FLORIDA POWER CORPORATION
CRYSTAL RIVER UNIT NO. 3 (OL)
DOCKET NUMBER 50-302
SAFETY EVALUATION REPORT UPDATE

MATERIALS ENGINEERING BRANCH

REACTOR

Reactor Vessel Internals

General Material Considerations

We have reviewed the selection of materials for the reactor vessel internals required for reactor shutdown and components relied upon for adequate core cooling. All materials are compatible with the reactor coolant, and have performed satisfactorily in similar applications. Undue susceptibility to intergranular stress corrosion cracking has been prevented by avoiding the use of sensitized stainless steel according to the methods recommended in Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

The use of materials proven to be satisfactory by actual service experience and avoidance of sensitization by the methods recommended in Regulatory Guide 1.44, provides reasonable assurance that the reactor vessel internals will not be susceptible to failure by corrosion or stress corrosion cracking.

The applicant has described the measures that were taken to ensure that deleterious hot cracking of austenitic stainless steel welds was prevented. All weld filler metal was of selected composition, and welding processes were controlled to produce welds with adequate delta ferrite, in

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conformance with the recommendation in Regulatory Guide 1.31, "Control of Stainless Steel Welding." Following these recommendations provides reasonable assurance that no deleterious hot cracking will be present that could contribute to loss of integrity or functional capability.

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 performed by the applicant to provide assurance that the ferritic materials used for pressure retaining components of the reactor coolant boundary possess adequate toughness under test, normal operation, and transient conditions. The ferritic materials, not including piping, were ordered and tested in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III (1965 Edition and with Addenda through Summer 1967). Piping met the requirements of USAS Standard B31.7, dated February 1968, including the Errata dated June 1968. The construction permit of the Crystal River Unit No. 3 was issued on September 25, 1968.

The accepted guidelines for the fracture toughness requirements for the ferritic materials of the reactor pressure retaining components were published in Appendices G and H, 10 CFR Part 50, in the Federal Register on July 17, 1973. Appendix G, "Fracture Toughness Requirements," references Appendix G, "Protection Against Non-ductile Failure," of the ASME Boiler and Pressure Vessel Code, 1971 Edition, including Winter 1972 Addenda, which describes procedures for estimating the pressure-temperature test and operational limitations. Appendix H references ASTM E 185-73, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels," which describes the procedures for monitoring and evaluating irradiation induced changes in the beltline region of light water reactors receiving fluences greater than 10^{17} nvt (E >1 MeV).

Compliance with all the guidelines and provisions of Appendices G and H, 10 CFR Part 50, including the referenced documents, was not possible for the pressure retaining components of the Crystal River Unit No. 3 reactor since all of the required test materials were not available. Appendix G requires that test specimens for monitoring the reactor vessel beltline be taken from excess material and welds in the shell courses following completion of the production longitudinal weld joint. In addition, a minimum value of the upper shelf energy of 75 ft. lbs. is required. Justification for deviation from the guidelines and provisions has been presented for the Crystal River Unit No. 3 reactor in Topical Reports BAW-10046P, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G," and BAW-10100A, "Reactor Vessel Material Surveillance Program - Compliance with 10 CFR 50, Appendix H, for Oconee Class Reactors." The details of the Integrated Surveillance Program for Oconee Class Reactors and proprietary fracture toughness test information has been provided for our evaluation of the Crystal River Unit No. 3 reactor.

Topical Report BAW-10100A has been approved for reference in license application to describe the materials surveillance program for the reactor vessel. Topical Report BAW-10046P is in the process for review by the Materials Engineering Branch. Our evaluation of the Integrated Materials Surveillance test methods and procedures, materials properties, and fracture toughness characteristics, including estimates of the adjusted reference temperature, presented by the applicant for the Crystal River Unit No. 3 reactor shows compliance to the extent practical to Appendices G, "Fracture Toughness Requirements," and H, "Reactor Vessel Materials Surviellance Program Requirements," 10 CFR Part 50.

Compliance to the fracture toughness test methods and procedures to the extent practical required by Appendices G and H, 10 CFR Part 50. and Appendix G ASME Boiler and Pressure Vessel Code, 1971 Edition, including Winter 1972 Addenda, provide reasonable assurance that adequate safety margins exist 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, 10 CFR Part 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 operating limitations, based on fracture toughness tests conducted in accordance with Appendices G and H, 10 CFR Part 50, and ASTM E 185-73, will ensure adequate safety margins during operation, testing, maintenance, and postulated accident conditions. Compliance with the intent of these recommendations constitute an acceptable basis for satisfying the requirements of NRC General Design Criterion 31, Appendix A, 10 CFR Part 50.

