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NOV 15 1982

Docket No. STN 50-447

General Electric Company
ATTN: Mr. Glenn G. Sherwood, Manager
Safety and Licensing Operation
Nuclear Power Systems Division
175 Curtner Avenue, Mail Code 682
San Jose, California 95125

Dear Mr. Sherwood:

Subject: Request for Additional Information Regarding the General Electric Application for an FDA for a Standardized Nuclear Island (GESSAR-II)

In our review of your request for a Final Design Approval (FDA) of your standardized nuclear island, we have identified a need for additional in ormation; our request is contained in the enclosure. The information sought is in those areas reviewed by the Reactor Systems Branch and the Core Performance Branch. We request that you submit your responses by December 30, 1982.

Recognizing the relatively compressed review schedule for your application, we suggest that you indicate within two weeks of receipt of this letter, when you could meet with us to discuss these matters if you deem that necessary. Additionally, it would be mutually advantageous if you could respond to the enclosed questions more quickly than requested so as to avoid any schedule slip.

Sincerely,

/s/

Frank J. Miraglia, Assistant Director
for Safety Assessment
Division of Licensing

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Enclosure:
As stated

cc w/enclosure:
See next page

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ENCLOSURE

ROUND 1 QUESTIONS ON GESSAR-II

DOCKET NO. STN 50-447

Reactor Systems Branch	440.01 to 440.23
Core Performance Branch	490.01 to 490.06

440.0 REACTOR SYSTEMS BRANCH

440.01 Indicate whether the design of your proposed 238 nuclear island conforms to the LRG-II positions. If there are any known exceptions at this time, so indicate.

440.02 (5.4.6) In Section 5.4.6.1.2.1 of your FSAR, you discuss the capability of performing functional testing of RCIC systems during normal plant operation. In this discussion, you state that system control provides automatic return from the test mode to the operating mode if system initiation is required. (This information is repeated in Section 5.4.6.2.4). In these sections, three exceptions are cited for which some operator action is needed. Accordingly, provide a discussion of these exceptions, including a brief description of the required operator actions, the time needed for these operator actions and whether all these actions can be performed from the control room. Additionally, address the apparent inconsistency between the sections cited above, and Section 5.4.6.2.5 in which there is no mention of any need for operator action.

440.03 (5.4.6) In Table 1.8-1 and in Section 6.3.2.2 of your FSAR, you indicate that the design of the emergency core cooling systems (ECCS) provides adequate net positive section head (NPSH) for the pumps in this system in compliance with Regulatory Guide 1.1. However, no other reactor systems are mentioned. Accordingly, indicate whether the reactor core isolation cooling (RCIC) system also complies with this regulatory guide. Additionally, provide a description of your calculations for NPSH for the RCIC system for the most limiting operating conditions. Include appropriate isometric drawings, piping sizes, elevations and flow rates.

440.04 (5.4.6) Discuss the overpressure protection design features of the low pressure portions of the RCIC system. Make reference to appropriate P&IDs to identify the low pressure piping and pressure relief devices.

440.05 (5.4.7) In Section 5.4.7.1.5 of your FSAR, you provide a discussion of the reactor heat removal (RHR) system alternate shutdown cooling mode in which water is discharged through the automatic depressurization system (ADS) valves. Provide, or make reference to, test data confirming that the ADS valves used in your design can pass sufficient water in this mode for the most limiting conditions. Include a discussion of the applicability of the particular tests which you reference.

440.06 Provide a brief description in Section 5.4.7.2.3 of your FSAR, of the function and location of relief valve E12-F030 which is discussed on page 5.4-55.

440.07 (5.4.7) On page 5.4-57 of your FSAR, you discuss the potential for water hammer caused by the sudden closure of the condensate discharge pressure control valve when the plant is in the steam condensing mode. Describe how the water level in the RHR heat exchangers is measured during this mode of operation including the type of sensor and its readout and the location of the readout. Briefly describe the procedure which will be used by the operator to control the water level to ensure that adequate protection against water hammer is provided.

440.08 (5.4.7) State whether there is a potential for water hammer due to leaking valves in the steam line connecting the RCIC system with the RHR heat exchangers thereby causing steam pockets in the RHR lines in the steam condensing mode. If so, indicate what design features you have incorporated into your design and what operational procedures are available to prevent or mitigate such occurrences.

440.09 (5.4.7) Discuss your system design provisions to prevent damage to the RHR pumps while operating in the LPCI mode under pump runout conditions during actuation of the ECCS and when operating in test modes.

