GENERAL ELECTRIC

APPLICATION

General Electric hereby submits supplemental information to its application of January 16, 1978, including revisions of the Technical Specifications for License TR-1, Docket 50-70.

To the best of my knowledge and belief, the information contained herein is accurate.

By: R W Darmitzel
R. W. Darmitzel, Manager
Irradiation Processing Operation

Submitted and sworn before me this 11th day of December, 1979. Audrey J. Costa,
Notary Public in and for the County of Alameda, State of California.

AUDREY J. COSTA
NOTARY PUBLIC
Alameda County, California
My commission expires Sept. 16, 1982

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SUPPLEMENTAL INFORMATION FOR THE GENERAL ELECTRIC TEST REACTOR (GETR) APPLICATION OF JANUARY 16, 1978

I. Technical Specification Revisions

A. In General Electric's submittal of November 1, 1978, a new Technical Specification #1.6 was proposed which defined the term "Cold Shutdown". This term required definition because of NRC concern over the use of the term "Cold Shutdown" in the proposed Section 11 to the GETR Technical Specifications submitted by General Electric on January 16, 1978. We would like to revise the proposed Technical Specification #1.6 to read as follows:

1.6 Cold Shutdown - As used in these Technical Specifications, the reactor is in "cold shutdown" whenever: 1) the reactor is defueled (i.e. all fuel including the fuel follower assemblies is removed from the core); 2) there are no fuel elements or experiments in the reactor vessel, pool or canal, that could cause a release of radioactive material if loss of coolant from the reactor vessel, pool or canal should occur; 3) the Experiment Exhaust System (EES) has been secured and isolated, and 4) the reactor has been shut down continuously for a period of at least 30 days.

B. In addition, as described in the General Electric submittal of November 1, 1978, Technical Specification 3.2 should be changed to read:

3.2 Containment integrity shall be maintained whenever the reactor is neither secured nor in cold shutdown status. Even when the reactor is secured, but not in cold shutdown, the time during which containment integrity is not maintained shall be kept to a minimum. 1573 282

C. In recent submittals to the NRC (such as the submittal of January 30, 1978), General Electric has used the fuel melt point as the important fuel damage criteria. (The justification for using the fuel melt point is contained in the attached supplemental information.) In our recent submittals, 1200°F has been employed as the melt point. General Electric now proposes to use 1184°F instead of 1200°F. The use of the melt point (1184°F) necessitates revising existing Technical Specifications 7.1 and 8.1 as follows:

7.1 The emergency recirculation system shall be operable any time draining of the reactor vessel could result in fuel clad temperatures in excess of 1184°F. This system shall be tested for operability at least annually. If for any reason the coolant recirculation system is not operable, or if the emergency generator is not operable and cannot be made operable within 15 minutes at the time when the emergency coolant recirculation system is required to be operable, the reactor shall be shut down and defueled.

8.1 All fuel elements or fueled experiments shall be stored and handled in a geometry such that the calculated K_{eff} is less than 0.85 under optimum conditions of water moderation and reflection. All irradiated fuel whose clad surface temperatures could exceed 1184°F in air shall be stored submerged in water.

II. Supplemental Information

The following information supplemental to that given on January 16, 1978 and November 1, 1978, is provided in response to questions submitted by the NRC staff on February 14, 1979. The numbering of items is the same as in your request letter.

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Response to Request No. 1a

The method of analyzing fuel clad temperature is discussed in the letter from R. W. Darmitzel to Victor Stello of January 30, 1978. Attached to the letter is an internal memorandum which describes the physical geometry, heat generation rates, heat transfer mechanisms and analytical model used to analyze the fuel presently stored in the canal. Future calculations will utilize the same method of analysis.

Response to Request No. 1b

Experiments are analyzed to determine their capacity to retain stored radioactive materials following loss of water coolant.

For experiments containing an inventory of radioactive material which could be released if the target material (normally fuel) melts, the target must remain below the melt point following a loss of coolant accident.

