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

FINAL

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

GENERAL ELECTRIC CORPORATION

NUCLEAR TEST REACTOR (NTR)

VALLECITOS BOILING WATER REACTOR (VBWR)

ESADA-VALLECITOS EXPERIMENTAL SUPERHEAT REACTOR (EVESR)

DOCKET NOS. 50-73, 50-18, and 50-183

Introduction

In accordance with the Order to Show Cause issued by the Commission on October 24, 1977, the NRC staff reviewed all NRC-licensed activities at Vallecitos Nuclear Center (VNC) other than the General Electric Test Reactor (GETR) which is the subject of the Order to Show Cause. On the basis of information previously prepared by the NRC staff and further information provided orally (on October 28, 1977) by General Electric Company (the licensee), the NRC issued a preliminary safety evaluation dated November 7, 1977. This evaluation considered the latest information available to the NRC staff regarding the geologic and seismologic environment at VNC. Of particular concern to the staff was the continued operation of the General Electric Nuclear Test Reactor (GENTR) which operates under License No. R-33.

By the same letter dated November 7, 1977, we requested that the licensee respond to seven items of supporting information. In response to this request, the licensee submitted replies to the seven items on November 29, 1977, and additionally submitted a seismic analysis of the GENTR for events which would produce peak ground accelerations of 0.8 g. Subsequently, on October 18, 1979, the licensee submitted a clarification of the response to the first of the seven requested items and on April 3, 1980, submitted six Technical Specification change requests because of additional oral questions by the NRC staff.

This safety evaluation, when combined with the safety evaluation issued November 7, 1977, is considered to be final for the Nuclear Test Reactor, the Vallecitos Boiling Water Reactor and the Esada-Vallecitos Experimental Superheat Reactor.

Discussion and Evaluation

Vallecitos Boiling Water Reactor (VBWR)

The staff concluded, in the preliminary safety evaluation, that the accidental spill of about 2525 gallons of contaminated water from the cooling system of the deactivated VBWR would not present an undue safety hazard. However, we requested that the licensee confirm that the emergency procedures for the VBWR include provisions for sampling water supplies in the area, if a release of

8011120 624

radioactive liquid should occur. The licensee has stated that its Reactor Irradiations organization is responsible for routine surveillance of the VBWR containment building and additionally for the operation and maintenance of the GETR. Procedures for the inspection, access control, facility use, and facility modification are included in the GETR procedures. Specifically, the GETR Emergency Plan contains instructions for actions to be taken in case of an earthquake. This section has been modified to require measurement of the VBWR cooling system water level and, if the measurement indicates a loss of water, to take samples of offsite potable water supplies. We concluded that these changes in procedures, combined with the possible natural dilutions and initially low concentrations of radioactivity, make it unlikely that exposures greater than those allowed from routine releases under the provisions of 10 CFR 20 of the Commission's regulations could occur. We therefore find these changes acceptable.

Esada-Vallecitos Experimental Superheat Reactor

The section of the safety evaluation issued November 7, 1977 concerning the Esada-Vallecitos Experimental Superheat Reactor is complete as issued. No further evaluation on that facility was necessary since there were no unresolved issues.

General Electric Nuclear Test Reactor

The GENTR is a 100-kwt light-water-cooled and moderated, graphite-reflected research reactor composed of highly enriched uranium fuel. The fuel assemblies, consisting of aluminum clad uranium-aluminum alloy discs on aluminum shafts, are arrayed in a horizontal, cylindrical tank centered in a five-foot cube of graphite with a central graphite thermal column. The reactor is equipped with four safety rods which automatically insert on a scram signal and three control rods (which do not insert on a scram signal) which are driven horizontally along worm screws through one vertical face of the graphite into the reflector region immediately surrounding the core annulus. "Manual poison sheets," composed of cadmium, are manually positioned in the graphite reflector around the core annulus.

The reactor is housed in a thick-walled concrete cell within Building 105 of the VNC.

During the October 28, 1977 meeting, the licensee presented the results of calculations which showed that if the total reactivity insertion were limited to $0.8, no fuel melting (that is, no release of fission products) would occur even if all control rods were assumed to be withdrawn as a result of earthquake damage. The licensee submitted analyses in correspondence dated November 29, 1977 and October 18, 1979 for our review. The methods used in the kinetic analyses are the same as those used by the licensee in previously approved safety analyses and are considered by the NRC staff to be applicable to the present accident evaluation. Although the licensee showed by analysis that limiting the excess reactivity to $0.8 would result in no fuel damage, he proposed to further limit the excess reactivity to $0.76, resulting in applying additional conservatism to assure that there are no mechanisms which could cause fuel damage.

