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

GEORGIA POWER COMPANY, ALVIN W. VOGTLE NUCLEAR PLANT, UNITS 1, 2, 3 & 4 (CP), DOCKET NOS. 50-424/425/426/427

Plant Name: Alvin W. Vogtle 1, 2, 3 & 4
Licensing Stage: CP
Docket Numbers: 50-424/425/426/427
Responsible Branch and Project Manager: LWR 2-2; L. P. Crocker
Requested Completion Date: December 21, 1973
Description of Response: Safety Evaluation
Review Status: Complete

The information submitted by the applicant, including Amendment No. 13, has been reviewed by the Materials Engineering Branch, Directorate of Licensing. Our sections of the Safety Evaluation are enclosed.

The applicant has submitted all of the required information.

Original signed by
R. R. Maccary

R. R. Maccary, Assistant Director
for Engineering
Directorate of Licensing

Enclosure:
Materials Engineering Branch Safety Evaluation for Alvin W. Vogtle 1, 2, 3 & 4

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GEORGIA POWER COMPANY
ALVIN W. VOGTLE NUCLEAR PLANT, UNITS 1, 2, 3 & 4 (CP)
DOCKET NOS. 50-424/425/426/427
SAFETY EVALUATION

MATERIALS ENGINEERING BRANCH, L

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 will be 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 will provide 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 will be taken to ensure that deleterious hot cracking of austenitic steel welds is prevented. All weld filler metal will be of selected composition, and welding processes will be controlled to produce welds with at least 5% delta ferrite, in conformance with the recommendation in Regulatory Guide 1.31, "Control of Stainless Steel Welding." Following these recommendations will provide 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

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. All ferritic materials will meet the toughness requirements of the ASME Boiler and Pressure Vessel Code, Section III (1971 Edition). In addition, materials for the reactor vessel will meet the additional testing and acceptance criteria of the Summer 1972 Addenda, and Apppendix 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 Part 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.

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, Summer 1972 Addenda, 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 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.

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 all the requirements of ASTM E 185-73 and Appendix H, 10 CFR 50 (July 17, 1973). The composition of reactor vessel beltline material, including welds, will be controlled to minimize the copper and phosphorus content, thus ensuring that the sensitivity to radiation damage will be low, but the number of capsules provided in the surveillance program is conservatively based on assuming high values of sensitivity. In addition to the specimen requirements of ASTM E 185-73, small fracture toughness specimens (1/2 T Compact Tension specimens) will be included in the program.

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 surveillance requirements of ASTM E 185-73 and Appendix H, 10 CFR Part 50 are met. Compliance with these documents will ensure that 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 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. We agree that the methods proposed are feasible and would be effective if needed.

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 corrosion problems 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, and conclude that it will be in conformance with Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

The possibility that serious corrosion or stress corrosion problems would occur in the unlikely event that ECCS or containment spray system operation is requi[illegible]d will be minimized because t[illegible] pH of the circulating coolant will be maintained above 7.0 by hydroxide additions.

The applicant has stated that the secondary water chemistry will be controlled using a balanced phosphate treatment 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 external thermal insulation in conformance with Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel," 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 precautions 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 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 flywheels is adequate to withstand the forces imposed in the event of design overspeed transients without loss of their 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.

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 tool 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.

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 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 major components of the system are the containment radiogas and atmosphere particulate radioactivity monitors, containment sump monitoring system, and the condensate measuring system. Indirect indication of leakage will be obtained from the containment humidity, pressure and temperature indicators. The applicant has complied with the requirements of Regulatory Guide 1.45, except that the airborne particulate radioactivity monitoring system has not been specifically designed to remain functional when subjected to the SSE, which requirement was subsequent to the PSAR. The leakage detection systems proposed to detect leakage from components and piping of the reactor coolant pressure boundary, in accordance with AEC Regulatory Guide 1.45, 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.

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 Alvin W. Vogtle Plant, Units 1, 2, 3, and 4.

The bases for our conclusion are that the design, material, fabrication, inspection and quality assurance requirements for Unit No. 1 will conform to the rules of the ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition, and all applicable Code Cases. The design of Unit No. 2 will conform to the rules of Section III and Addenda through Winter 1971 and applicable Code Cases. Units 3 and 4 will conform to the rules of Section III and all applicable Addenda and Codes in effect at the time the vessel is ordered.

The stringent fracture toughness requirements of the ASME Code, Section III, 1971 Edition, and the 1972 Summer Addenda will be met. 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:

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

3. Will be subjected to extensive inspection and testing 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.

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

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., airlocks, 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 to 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 AEC General Design Criteria 52, 53 and 54, Appendix A of 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 stated 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 hydroxide additions. He has also presented results of tests 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.

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.

General Materials Considerations

Material Specifications

ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition, plus Addenda through Summer 1973.

(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.

Chemistry of Reactor Coolant

AEC Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors," June 1973.

List of AEC Approved Code Cases, February 22, 1973.

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.

Austenitic Stainless Steel

AEC Regulatory Guide 1.31, "Control of Stainless Steel Welding," Revision 1, June 1973.

AEC Regulatory Guide 1.34, "Control of Electro-Slag Weld Properties," December 28, 1972.

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

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

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

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

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

Pump Flywheels

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

RCPB Leakage Detection Systems

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

Inservice Inspection Program

(1) AEC 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 Edition, including Winter 1971, Summer 1971, Winter 1972, and Summer 1973 Addenda.

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

Reactor Vessel Integrity

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

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

Containment Leakage Testing

(1) 10 CFR 50 - Appendix J, "Reactor Containment Leakage Testing for Water-Cooled Power Reactors," February 14, 1973.

(2) American National Standard ANSI N45.4-1972, "Leakage-Rate Testing of Containment Structures for Nuclear Reactors," March 16, 1972.