440.10 (6.3) We indicate in the Standard Review Plan (SRP) that we do not allow credit for operator action for 20 minutes following a loss-of-coolant accident (LOCA). However, you describe certain operator actions to initiate containment cooling which are needed within 10 minutes following a postulated LOCA. Accordingly, provide an estimate of the time required by an operator to complete the necessary actions to initiate containment cooling assuming that a limiting single failure has occurred requiring the operator to utilize the backup system. Describe the indications available to the operator in the control room to aid him in taking the proper actions to confirm correct valve alignment and the alarms in the control room to make the operator aware of system failures and/or unavailabilities. Provide an estimate of the maximum time available for an operator to complete the planned or corrective actions, if this is necessary, before plant safety criteria are exceeded, assuming the most limiting conditions.

440.11 (6.3) Our position regarding passive failure during the long-term cooling phase of a LOCA requires, as a minimum, the assumption of the loss of a pump shaft seal or valve packing with its concomitant loss of fluid from the system in question. Show that the worst passive failure has been identified and that it can be isolated during the long-term cooling phase in the spectrum of postulated LOCA's. Valves in operating parts of the ECCS should be considered as well as in other systems serving as a boundary to prevent fluid from entering or leaving the ECCS.

440.12 (6.3) Indicate what provisions you have made to protect from the effects of cold weather, the level instrumentation for the condensate storage tank and the lines from this tank leading to the RCIC and HPCS systems.

440.13 (6.3) Identify the relief valve discharge lines in the ECCS which penetrate primary containment and have outlets below the surface of the suppression pool. Since these lines form part of the primary containment, our concern is that excessive dynamic loads resulting from waterhammer during relief valve actuation may cause cracking or rupture of these lines. Provide additional information concerning measures you have taken to prevent this type of damage to these lines.

440.14 (6.3) Discuss your design provisions which permit manual override on the ECCS subsystems once they have received an ECCS initiation signal. Provide a discussion of any lockout devices or timers which prevent the operator from prematurely terminating ECCS functions. For example, if offsite power is not available, the operator must wait until the core is flooded and then secure several of the ECCS pumps to permit the manual starting of the RHR service water pumps without overloading the diesel-generators. Discuss your design provisions which permit the operator to shutdown these ECCS pumps after they have been automatically started.

440.15 (6.3.2) On page 6.3-12 of your FSAR, you indicate that there is an interlock on high drywell pressure to maintain the HPCS flow although there is a high water level condition in the vessel. We are concerned that maintaining HPCS flow under these conditions could lead to flooding of the steam lines and possibly damage the safety-relief valves. Accordingly, provide justification for not removing the high drywell pressure interlock.

440.16 (6.3.2) The ECCS contains manual as well as motor-operated valves. There is a possibility that manually operated valves might be left in the wrong position and remain undetected prior to the occurrence of an accident. Examples of such valves include those pairs of normally closed valves which are in the test/drain lines between the HPCS, LPCS and LPCI isolation valves. Provide a list of all manually-operated valves in the safety-related reactor systems, including their location and type. Discuss the methods which will be used to minimize such an occurrence. It is our position that you provide indication in the control room for all critical ECCS valves (manually or motor-operated).

440.17 (6.3.2) In the section of your FSAR describing the HPCS, LPCS and LPCI systems, you state that the motor-operated isolation/safety injection valves are capable of opening against the maximum differential pressure expected for these systems. Briefly describe, or make reference to, the tests which will be performed to verify the valve opening capability. State the margin existing between the pressure differential against which the valves are capable of opening and the expected operating pressure differential.

440.18 (6.3.3) You have proposed certain changes for your ECCS evaluation model; these changes are currently under review. State which, if any, of the proposed changes were used in the lead plant ECCS performance evaluation described in Section 6.3.3 of your FSAR.

440.19 (15.0) Provide a listing of the transients and accidents analyzed in Chapter 15 of your FSAR for which operator action is required to mitigate their consequences. Describe in either the NSOA tables or in the sequence of events listed in Chapter 15, the manual actions or automatic system changes required to place the plant in a cold shutdown condition. This desciption should include the estimated times at which these manual actions are required.

440.20 (15.0) We state in the SRP (e.g., in Section 15.1) that for anticipated transients, the most limiting plant systems single failure shall be identified and assumed in the analysis. Accordingly, describe the worst single failure for each event analyzed in Chapter 15 of your FSAR. Provide analyses including these postulated failures for the five most limiting events identified in your FSAR.

440.21 (15.0.4) Provide further justification for your statement in Section 15.0.4.5 that applicants referencing your FSAR will need to supply analysis results only for events identified as limiting in your FSAR since the relative results will not change. Where differences in specific plants exist (e.g., bypass capability), it is our position that other transients and not just limiting transients from your FSAR, should be reanalyzed.

440.22 (15.0.4) In Section 15.0.4.5 and in Table 15.0-2 of your FSAR, you classify as "infrequent", the events identified as Load Rejection without bypass and Turbine Trip without bypass. Until approval is granted to reduce their classification, it is our position that these events be classified as "moderate" frequency events.

440.23 Provide justification for using the value of 0.0 seconds for the Safety Function Delay (Item #26) in Table 15.0-1 of your FSAR.

