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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING CHANGES TO TECHNICAL

SPECIFICATIONS OF FACILITY LICENSE DPR-59

POWER AUTHORITY OF THE STATE OF NEW YORK

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

Introduction

By letter dated February 13, 1975 to Power Authority of the State of New York (PASNY), 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 March 31, 1975, PASNY 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. For the reasons set forth in this evaluation, this response from the licensee was found unacceptable. Appropriate changes to the Technical Specifications are needed to assure the proper operation and integrity of the pressure suppression primary containment system.

Discussion

The FitzPatrick Plant 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 the 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 13, 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. By letter dated May 2, 1975, the licensee indicated that the requested information would be furnished by July 1, 1975. 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. Moreoever, 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

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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.

during surveillance testing of relief valves provides operating flexibility while still maintaining a maximum heat-sink capacity.

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.

d. In addition to the new limits on temperature of the torus pool water, the discussion in the Basis includes a summary of required operator actions to be taken in the event of a relief valve malfunction. These operating actions are taken in order to avoid the development of temperatures approaching the 170F threshold for potential damage by the steam quenching phenomenon.

Conclusion

We have evaluated the GE recommendations consisting of new suppression pool temperature limits and operating procedures. We conclude that the procedures and temperature limits discussed above are appropriate and are needed to assure that the containment functions as designed in order to protect the public health and safety.

Date: July 15, 1975