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Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Ref: 1) License R-33, Docket 50-73

2) Letter, R. W. Darmitzel to E. G. Case, November 29, 1977

3) Summary Safeguards Report for the General Electric Nuclear Test Reactor, APED 4444A

4) VNC Site Emergency Plan, Section H-5, "Earthquake, Tornado, and Hurricane Plan"

5) Kramer, A. W., Boiling Water Reactors, Addison-Wesley Publishing Company, Reading, Massachusetts, 1958

Dear Mr. Diianni:

This letter is in response to several questions you had concerning the behavior of the General Electric Nuclear Test Reactor (NTR) following a seismic event.

The response to Item 1 of Attachment "B" of Reference 1, described a hypothetical reactivity transient for the NTR. The reactivity transient was the result of a postulated structural failure of a certain portion of the NTR. It was assumed that the structures used to support the control and safety rod mechanisms, as well as experiments, failed in such a preferential manner as to withdraw the control rods and experiments from the core region and prevent operation of the safety rods. The results of the previous analysis demonstrated that as long as the total reactivity insertion by the control rods, experiments and water temperature was less than 0.76$, the power transient would be terminated by bulk boiling before departure from nucleate boiling occurs (refer to Table 1 and Figure 1 of Attachment "B" of Reference 1.

Reference 1 described the sequence of events for the first two minutes of the hypothetical event. This letter describes several possible sequences of events from the two minutes point in time out to the final state for the reactor.

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The diagram presented in Figure 1 of this letter shows the various possible states for the NTR following the initial reactivity transient. If no operator intervention is taken, the final state of the reactor will always be state (A) (reactor shut down due to loss of coolant). The extremely conservative loss of coolant analysis presented in Reference 2 has demonstrated that the loss of coolant for the NTR has no significant consequences.

The performance of the reactor in the near term after the postulated seismic event depends on the extent of damage to the remainder of the reactor system. The most significant items are: 1) the primary system piping, 2) the primary pump, 3) the secondary water supply system, and 4) the electrical supply to the reactor system.

It is highly unlikely that the primary system for the reactor would still be intact after a seismic event severe enough to result in the reactivity addition by the massive structural failure postulated here. If the primary system were failed at the same time as the reactivity addition, the reactivity transient would not be significantly altered. The loss of coolant from the reactor results in shutdown by voiding the reactor core, state (A) on Figure 1.

We may also assume loss of electric power as: 1) it is highly improbable that electric power to the site-leave alone power to the NTR-would survive the event postulated here; and 2) even in such an improbable circumstance site emergency procedures call for the termination of all utility services to any buildings or facilities believed to have suffered potential damage.

As the loss of electrical supply automatically deactivates the primary system pump and automatically closes off flow to the secondary system, the structural fate of the secondary system becomes a moot question, and we need consider only the possibility that the primary system somehow survives the event. If the primary system does not fail or leaks at a very slow rate the system will arrive at state (B) . For this state, the reactor will operate in a natural circulation mode at less than half of the pumped flow rate and at a power of less than 20 kW. Since there is no secondary cooling, the inlet temperature to the reactor core will be at the saturation temperature and the coolant will be boiled away or evaporated at a rate of less than one gallon per hour. There are roughly 1,000 gallons of the 1,800 gallons of water in the fuel storage tank which could drain into the reactor core can via the fuel loading chute to make up for the boiloff. If no water were made up to the system and no action were taken to shut the reactor down, it would operate for forty days or more at the 20 kW or less power level. The loss of coolant by boiling will be a less severe event than the loss of coolant event described in Reference 2 for two reasons. First, the reactor power is lower (20 kW rather than 100 kW) and, if a primary system leak was not developed, the loss of coolant is not complete. In fact, the slow

loss of coolant will result in a slow decrease in power and only a partial loss of coolant will occur. The core can be voided from the top by nearly 20% before any single fuel element is totally uncovered and the surface heat flux would be so low that it could quite easily be cooled by convection to the steam and radiation to the relatively cool environment.

As the maximum fuel temperature at a power level of 100 kW is 650°F, no fuel melting can be expected. Furthermore, Reference 2 has demonstrated that 10CFR100 criteria would not be violated even if there were full meltdown with an attendant fission product release.

Concerning the questions on the NTR temperature and void coefficient the temperature coefficient used in the NTR transient analysis is discussed in Section 4.7.4 of APED-4444-A and is given by:

$$\frac{d\rho}{dt} = -5.7 \times 10^{-3}(T-124)¢/°F$$

where T is the primary coolant temperature in °F and ρ is the reactivity of the system in ¢.

