OCT 20 1998

Mr. G. L. Stimmell, Manager
Irradiation Processing Product
Section
Vallecitos Nuclear Center
General Electric Company
Post Office Box 460
Pleasonton, CA 94566

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION (TAC NO. MA0099)

Dear Mr. Stimmell:

We are continuing our review of your September 30, 1997, license renewal application for the General Electric Company Nuclear Test Reactor at the Vallecitos Nuclear Center. During our review, questions have arisen for which we require additional information and clarification. Please provide responses to the enclosed request for additional information. In accordance with 10 CFR 50.30(b), your response must be executed in a signed original under oath or affirmation. This request affects nine or fewer respondents and therefore, is not subject to Office of Management and Budget review under P.L. 96-511. Following receipt of the additional information, we will continue our evaluation of your application.

If there should be any questions regarding this review, please contact me at (301) 415-1128.

Sincerely,

ORIGINAL SIGNED BY:

Marvin M. Mendonca, Senior Project Manager
Non-Power Reactors and Decommissioning
Project Directorate
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Docket No. 50-73

Enclosure: As stated

cc w/enclosure: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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cc:

Mr. Steve Hsu
Radiologic Health Branch
State Department of Health Service
P.O. Box 942732
Sacramento, California 94234-7320

Mr. Ben Murray
Irradiation Processing Product Section
Vallecitos Nuclear Center
General Electric Company
P. O. Box 460
Pleasanton, California 94566

REQUEST FOR ADDITIONAL INFORMATION

GENERAL ELECTRIC COMPANY NUCLEAR TEST REACTOR

DOCKET NO. 50-73

General — In applicable chapters of the SAR, describe the interlocks and permissive circuits related to the NTR that are not already discussed in the SAR. Provide or reference an analysis that demonstrates the adequacy to achieve design functions of all NTR-related interlocks or permissives without a Technical Specification (TS) requirement. Alternatively, provide TS requirements for the applicable interlocks or permissives.

General — Abnormal, off normal, unusual event or conditions, anticipated operational occurrences and a few other terms are used in the SAR without apparent distinctions between these terms. Provide a consistent use of terms in the SAR.

Page 1-2 — Provide the reasons that potential human error was not included as a postulated initiating event or demonstrate that the considered events bound potential human error type events.

Page 1-7 — Current NRC licenses include the VBWR, GETR, and the EVESR, although section 1.8 indicates these licensed activities had end dates. Provide clarification.

Page 1-10 — Provide discussion of the purpose of this letter and Appendix A to it.

Page 2-5 — Provide additional description for title "Safety."

Page 2-9 — The second paragraph in Section 2.1.2 "Population Distribution" indicates that "[t]here are approximately ten houses immediately west of the site." Provide a quantitative indication of the distance of the nearest to the NTR. Verify that these are within the 1-km radius or not. If they are within the 1-km radius, provide clarification on the first sentence of this paragraph that implies there are only four houses and one BMX race track.

Page 2-11 — As described earlier in the Safety Analysis Report, the site is surrounded by barren mountains and rolling hills. Provide an analytical description of the effects of this terrain on the effluent releases from the stack in the meteorology section or other appropriate section.

Page 2-11 — Provide conclusions for the meteorology and hydrology sections 2.3 and 2.4, respectively.

Page 2-11 — The meteorology, hydrology, and seismology sections refer to documents that are not readily apparent or available. Provide these documents or appropriate portions of these documents, or provide readily available reference to these documents.

Page 3-1	Provide an analysis that demonstrates the adequacy to maintain personnel radiation exposure less than 10 CFR Part 20 limits of the beam shutters interlocks without a Technical Specification (TS) requirement. Alternatively, provide TS requirements for the beam shutters interlock
Page 3-2	Provide clarification on the use of the term reactor confinement building in the first paragraph.
Page 3-2	Provide an analysis of the potential compaction of the fuel without primary water loss.
Page 3-2	Provide an analysis to verify that the cadmium poison sheets "will not move relative to the core during a seismic event."
Page 3-3	Provide the applicable reference(s) for the seismic structural analysis.
Page 3-3	Provide a description of the administrative and system controls, interlocks and TS requirements for the reactor cell and stack ventilation system that are required to protect against potential fueled experiment failure.
Page 4-1	Since the current core "container was put into service in 1976 after the previous container, which had been in service for approximately 18 years, sprung a leak in a weld area," provide an analysis of the current condition of the container. Include potential weld and material neutron embrittlement considerations. Also, according to the discussion on page 4-10 the configuration allows inspection of the fuel container. Provide the results of these inspections. In addition, provide reasoning for a TS requirement on frequency for future inspections.

In this regard, please discuss, from an aging perspective, the current condition of other safety related reactor components such as control and safety rods and their drive mechanisms, wiring, coolant piping and other safety related wiring and relays, including relay contacts. If inspections of these components have been performed, please provide the results.