3. Reactor Vessel Material Surveillance Program

The toughness properties of the reactor beltline materials will be monitored for irradiation and temperature effects throughout the service life with an integrated materials surveillance program, complying to the extent practical with Appendix H, 10 CFR Part 50. Appendix H requires in part that the surveillance specimens be selected from material adjacent to fracture toughness specimens. In lieu of the actual material, surveillance specimens have been selected on the basis of equivalent composition. The program is consistent with surveillance programs that have been found acceptable for other light water reactor plants. The copper and phosphorus contents of the reactor vessel beltline materials have been determined. The number of capsules and the specimen identity has been provided for the integrated surveillance program, and are conservatively based on the estimated shift in RT_{NDT} and the anticipated degradation in the upper shelf energy. In addition to the specimens required for compliance to intent of ASTM E 185-73, compact tension specimens are included for withdrawal in order to substantiate Charpy V-notch fracture toughness information as it is being developed.

The integrated surveillance program constitutes an acceptable basis for monitoring irradiation and temperature induced changes in the fracture toughness of the materials constituting the Crystal River Unit No. 3 reactor vessel beltline region, and will satisfy the requirements of NRC General Design Criterion 31, Appendix A, 10 CFR Part 50.

In the event the test results from the integrated surveillance program indicate inadequacy in material fracture toughness, the reactor vessel can be annealed to restore mechanical properties to acceptable values.

General Material Considerations

We have reviewed the materials of construction for the reactor coolant pressure boundary to ensure that the possibility of serious corrosion or stress corrosion is minimized. All materials used are compatible with the expected environment, as proven by extensive testing and satisfactory service performance. The applicant has shown that the possibility of intergranular stress corrosion in austenitic stainless steel used for components of the reactor coolant pressure boundary will be minimized because sensitization will be avoided, and adequate precautions will be taken to prevent contamination during manufacture, shipping, storage, and construction. The plans to avoid sensitization are in general conformance with Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," and include controls on compositions, heat treatments, welding processes, and cooling rates.

The use of materials with satisfactory service experience, and the high degree of conformance with Regulatory Guide 1.44, "Control of Sensitized Stainless Steel," provide reasonable assurance that austenitic stainless steel components will be compatible with the expected service environments, and the probability of loss of structural integrity is minimized.

Water Chemistry Control

Further protection against materials corrosion will be provided by control of the chemical environment. The composition of the reactor coolant will be controlled; and the proposed maximum contaminant levels, as well as the proposed pH, hydrogen overpressure, and boric acid concentrations, have been shown by tests and service experience to be adequate to protect against corrosion and stress corrosion problems.

We have evaluated the proposed requirements for the external insulation used on austenitic stainless steel components. Chloride and silicate contents will be controlled.

The possibility that serious corrosion or stress corrosion problems would occur in the unlikely event that ECCS or containment spray system operation is required will be minimized because the pH of the circulating coolant will be maintained above 9.0 by hydroxide additions.

The applicant has shown that the secondary water chemistry will be controlled to prevent stress corrosion of the steam generator tubing, and that the adequacy of the compositional limits used has been proven by satisfactory service experience.

The controls on chemical composition that will be imposed on the reactor coolant, secondary water, emergency core cooling water, and the use of low chloride external thermal insulation, provide reasonable assurance that the reactor coolant boundary materials will be adequately protected from conditions that would lead to loss of integrity from stress corrosion.

Control of Stainless Steel Welding

We have reviewed the controls proposed to prevent hot cracking (fissuring) of austenitic steel welds. These precuations include control of weld metal composition and welding processes to ensure adequate delta ferrite content in the weld metal. The proposed methods comply with Section III of the ASME Code, and are in essential conformance with Regulatory Guide 1.31, "Control of Stainless Steel Welding." The use of materials, processes, and test methods that are in accordance with these requirements and recommendations will provide reasonable assurance that loss of integrity of austenitic stainless steel welds caused by hot cracking during welding will not occur.

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 designed for a 125% overspeed condition while the maximum anticipated overspeed is 110% of normal speed. In the unlikely event of a 125% overspeed condition the maximum primary stress at the bore is approximately 70% of the yield strength, the flywheel was purchased prior to the requirements of Regulatory Guide 1.14 which permits 2/3 of yield at the design overspeed condition. In addition, a 100% ultrasonic volumetric inspection of the flywheel, using ASME Section III acceptance criteria, was performed.

Inservice inspections of the flywheel will be performed in accordance with the provisions of Regulatory Guide 1.14.

We conclude that the provisions for material selection and flywheel design, and the use of a Regulatory Guide 1.14 inservice inspection program ensure adequate flywheel integrity.

Inservice Inspection Program

To ensure that no deleterious defects develop during service, all welds 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 methods will be provided 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.

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 intersystem leakage is by means of radioactivity monitors and flow and level monitors. The leakage 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, will be provided with suitable control room alarms and readouts. The major components of the system are the containment airborne particulate and gas radioactivity monitors, and level and flow indication on the containment sump. Indirect indication of leakage can be obtained from the containment humidity, pressure, and temperature indicators.

The leakage detection systems will provide reasonable assurance that any structural degradation resulting in leakage during service will be detected in time to permit corrective action satisfying the requirements of NRC General Design Criterion 30, Appendix A of 10 CFR Part 50, and is thus acceptable.