Since both the void and temperature coefficients are predominantly a function of water density, the void coefficient can be accurately estimated by determining the reactivity change for a known change in water density.

Integrating the formula for the temperature coefficient for temperature $\geq$ 124°F results in:

$$\rho = -2.85 \times 10^{-3}(T-124)^2¢$$

Using this relationship and the specific volume values from steam tables, the data shown in the table below are obtained for various temperatures up to the saturation temperature of 228°F.

Temperature	°F	150	165	200	228
Specific Volume	cu.ft./lb.	.01634	.01642	.01663	.01683
Density	grams/cc	.9802	.9754	.9631	.9517
Reactivity (124 to T)	¢	-1.93	-4.79	-16.46	-30.83

The average void coefficient in ¢/% void for any temperature range can be computed by dividing the change in reactivity by the % change in density.

For the 150 to 165°F range, the void coefficient is 5.8¢/% void, essentially the same value as the 5.7¢/% void value stated in Section 4.7.5 of APED-4444-A and used in the transient analysis. The void coefficient for the 165 to 200°F range is 9.2¢/% void and for the 200 to 228°F range is 12.1¢/% void. At 228°F the void coefficient is calculated to be 16.5¢/% void, about a factor of 3 larger than the value used in the transient analysis.

It is reasonable to conclude from the above that NTR transient analyses which are controlled or terminated by voids are extremely conservative even if significant pressure suppression of voids or uncertainty in the true void coefficient were assumed.

Very truly yours,

G. E. Cunningham
Sr. Licensing Engineer

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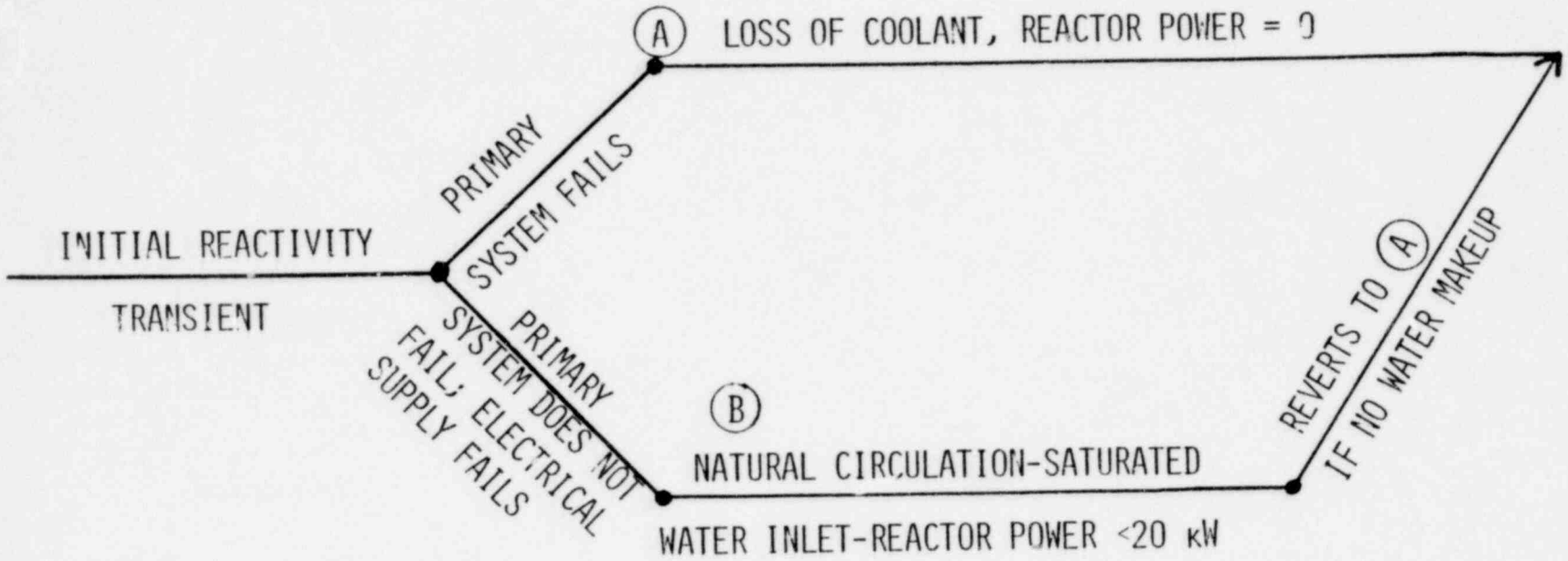

FIGURE 1

POSSIBLE REACTOR STATES FOLLOWING
THE POSTULATED SEISMIC EVENT

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