Page 4-6	Provide indication whether the safety and control rods are powder or solid boron carbide material. If the rods are powder, discuss features or controls that ensure detection of potential rod swelling.
Page 4-10	The cable held retractable irradiation system was not observed in Figure 1-1 or in Chapter 10. Provide an appropriate drawing of this system.
Page 4-12	Provide the figures for Chapter 10 that are referenced under the first paragraph of the "Reactor Shield" section. Similarly, for the referenced Chapter 10 figure in the "North Room Modular Stone Monument" section on page 4-14. Also, pages 10-1 through 10-8 reference drawings for the

experimental facilities, but they were not included in Chapter 10. Page A-4 in Appendix A also refers to Chapter 10 figures.

Page 4-15 The maximum pressure specified in the core may be inconsistent with the description of an unpressurized primary coolant system on page 5-1. Provide clarification.

Page 4-15 "Potential excess reactivity" is defined in the TS. Provide a definition when first used in the SAR or reference the TS definition.

Page 4-23 Regarding shutdown margin, TS 4.1.3.3 requires "calculation or measurement whenever a decrease in the reactivity worth of a safety rod is suspected." Provide description of conditions when this calculation or measurement is required.

Page 4-25 Provide reference 7 for the Jens-Lottes correlation.

Page 4-25 Provide further detail on the derivation of the peaking factors. Provide a description or reference for what combination of experimental and analytical methods to derive these peaking factors. Similarly, provide an explanation of "with neutron flux peaked on one side of the core." Describe the measurements and/or analyses used to establish circumferential and axial power profiles. Compare the results of these derivations to the assumptions used in this safety analysis report.

Page 5-4 Regarding the allowable primary system leakage rate of 10 gallons/day, provide or reference, a calculation of the maximum radiological exposure to radiological workers and to the public considering the maximum allowable radioactivity in the primary coolant system. Also, a similar calculation should be made for the potential loss of coolant accident analysis in Chapter 13. The radioactive source term should be consistent with the statement on page 5-9 that "[a]n inadvertent release of excess radioactivity in the primary coolant, of high enough level, would cause the reactor cell remote area monitor to alarm." That is, the level of radioactivity in the primary should be that just below what would initiate this alarm, or at a level corresponding to other indications that would terminate facility operation and initiate corrective actions.

Page 5-5 Provide clarification on the discussion of the primary coolant high core outlet temperature <200°F. If this is the alarm that is discussed in the next paragraph, provide indication of such.

Page 5-5 Provide clarification or reference to the use of the 222°F value in the safety analyses to ensure no fuel damage.

Page 5-9 Provide the analysis that shows that the 10^6 mR/hr alarm provides acceptable personnel and reactor protection. Include any other systems or alarms that ensure detection and correction of inadvertent release of radioactivity in the primary.

Page 5-9&10 Provide or reference the analyses that demonstrate that the specification on primary coolant water conductivity ensures aluminum corrosion is within acceptable levels.

Page 6-1 Provide the configuration of the control rods during refueling. Also, discuss the position of any safety rods that are not inserted to satisfy minimum shutdown margin requirements.

Page 6-1 Since the position of the reactor cell door during refueling appears to be optional, provide an analysis to demonstrate that it need not be maintained closed.

Page 6-5 Provide information on the stack configuration (e.g., height, diameter, draft velocity, close by buildings, delta p, effect of hilly terrain).

Page 6-6 Provide more specific reference to discussion of stack sampling/monitoirng system in Appendix A.

Page 7-1 The first paragraph and Figure 7-1 indicate a high log N power trip, but Table 7-1, Section 7.3.3 and the Technical Specifications do not appear to discuss this trip. Another trip that appears to be only in the Figure is that for seismic disturbance. Provide clarification or reference to another section of the SAR for which reactor safety system scrams, interlocks or other instrument and control systems and components are required for the accident analyses and by Technical Specification. Also, provide clarification or reference to another section of the SAR on which scram, interlocks or other instrument and control systems and components provide additional protection functions that are not specifically used to mitigate accident analyses or are required. A table may be appropriate for this clarification. Further, provide verification that the Technical Specifications Tables 3-1 and 3-2 are consistent with this safety analysis.

Page 7-5 This page indicates that bypasses are not provided on most scram circuits. The next page discusses an automatic bypass for low primary coolant flow while at powers less than 0.1 kW. Page 7-9 provides a description of other bypasses. Provide clarification of the bypasses for reactor scrams, interlocks or other instrument and control systems and components that provide safety functions. Include when the bypasses are allowed. The suggested table for the preceding question may provide a convenient mechanism for this clarification.

