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NUCLEAR POWER

SYSTEMS DIVISION

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June 15, 1983

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, DC 20555

Attention: Mr. D.G. Eisenhut
Division of Licensing

Gentlemen:

SUBJECT: IN THE MATTER OF 238 NUCLEAR ISLAND
GENERAL ELECTRIC STANDARD SAFETY ANALYSIS REPORT
(GESSAR II) DOCKET NO. STN 50-447

QUESTION 410.24 AND AVOIDANCE OF INTERGRANULAR
STRESS CORROSION CRACKING

Attached please find:

1. Supplemental response to Question 410.24 of the Commission's August 25, 1982 information request; and

2. Summary of the measure taken in the GESSAR II design to avoid intergranular stress corrosion cracking.

The attached information will be included in Amendment No. 18 scheduled to be filed in early July 1983.

Sincerely,

Willa for

Glenn G. Sherwood, Manager
Nuclear Safety & Licensing Operation

E003
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Attachments

cc: F.J. Miraglia (w/o attachments)
D.C. Scaletti
C.O. Thomas (w/o attachments)
L.S. Gifford (w/o attachments)

SUPPLEMENTAL RESPONSE TO
QUESTION 410.24

19.3.9.24 QUESTION/RESPONSE 9.24 (410.24)

QUESTION 9.24

Verify that the information provided in Section 9.1.3 of your FSAR is based on the new high density spent fuel pool storage capacity. Provide additional information regarding the spent fuel decay heat load for the maximum, normal and abnormal heat loads as discussed in Items 1.d and 1.h of the review procedures in Section 9.1.3 of the SRP. (9.1.3)

RESPONSE 9.24

Paragraph 9.1.3.1.2(4) for the Power Generation Design Basis states that the heat load is the sum of (1) the 37 percent core batch just removed at the last 18-month equilibrium fuel cycle, with 4-year exposure, and (2) the 37 percent core batch from the previous refueling outage. Therefore, the heat load is a function of two 37 percent batches, which means that the entire heat capacity of the fuel storage pool does not enter the design. The fresh core supplies about 90 percent of the heat load and the aged core fraction supplies the other 10 percent of the design load. The density of the fuel racks would change the heat load calculation only if all of the potential batches stored within the pool were used toward the total design value. Even under these conditions, the design value would be only slightly affected.

Paragraph 9.1.3.2 describes that the above design core load for heat capacity is based upon maintaining 125°F in the pool. This is the system design maximum load and temperature combination. However, if conditions exist as described in Paragraph 9.1.3.3, wherein up to a full reactor core is

19.3.9.24 QUESTION/RESPONSE 9.24 (410.24) (Continued)

placed into the pool, instead of the 37 percent batch, the pool may go to 150°F. But adding the RHR cooling capacity will keep the temperature at a maximum of 125°F.

Item 1.h(ii) states that the normal maximum spent fuel heat load is set as one refueling load at equilibrium conditions after 150 hours of decay, with one refueling load after 1 year of decay, and 140°F pool temperature. The GESSAR II design basis is more conservative in that the refueling load is assumed at 112 hours of decay and the maximum pool temperature is set at 125°F. The shorter fresh batch decay time adds to the total heat load sum of the two batches.

Item 1.h(iii) states that the Spent Fuel Pool Cooling System will have capacity for a full core at equilibrium and one refueling load, at 36 days, for a total of 1-1/3 core fraction. Item 1.h(iv) further adds 1/3 core for pool capacity over 1-1/3 batches. If RHR cooling capacity is included in the Spent Fuel Pool Cooling System, then the cooling capacity is more than adequate to meet these criteria. The two fuel pool heat exchangers cover only the normal maximum, reflecting Item 1.h(ii), and the RHR covers any additional load while the reactor is open.

The failure of one of the two active pumps or heat exchangers will reduce the capacity of the fuel pool cooling system. The amount of cooling required by the pool is a function both of the amount of core placed in the pool at the last refueling and of the time for decay of that core fraction. If the decay heat exceeds the removal capacity, the RHR system shall be employed to maintain fuel pool temperature. During this time of RHR use, the reactor will not be restarted following the refueling shutdown that placed the fuel in the pool. When the decay heat is less than the capacity of the Fuel Pool Cooling system, either by reduction due to time or by increased system cooling capacity, the use of the RHR will be stopped and the reactor startup will be allowed.