490.0 CORE PERFORMANCE BRANCH

490.01 (4.2.1) GESTAR-II (NEDE-24011), which contains the fuel system design safety analysis for GESSAR II, does not contain clearly identifiable design bases for most of the fuel damage, fuel failure and coolability phenomena listed in Item II.A of Section 4.2 of the Standard Review Plan (SRP). Thus, except for cladding overheating (Item II.A.2(c)) and fuel pellet overheating (Item II.A.2(d)), we have not been able to identify design basis statements in the text of GESTAR-II or in the referenced documents, even with the aid of Appendix A in Amendment 5 to NEDE-24011. While it is possible in certain cases to infer the design bases, it is preferable to have them clearly stated. Therefore, for each of the fuel system phenomena discussed in Section 4.2 of the SRP, except the two cited above, provide a concise design basis statement which indicates the design objective related to that issue. In responding to this question, provide a cross-reference to Question 490.05.

490.02 Unless otherwise stated in Section 4.2 of the SRP, you should provide a design limit for each design basis. This design limit should be a numerical value of some parameter which provides assurance that the design basis (i.e., the objective or need) will be met. For all but the following phenomena, adequate design limits have been supplied or adequate explanations have been provided for the lack of design limits: (1) Fretting Wear (Item II.A.1(c)); (2) External Corrosion and Crud Buildup (Item II.A.1(d)); (3) Fuel and Burnable Poison Rod Pressures (Item II.A.1(f)); (4) Fuel Assembly Liftoff (Item II.A.1(g)). Accordingly, provide design limits for the above listed phenomena. Alternatively, discuss why no limits are required. Design limits for cladding rupture (Item II.A.2(g)), mechanical fracturing (Item II.A.2(h)), ballooning (Item II.A.3(c)) and fuel assembly structural damage (Item II.A.3(e)) are being addressed as part of separate generic reviews and need not be discussed in your FSAR now. When our generic review of these matters is completed, you should incorporate the appropriate resolutions in your FSAR.

490.03 (4.2.2) The fuel assembly description and drawings contained in GESTAR-II are much less comprehensive than called for by Item II.B of Section 4.2 of the SRP. This particular item in the SRP contains a list of the information commensurate with an acceptable fuel system description. Accordingly, provide the information identified in Section 4.2 of the SRP.

490.04 In the recently submitted Appendix A to NEDE-24011-P-A-5, you state that the channel deflection analysis is provided in Section 5.3.2 of NEDE-21354-P. However, no such section exists in that topical report. Correct this reference. Furthermore, since the referenced channel box deflection report is relatively old (1976) and more data are available now regarding the magnitudes and rates of channel box deflection as a function of service, indicate whether: (1) the data verify the predictions of the deflection model in NEDE-21354-P; (2) your model adequately addresses channel bowing as well as bulging; and (3) you still recommend the periodic settling friction tests and measurements described in NEDE-21354-P, and if so, on what schedule. If you now recommend some other approach, or if the NEDE-21354-P procedures have been revised, describe the changes and discuss their rationale.

490.05 (4.2.3) In GESTAR-II (NEDE-24011), which is the primary support document for the fuel system for your proposed 238 nuclear island design, you have not provided a discussion of fuel assembly liftoff for normal operation and "abnormal transients" which are separate and distinct from our concerns regarding the seismic-and-LOCA-loads liftoff. As indicated in Item II.A.1(g) of Section 4.2 of the SRP, however, worst-case hydraulic loads for normal operation should not exceed the holddown capability of the fuel assembly. Although your letter from Gridley to Eisenhut, dated July 11, 1977, addresses this issue for plants and fuel designs of 1977 vintage, it is not evident that assembly liftoff will be precluded for normal operation, including anticipated operational occurrences or abnormal transients in your proposed 238 nuclear island. Accordingly, provide a discussion of your analysis of this issue. The design basis and limits aspects of this issue should be addressed as part of your response to Questions 490.01 and 490.02.

490.06 (4.2.1) You state in Section A.4.2.1.1.6 of GESTAR-II that there is no limit for internal gas pressure. The internal pressure is used in conjunction with other loads on the fuel rod cladding in calculating cladding stresses. The results of such calculations which are provided in Section 2.5.1 of NEDE-24011, show that the calculated cladding stresses can be accommodated. Although this analysis may satisfy our acceptance criteria for cladding stress (Item II.A.1.a of Section 4.2 of the SRP), it does not satisfy our acceptance criterion for rod internal pressure (refer to Item II.A.1.f of SRP Section 4.2 and Question 490.01) because this criterion involves more than stress limits on the cladding. The rod internal pressures used in your cladding stress calculations are well in excess of the nominal coolant system pressure. Accordingly, justify operation under these conditions and explain why the absence of an internal gas pressure limit does not appreciably decrease the margin of safety in calculating fuel system damage.