Page 7-9 Provide verification that the physical system does not allow bypass of more than one of the picoammeter trips. Provide verification that when only two picoammeters are operable, that bypass of a picoammeter is not allowed or describe the controls or mitigation to allow such bypass.

Page 7-15	Provide a description of the leak testing procedures and requirements for the neutron source. Include consideration of frequency and whether or not a Technical Specification requirement is needed.
Page 9-1	Reference to the ventilation system is to section 6.2. Section 6.2.1 does not appear in the section 6.2 on penetrations. Reference is also made to section 7.7 for neutron source. Provide clarification.
Page 9-1	Provide clarification on the statement that the NTR core is designed to last a lifetime, since it is understood that fuel burnup could result in the need for additional fuel by 2007.
Page 9-2	Provide verification that the reactor will be shut down for fires in the nuclear test reactor cell, and other specific locations.
Page 9-2	Provide clarification of "small and contained" in the context of fire definition.
Page 10-3	Provide verification that there are procedural controls to enter the reactor cell during operations to ensure compliance to regulatory requirements.
Page 11-2	Provide a description of or reference (e.g., section 16.1) to the other methods for detecting fuel leakage other than the primary coolant sampling. Provide an analysis to demonstrate the sensitivities of these other methods to detect fuel failure. This analysis should demonstrate that the combination of these other methods and the annual primary coolant sampling are acceptable to detect and mitigate operation with leaking fuel to prevent unacceptable radiological exposure or fuel damage.
Page 12-3	Provide clarification that licensed Reactor Operators or licensed Senior Reactor Operators will direct the activities of trainees, and that licensed Senior Reactor Operators will direct the activities of licensed Reactor Operators.
Page 12-10	Provide or reference the definition of "abnormal occurrence" in section 12.3.2. Provide or reference who can authorize restart after an abnormal occurrence. Acceptable guidance for LCO and Safety Limit violations and other events is contained in ANSI/ANS 15.1. This guidance is also applicable to section 12.4 on required actions. Additionally, all this information and guidance should satisfy the requirements of 10 CFR 50.36(c)(1) which require NRC notification and restart authorization for safety limit violations. Time periods should also be specified for reporting to NRC. The Technical Specifications (e.g., 6.5.2) should also be verified to be consistent with this guidance.

Page 12-17 Provide verification in the first paragraph of section 12.9 that the QA program includes managerial and administrative controls to ensure safe operation in accordance as required. That is, QA is not limited to design and construction. Also, identify the differences between this QA program and the guidance in ANSI/ANS 15.8 "Quality Assurance Programs for Research Reactors." Provide an analysis to justify the differences or adopt the guidance of ANSI/ANS 15.8.

Page 12-23 Provide clarification that the references are for section 12 or 13. If necessary provide corrections.

Page 13-8 Provide clarification on the net reactivity characteristics that are said to be shown in Figure 13-2.

Page 13-9 Figure 13-2 seems to assume a pump start from 100 kW at time zero that is not allowed by Tech Specs. Provide additional explanation of Figure 13-2 to describe the transient or conditions it is depicting.

Page 13-11 Figure 13-3 indicates "15-kW Scram." Provide clarification as the text and TS indicate 150 kW for scram. Verify that this demonstrates that reactivity transiants from 150 kW is more limiting than from the low power conditions for all potential reactivity transient conditions.

Page 13-30 Provide consistent terminology for "independent review unit" with that referred to in Section 12. Also, provide a more specific reference to the referred to "modulus operandi" in Section 12.

Page 13-31 Provide reference or clarification for the radiological consequence evaluation in section 13.5.3.1.

Page 13-35 Provide verification of procedural controls within 50 feet of explosive devices for the operation of unshielded high-frequency generating equipment.

Page 13-37 Provide stack draft characteristics to further demonstrate the conservatism in the concentration calculation as indicated in the footnote.

Page A-6 Provide clarification or reference (e.g., conduct of operations Chapter) for the "Specialist, Industrial Safety and Hygiene" with regard to the organization location and responsibilities.

Page A-9 Provide reference to the SAR Chapter and section that analyzed the expulsion of the beam preparation device.

Technical Specifications

Page 3-2	Provide or reference analyses that demonstrate the structure required to ensure a maximum ½ inch movement on manual poison sheets, and that demonstrate the reactivity effect of this allowed reactor core parameter.
Page 3-8	Provide clarification on the fuel loading tank low water level alarm with regard to compliance to Technical Specifications 3.3.3.1 and 3.3.3.2.
Page 3-13	Provide the bases for Technical Specification 3.4.3.4.
Page 6-13	Currently Region IV does not have regulatory oversight responsibility for research reactors. Also, the provisions of 10 CFR 50.73(d) do not apply to research reactors (this regulation is also mentioned on page 6-11). Current guidance in this regard is to send written report to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington D.C. 20555. Provide clarification.