SUMMARY OF MEASURES TAKEN
IN THE GESSAR II DESIGN TO AVOID
INTERGRANULAR STRESS CORROSION CRACKING

Table 1.5-2

COMMITMENT ITEMS

	Additional Technical Information	Reference Where Item Discussed
1.	Preop Piping Vibration Test Program	Subsection 3.9.2
2.	Reactor Internals Preop Vibration Test Program Results	Subsection 3.9.2
3.	Dynamic Analysis of Reactor Internals and Piping	Section 3.9
4.	Seismic Qualification of Class IE Electrical Equipment	Subsection 3.9.2.2 & 3.10
5.	Environmental Qualification of Class IE Electrical Equipment	Section 3.11
6.	Electrical Isolation Devices Test Program Results	Chapter 7
7.	Fuel Experience Update - NEDO-10505	See Note A
8.	Fuel Surveillance Program Results	See Note B
9.	Fuel Assembly Components Stress Report	See Note C
10.	Fuel Assembly Pressure and Temperature Capability	See Note D
11.	Fuel Assembly Dynamic Analysis	See Note C
12.	Fuel Assembly Analysis Method for Creep-Rupture	See Note E
13.	Fuel Assembly Design Limit for Instability	See Note D
14.	Fuel Channel Deformation Analysis Methods	See Note F
15.	Fuel Assembly Stress Limits	See Note D
16.	Fuel Rod 0.060 Inch Deflection Justification	See Note D
17.	Gadolinia Rods Performance Experience	See Note G
18.	Process Computer Performance Evaluation Accuracy Update	See Note H
19.	Lattice Physics Methods Verification	See Note I
20.	Boiling Water Reactor Simulator Verification	See Note J
21.	Void and Doppler Reactivity Coefficients	See Note K
22.	Full Power Scram Reactivity Function	See Note L
23.	Feedwater Flow Rate Uncertainty Justification	See Note M
24.	Resolution of Feedwater Nozzle Design and Verification	See Note N
25.	Description of WHAM Code and Loads on Internals During LOCA	See Note O
26.	Safety/Relief Valve Surveillance Program Details	(later)
27.	Update PGCC LTR NEDO-10466	See Note P
28.	Analytical Methods of Plant Transient Evaluation	Chapter 15
29.	ATWS	See Note Q
30.	Test Program for Safety/Relief Valve Solenoids	See Note R
31.	Fire Protection for PGCC	See Note P Appendix 9A
32.	Primary Coolant Pump Seals Leakage Characteristics	See Note S
33.	Large Scale Mark III Test	Appendix 3B
34.	Environmental Design of Isolation Valves and Safety Related Equipment	See Note T
35.	Post LOCA Manual Operator Actions	See Note U
36.	Instrument and Control Systems	Chapter 7
37.	HPCS Onsite Electrical Systems	Chapter 8
38.	Fire Protection for Nuclear Island Conformance	Appendix 9A

	Development and Verification Test Programs	
1.	Fuel Surveillance Program	See Note B
2.	Safety Relief Valve Surveillance Program	Section 5.2.2
3.	Core Spray Distribution	See Note V
4.	Fast Scram Design Verification	See Note W
5.	Feedwater Nozzle Design Verification	See Note N
6.	Long Term Pipe Replacement Program	~~Chapter 5~~ See Note Z
7.	Instrumentation for Vibration and Loose Parts Detection	X
8.	Pressure Suppression Design Verification	Y
9.	Suppression Pool Dynamics	Appendix 3B
10.	Evaluate Effects of Relief Valve Blow-Down	Appendix 3B and Chapter 15

Table 1.5-2

FDA COMMITMENT ITEMS (Continued)

NOTES (Continued)

Z. The GESSAR II design complies with Regulatory Guide 1.44, Rev. 0 and with the requirements of NUREG-0313, Rev. 1. Regulatory Guide 1.44 addresses 10 CFR50, Appendix A, GDCs 1 and 4, and Appendix B, i.e., requirements for components important to safety shall be designed in accordance with appropriate codes and to accomodate the effects of and to be compatible with the environmental conditions associated with reactor operations. Experience with operating reactors has demonstrated that certain wrought austenitic stainless steels, when welded, are sensitized to the degree that they are susceptible to stress corrosion cracking in reactor coolant water environments. With selection of alloys and control of thermo-mechanical processing, intergranular stress corrosion cracking (IGSCC) of reactor coolant pressure boundary components can be avoided.

All austenitic stainless steel material that is fabricated into components which see temperatures in reactor environment greater than 200^{0}F is purchased as low carbon grade or nuclear grade and in the solution annealed condition in accordance with the applicable ASTM and ASME specifications.

Cooling rates from solution annealing heat treatment temperatures are required to be rapid enough to prevent sensitization. Resistance to IGSCC is verified using ASTM A262, Practice A methods.

Material changes have been made to minimize the possibility ot IGSCC. All welded wrought austenitic stainless steel in the reactor coolant pressure boundary is low carbon nuclear grade 316LN with .02% maximum carbon content and notrgen control for strength. There is no piping which is service sensitive or nonconforming as defined in NUREG-0313, Rev. 1. All other applications of stainless steel (types 304 and 316) are of the L grade (0.03% maximum carbon content).

Table 1.5-2

FDA COMMITMENT ITEMS (Continued)

NOTES (Continued)

Z.(continued)

Welding heat input controls are required for all stainless steel welds. For machine, automatic and manual welding, interpass temperatures are restricted to 350^{0}F maximum for all stainless steel welds. High heat welding processes such as block welding and electroslag welding are not permitted. All weld filler metal consumable inserts and castings are required by specification to have a minimum of 5% ferrite.

The above practices avoid intergranular stress corrosion cracking and are reflected in Subsections 1.8.44, 4.5.1, 4.5.2, and 6.1.1.