Programs

10 CFR 50 - Appendix H. "Reactor Vessel Material Surveillance Program Requirements," June 1, 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

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

RCPB Leakage Detection Systems

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

Inservice Inspection Program

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

ASME Boiler and Pressure Vessel Code, Section XI, 1971 Edition, including Winter 1971, Summer 1971, Winter 1972, and Summer 1973 Addenda.

Regulatory Guide 1.51, "Inservice Inspection of ASME, Class 2 and 3 Nuclear Power Plant Components," May 1973.

Reactor Vessel Integrity

ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition plus Addenda through Winter 1972.

ASME Boiler and Pressure Vessel Code, Section XI, 1971 Edition plus Addenda through Winter 1972.