For experiments containing an inventory of radioactive material which could be released at temperatures below the target melt point, an outer containment must be provided which remains intact following a loss of coolant accident. Containment integrity is determined by calculating containment steady-state temperature following the loss of coolant accident, and then evaluating the capacity of the containment to withstand imposed loads.

The experiment internal temperature distribution is conservatively determined using a steady-state, one-dimensional computer code. The code used is called the Radial Heat Transfer (RHT) code, and it is capable of handling a variety of heat transfer modes. Basically, this code determines the temperature distribution through a series of concentric shells consisting of the fuel cladding, surrounding annulus, thermal dams, external experiment containment, etc. Boundary conditions used in the RHT code are consistent with those used in evaluating reactor fuel within the pressure vessel.

Response to Request No. 1c

The fuel clad analysis (refer to the Response to Request No. 1a) takes credit for both radial and axial heat transfer. For heat transferred in the axial direction a complete loss of water was not the most severe case. The case with the bottom end of the fuel storage rack still immersed in water was found to be more severe than an air environment. Any other water level resulted in lower clad temperatures.

The experiment analysis assumes radial heat transfer only and does not consider axial heat transfer. The results are conservative, predicting higher but acceptable temperatures. All annuli and areas normally containing water (internal and external) are assumed to drain and to be filled with air. The radial heat transfer is severely retarded by the assumption of the existence of the air gaps in place of water. Any water not draining (or only partially draining) results in clad temperatures lower than the completely drained case.

Response to Request No. 1d

For all future analyses a fuel melt temperature of 1184°F will be used as the criterion for maintaining fuel integrity and assuring no release of fission products. There is virtually no release of fission products from uranium-aluminum alloy or aluminide fuel below the melt point of the fuel meat.[1,2,3]

The temperature at which U-Al alloy and UAl_x aluminide fuel starts to off-gas is 640°C (1184°F)[1]. Below this temperature essentially no fission product release is experienced.

The 800°F criterion used earlier was established because above 800°F the fuel cladding could potentially warp, possibly precluding reuse of any fuel subjected to higher temperatures. The use of 800°F was an economic rather than a safety consideration.

In order to be consistent it is requested to change the temperature limit listed in Technical Specification 7.1 and 8.1 from 800°F to 1184°F. The change from 800°F to 1184°F necessitates a change in technical specification 7.1 and 8.1 as described in Part I (page 2) above. Proposed Section 11.1.J does not require change.

Response to Request No. 1e

There is no credible condition which could limit the availability of some form of coolant to the fuel or experiments. The heat transfer environment will be either water, steam or air (or some combination of the three).

The most recent analysis of the fuel stored in the canal takes credit for heat transfer by the natural convection of air. The heat transfer mode is strictly natural convection and not forced convection. There is no credible accident which could interfere with the natural convection heat transfer.

Response to Request No. 2

There are three postulated accidents which could potentially compromise integrity of the fuel or experiments. These accidents include loss of coolant, inadvertent criticality and mechanical damage. As described above, the loss of coolant accident will be mitigated by providing sufficient decay time prior to cold shutdown to assure the fuel will not melt if the coolant is lost. The other postulated accidents are prevented or mitigated as described below:

Inadvertent Criticality

Unrestricted movement of fuel and experiments is not allowed at GETR. Strict procedural controls are in place to prevent inadvertent criticality (as well as physical damage to fuel and experiments). In order to prevent an accumulation of a critical mass all special nuclear material is handled and stored in accordance with limitations specified in an approved criticality analysis. Only one fuel element is handled at a time. All storage is procedurally controlled and elements must be stored in the fuel storage baskets. Fueled capsules are also stored in these baskets or in other approved storage locations.

Mechanical Damage

Even though procedural controls are in place to prevent mechanical damage to a fuel element, damage is of little significance. Virtually no fission products are released from alloy or aluminate fuel below the melting point (refer to the Response to Request No. 1d).