ENGINEERED SAFETY FEATURES

Containment Design Evaluation

Containment Leakage Testing Program

The containment design includes the provisions and features planned which satisfy the testing requirements of Appendix J, 10 CFR Part 50. The design of the containment penetrations and isolation valves permits individual periodic leakage rate testing at the pressure specified in Appendix J, 10 CFR Part 50. Included are those penetrations that have resilient seals and expansion bellows, i.e., air locks, emergency hatches, refueling tube blind flanges, hot process line penetrations, and electrical penetrations.

The proposed reactor containment leakage testing program complies with the requirements of Appendix J, 10 CFR Part 50. Such compliance provides adequate assurance that containment leaktight integrity can be verified throughout service lifetime and that the leakage rates will be periodically checked during service on a timely basis to maintain such leakages within the specified limits.

Maintaining containment leakage rates within such limits provides reasonable assurance that, in the event of any radioactivity releases within the containment, the loss of the containment atmosphere through leak paths will not be in excess of acceptable limits specified for the site. Compliance with the requirements of Appendix J constitutes an acceptable basis for satisfying the requirements of NRC General Design Criteria 52, 53, and 54, Appendix A, 10 CFR Part 50.

CONTAINMENT HEAT REMOVAL AND ECCS SYSTEMS

General Material Considerations

(Compatibility with Coolant)

We have reviewed the materials selection proposed for the containment heat removal and ECCS systems, in conjunction with the expected chemistry of the cooling and containment spray system water. The applicant has shown that the use of sensitized stainless steel will be avoided, and that the pH of the containment spray and the circulating coolant will be controlled by sodium hydroxide additions. There are test data verifying that the proposed chemistry will not cause stress corrosion cracking of austenitic stainless steel under conditions that would be present during accident conditions.

We have concluded that the controls on material and cooling water chemistry proposed will provide assurance that the integrity of components of these systems will not be impaired by corrosion or stress corrosion.

(Control of SS Welding)

The applicant has stated that welding of austenitic stainless steel for components of these systems will be controlled to prevent deleterious hot cracking. The proposed control of weld metal composition and welding procedures are in general conformance with the recommendations of Regulatory Guide 1.31, "Control of Stainless Steel Welding," and will provide assurance that loss of function will not result from hot cracking of welds.

TECHNICAL SPECIFICATIONS

The pertinent sections of the Proof and Review Copy of the Crystal River Unit No. 3 Technical Specifications were reviewed by the Materials Engineering Branch. The sections reviewed comply with the Standard Technical Specifications for Babcock and Wilcox and are acceptable to the staff.

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, "NRC 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," Revision 2, September 1975.

Material Specifications

ASME Boiler and Pressure Vessel Code, Section III, 1974 Edition, including Summer Addenda 1974:

a. Paragraph NB-2121: Permitted Material Specifications

b. Paragraph NB-2122: Special Requirements Conflicting with Permitted Material Specifications

c. Specifications for Materials Listed in Tables 1-1.1, 1-1.2, and 1-1.3.

ASME Boiler and Pressure Code, Section II, 1974 Edition, including Summer Addenda 1974.

Chemistry of Reactor Coolant

NRC Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," May 1973.

Ferritic Steel

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

ASME Boiler and Pressure Vessel Code, Section III, through 1974 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 Code, Section II, Part A - Ferrous, 1974 Edition, including Summer, 1974 Addenda.

ASTM E 23-73, "Notched Bar Impact Testing of Metallic Materials," Annual Book of ASTM Standards, Part 31, July 1973.

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

NRC Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," May 1973.

NRC Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants," March 16, 1973.

Austenitic Stainless Steel

NRC Technical Position - MTEB 5-1, "Interim Position on Regulatory Guide 1.31, 'Control of Stainless Steel Welding.'"

NRC Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel," February 23, 1973.

NRC Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants," March 16, 1973.

NRC Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel," May 1973.

NRC Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," May 8, 1973.

NRC Regulatory Guide 1.66, "Nondestructive Examination of Tubular Products," October 1973.

NRC Regulatory Guide 1.71, "Welder Qualification for Limited Accessibility Areas," December 1973.

NRC Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," March 10, 1971.

ASTM A 262-70, Practice E, "Copper-Copper Sulfate-Sulfuric Acid Test for Detecting Intergranular Attack in Austenitic Stainless Steel," Annual Book of ASTM Standards, Part 3, April 1973.

ASTM A 393-63, "Recommended Practice for Conducting Acidified Copper Sulfate Test for Intergranular Attack in Austenitic Stainless Steel," Annual Book of ASTM Stadnards, Part 3, April 1973.

ANSI N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components for Nuclear Power Plants," Draft 2, Revision 0, November 15, 1973, American National Standards Institute.

<u>Inservice Inspection Program</u>

10 CFR Part 50, Appendix A, General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary."

10 CFR Part 50, Appendix A, General Design Criterion 36, "Inspection of Emergency Core Cooling System."

10 CFR Part 50, Appendix A, General Design Criterion 39, "Inspection of Containment Heat Removal System."

10 CFR Part 50, Appendix A, General Design Criterion 42, "Testing of Containment Atmosphere Cleanup System."

10 CFR Part 50, Appendix A, General Design Criterion 45, "Inspection of Cooling Water System."

ASME Boiler and Pressure Vessel Code, Section XI, 1974 Edition.

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