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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION SUPPORTING PROPOSED AMENDMENT TO PROVISIONAL OPERATING LICENSE DPR-16 AND PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS

SUPPRESSION POOL WATER TEMPERATURE LIMITS

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

Introduction

By letter dated February 15, 1975 to Jersey Central Power & Light Company, the Nuclear Regulatory Commission (NRC) requested that the licensee among other things, develop operating procedures and proposed changes to the Technical Specifications to preclude reaching elevated temperatures of the torus pool water and to provide for inspection of the torus as appropriate to identify any damage in the event of an extended relief valve operation. By letter dated April 1, 1975 Jersey Central submitted a response which stated that the present Technical Specifications provide adequate limits for the suppression chamber water temperature, thus the licensee proposed no change to the Technical Specifications. This response from the licensee was found unacceptable; and, as a result, the NRC staff prepared appropriate technical specification changes to revise the suppression pool water temperature limits for Oyster Creek Nuclear Generating Station. By letter dated July 16, 1975, the NRC staff advised the licensee of its intent to initiate steps to issue these technical specification changes unless the licensee objected in writing. By letter dated August 8, 1975, the licensee provided comments on the technical specification changes proposed by the NRC staff. Subsequent discussions between the NRC staff and the licensee resulted in a few mutually acceptable changes to the Technical Specifications proposed by the NRC staff. The licensee stated it would accept the proposed Technical Specifications with these changes.

Discussion

Oyster Creek is a boiling water reactor (BWR) which is housed in a Mark I primary containment. The Mark I primary containment is a pressure suppression type of primary containment that consists of a drywell and a suppression chamber (also referred to as the torus). The suppression chamber, or torus, contains a pool of water and is designed to suppress the pressure during a postulated loss-of-coolant accident (LOCA) by condensing the steam released from the reactor primary system. The reactor system energy released by relief valve operation during operating transients also is released into the pool of water in the torus.

Experiences at various BWR plants with Mark I containments have shown that damage to the torus structure can occur from two phenomena associated with relief valve operations. Damage can result from the forces exerted on the structure when, on first opening the relief valves, steam and the air within the vent are discharged into the torus water. This phenomenon is referred to as steam vent clearing. The second source of potential structural damage stems from the vibrations which accompany extended relief valve discharge into the torus water if the pool water is at elevated temperatures. This effect is known as the steam quenching vibration phenomenon.

A. Steam Vent Clearing Phenomenon

With regard to the steam vent clearing phenomenon, we are actively reviewing this generic problem and in our letter dated February 15, 1975 we also requested the licensee to provide information to demonstrate that the torus structure of the primary containment will maintain its integrity throughout the anticipated life of the facility. In its response dated April 1, 1975 the licensee stated that it was investigating this matter and the results of the investigation would be submitted to us on a schedule consistent with the timing which we proposed for licensee repsonse. Because of the apparent slow progression of the material fatigue associated with the steam vent clearing phenomenon, we have concluded that there is no immediate potential hazard resulting from this type of phenomenon; nevertheless, surveillance and review action on this matter by the NRC staff will continue in due course during this year.

B. Steam Quenching Vibration Phenomenon

The steam quenching vibration phenomenon became a concern as a result of occurrences at two European reactors. With torus pool water temperatures increased in excess of 170F due to prolonged steam quenching from relief valve operation, hydrodynamic fluid vibrations occurred with subsequent moderate to high relief valve flow rates. These fluid vibrations produced large dynamic loads on the torus structure and extensive damage to torus internal structures. If allowed to continue, the dynamic loads could have resulted in structural damage to the torus itself due to material fatigue. Thus, the reported occurrences of the steam quenching vibration phenomenon at the two European reactors indicate that actual or incipient failure of the torus can occur from such an event.

Such failure would be expected to involve cracking of the torus wall and loss of containment integrity. Moreover, if a LOCA occurred simultaneously with or after such an event, the consequences could be excessive radiological doses to the public. In comparison with the steam vent clearing phenomenon, the potential risk associated with the steam quenching vibration phenomenon (1) reflects the fact that a generally smaller safety margin[1/] exists between the present license requirements on suppression pool temperature limits and the point at which damage could begin and (2) is more immediate.

Evaluation

The existing Technical Specifications for Oyster Creek limit the torus pool temperature to 100F. This temperature limit has been reduced to 95F to provide 5F temperature difference between a scram requirement discussed below and provisions for performing necessary surveillance. The temperature of 95F assures that the pool water has the capability to perform as a constantly available heat-sink with a reasonable operating temperature that can be maintained by use of heat exchangers whose secondary cooling water (the service cooling water) is expected to remain below 95F. While this 95F limit provides normal operating flexibility, short-term temperatures permitted by operating procedures exceed the normal power operating temperature limit, but accommodates the heat release resulting from abnormal operation, such as relief valve malfunction, while still maintaining the required heat-sink (absorption) capacity of the pool water needed for the postulated LOCA conditions. However, in view of the potential risk associated with the steam quenching vibration phenomenon, it is necessary to modify the temperature limits now in the license Technical Specifications. This action was, as discussed in our February 13, 1975 letter, first suggested by General Electric Company (GE) who had earlier informed us of the steam quenching vibration occurrences at a meeting on November 1, 1974 and provided related information by letters to us dated November 7, and December 20, 1974. The December 20 letter stated that GE had informed all of its customers with operating BWR facilities and Mark I containments of the phenomenon and included in these communications GE's recommended interim operating temperature limits and proposed operating procedures to minimize the probability of encountering the damaging regime of the steam quenching vibration phenomenon.

Our implementation of the GE recommended procedures and temperature limits via changes in the Technical Specifications are evaluated in the following paragraphs:

1/ The difference, in pool water temperature, between the license limit(s) and the temperature at which structural damage might occur is the safety margin available to protect against the effects of the phenomenon discussed.

a. The new short-term limit applicable to all conditions requires that the reactor be scrammed if the torus pool water temperature reaches 110F. This requirement to scram at 110F provides additional assurance that the torus temperature will remain below the 170F temperature related to potential damage to that torus.

b. For specific requirements associated with surveillance testing, i.e., testing of relief valves, the water temperature shall not exceed 10F above the normal power operation limit. This new limit during surveillance testing of relief valves provides additional operating flexibility while still maintaining a maximum heat-sink capacity. The current limits in the Technical Specifications make no provision for these requirements.

c. For reactor isolation conditions, the new temperature limit is 120F, above which temperature the reactor vessel is to be depressurized. This new limit of 120F assures pool capacity for absorption of heat released to the torus while avoiding undesirable reactor vessel cooldown transients. Upon reaching 120F the reactor is placed in the cold, shutdown condition at the fastest rate consistent with the technical specifications on reactor pressure vessel cooldown rates.

Conclusion

We have concluded, based on the consideration discussed above that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: OCT 8 1975