Mechanical damage to a fueled experiment is discussed in the Response to NRC Order to Show Cause, dated 10/24/77, Attachment 5, Evaluation No. 3. In the analysis it is assumed the fission products from five fueled experiments are released after only two hours decay. The result of an analysis done in accordance with Regulatory Guide 1.4 is a potential 50-year thyroid dose of 9.23 Rem to an individual for a two hour exposure at the site boundary. No other organ dose exceeds 1 Rem. Taking credit for the fact only one capsule could be damaged and using a 30 day decay period (minimum period before cold shutdown would be declared), the potential 50-year thyroid dose to the "fence post man" (2 hour exposure) would be on the order of 150 mRem.

Response to Request No. 3

When the reactor is in the "cold shutdown" mode, there is no need to maintain containment integrity (see Responses 1 and 2 above). Therefore, overpressurization or evacuation of the containment building has no safety significance.

Response to Request No. 4

In the safety analysis for our January 30, 1978 request we stated: "While the reactor is in the cold shutdown condition there are no postulated accidents that would lead to a release of radioactive material while maintaining reactor containment integrity. Therefore, there is no need to periodically perform the test referred to in proposed Technical Specification 11.1a, b, c and d.".

This paragraph should be modified to read:

"While the reactor is in the cold shutdown condition there are no postulated accidents that would lead to a release of radioactive material which would require maintaining reactor containment integrity.".

Response to Request No. 5

With reference to Response No. 1a, the analysis performed for the present shutdown period showed that 58 days of decay were required before the GETR fuel could be air cooled. Therefore only after a period of 58 days could the reactor be considered in a "cold shutdown" condition. With this or any extended shutdown the fuel would be removed from the reactor core (proposed Technical Specification 1.6 requires the reactor to be defueled before cold shutdown can be declared). With the core defueled there is no requirement to maintain operability of the control rods, the poison injection system or the nuclear and process scram instruments.

Our proposed Technical Specification 11.2 states that "all postponed tests and calibrations shall be performed prior to returning fuel or experiments to the reactor or pool except for the control rod drop time tests which shall be performed prior to startup". Therefore the operability of the poison injection system and the calibration of the nuclear and process scram instruments shall be performed prior to refueling and the control rod operability tests performed prior to startup.

Response to Request No. 6

Over the last ten-year period the total time that off-site power was lost to VNC has been approximately nineteen minutes with the longest single period of loss being fifteen minutes. The diesel generator was operative for all the times off-site power was unavailable. It is considered incredible to have loss of both off-site and on-site emergency power sources during a rare cold shutdown period coupled with no containment integrity and a release of radioactive effluents.

In any case, liquid effluent would be contained in the normal retention tanks for reuse. Any gaseous release would be detected by the routine environmental surveillance program[4].

Response to Request No. 7

The Experiment Exhaust System (EES) is secured and isolated by shutting down the jet pumps and closing all inlet and outlet valves. In this configuration, no effluent could be exhausted from the containment building to the EES hold-up tanks nor exhausted from the EES tanks to the stack.

As stated in our submittal of November 1, 1978, the EES hold-up tanks are not storage tanks. Rather, they are tanks which provide a prolonged residence time for gaseous effluents prior to release. Normally, there is no significant inventory of radioactive material in the gaseous effluent which is routed through the EES hold-up tanks.

In a cold shutdown condition (which cannot occur until at least 58 days after shutdown) all normal radioactive effluent would have been exhausted or substantially decayed to the point of insignificance.

If a substantial amount of radioactive effluent was exhausted to the hold-up tanks just prior to an extended shutdown (an extremely rare occurrence) the EES system would not be isolated until the associated activity was decayed and slowly released in the normal manner. In this case the cold shutdown mode would not be established until the effluent had been exhausted.

REFERENCES

1. W. C. Francis, Metallurgy and Materials Science Branch Annual Report Fiscal Year 1970, IN 1437, (November 1970), pg. 27-34.

2. T. J. Thompson and J. G. Beckerley, The Technology of Nuclear Reactor Safety, The M.I.T. Press, Cambridge, Mass., (1973), Chapter 18, "Fission-Product Release".

3. Telephone conversation between G. W. Parker (ORNL) and G. D. Hoggatt (GE), January 30, 1979.

4. GETR Environmental Information Report, NEDO 12623, July 1976.