If the primary system is assumed to fail at the same time excess reactivity amounting to \$0.76 were added to the reactor operating at a power level of 10^{-7} kw, a transient of about 40 seconds duration occurs which is terminated by bulk boiling and leakage from the primary system. For this case, the reactivity addition due to the positive temperature coefficient up to 124°F is important to the transient, but peak fuel temperature is limited to about 255°F, well below the clad melting temperature of about 1200°F. Therefore, we have concluded that fuel damage will not occur by the combined primary system failure and the addition of \$0.76 excess reactivity. If, on the other hand, the primary system is assumed to remain intact except for a failure of the fuel storage tank (1,800 gallon capacity of which 1,000 gallons can be drained through a failure into the primary tank) while the electrical system fails removing primary pumping power, the transient takes a different course. Assuming no intervention by the licensee, the reactor would operate in the natural convection mode at 20 kw or less for 40 or more days while the moderator (including the 1,000 gallons from the fuel storage tank) slowly boiled away. Approximately 20% of the core can be voided from the top before any fuel element is totally uncovered, but the heat flux would be low enough that the element could be easily cooled by convection to the steam and by radiation to the relatively cool environment. Since the peak power level for this transient would be 20 kw, the peak fuel temperature would be less than 650°F which corresponds to the maximum fuel temperature at 100-kw power operation. Consequently, we have concluded that this transient will not result in fuel element failure since the peak fuel temperature is well below the clad melting temperature of 1200°F.

The three proposed Technical Specification changes which limit the excess reactivity from the temperature coefficient, poison rods, and experiments to not greater than \$0.76, will provide ample assurance that transients no worse than discussed above could occur due to severe earthquake damage.

In our preliminary safety evaluation, we assumed the reactivity insertion occurrence of the magnitude existing in the Technical Specification in conjunction with control rod failure. In this case, fuel melting occurs and the NRC staff evaluated the resulting fission product release using the meteorological assumptions of Regulatory Guide 1.4. The staff requested from the licensee onsite meteorological data to confirm the assumptions. However, with the Technical Specification change limiting the available excess reactivity such that no fuel melting would occur after an accident, the data are no longer required. Further, the licensee was requested to evaluate proposed attachments of experiments to the core to assure that no motion of the experiments relative to the core could occur as a result of an earthquake. Again, since the reactivity worth of all experiments is now limited in conjunction with poison rod reactivity worth and temperature coefficient to safe levels, no data or analyses of attachments to the core are necessary because the experiments are assumed to fail in the event of an earthquake.

The excess reactivity of the core is limited by the use of manual cadmium poison sheets. It is necessary that these sheets be secured in place relative to the core during and after an earthquake in order that their large worth (about \$5 for the total of six) not contribute to the magnitude of the excursion. The licensee tested a mocked-up latching mechanism to 500 lbs without failure of the latch. This force is approximately 100 times higher than that needed to restrain the manual sheets during a 1-g acceleration. Although the magnitude at a postulated design basis earthquake at VNC is not adequately defined, we find the latching mechanism acceptable in maintaining the manual poison sheets in place for such a postulated accident. This position is based on test results that show that the latching mechanism can adequately withstand forces well in excess (by factor 100) of those that could be generated by an earthquake having a 1-g acceleration. Furthermore, it is reasonable to assume that the latching mechanism can withstand forces generated by any reasonably assumed design basis earthquake.

The fourth proposed Technical Specification change would require all six sheets to be restrained in their respective slots in the graphite reflector during reactor operations and limit their potential movement relative to the core to less than 0.5 inch (the expected movement is estimated to be less than 0.125 inch). This slight relative movement of the manual poison sheets with respect to the core is judged to have an insignificant effect on the core reactivity insertion and, therefore, any power increase would be minimal (i.e., within calculational accuracies).

Based on the above, we agree with the licensee that the latching mechanism is adequate to prevent movement of the manual poison sheet during a seismic event.

Although all attachments of experiments to the core are assumed to fail during seismic events, movement of objects in the fuel loading chute (where they could fall closer to the core) is to be avoided during all normal operations. The licensee proposed a fifth change to the Technical Specifications to resolve this concern in that the experiments in the fuel loading chute which are assumed to fail during seismic events must be adequately attached to the core so that experiments remain in place for normal operating conditions. This change results in achieving an adequate level of assurance that the core/experiment geometry is stable, thus minimizing challenges to the reactor safe shutdown system. We find this change addresses our concern and is therefore acceptable.

In the case of the sixth proposed Technical Specification change, the licensee requested the deletion of the requirement for securing experiments with a negative reactivity worth greater than $0.8. Since, by the proposed Technical Specification change, the total reactivity worth from all sources would be limited to $0.76 as discussed above, then there will never be such an experiment having negative reactivity worth greater than $0.8 making such a requirement in the existing Technical Specification meaningless. We agree with the licensee that such a requirement is not necessary and should be deleted.

Environmental Consideration

We have determined that this amendment will not result in any significant environmental impact and that it does not constitute a major Commission action significantly affecting the quality of the human environment. We have also determined that this action is not one of those covered by 10 CFR § 51.5(a) or (b). Having made these determinations, we have further concluded that, pursuant to 10 CFR § 51.5(d)(4), an environmental impact statement or environmental impact appraisal and negative declaration need not be prepared in connection with issuance of this amendment.

Conclusion

The staff has concluded, based on the considerations discussed above, that: (1) because this action does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in safety margin, this action does not involve a significant hazards consideration; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; and (3) such activities will be conducted in compliance with the Commission's regulations and this action will not be inimical to the common defense and security or to the health and safety of the public.

Dated: November 5, 1980