UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, DC 20555-0001

November 14, 1996

NRC INFORMATION NOTICE 96-60: POTENTIAL COMMON-MODE POST-ACCIDENT FAILURE OF RESIDUAL HEAT REMOVAL HEAT EXCHANGERS

Addressees

All holders of operating licenses or construction permits for nuclear power reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to alert addressees to a potential common-mode post-accident failure of boiling water reactor (BWR) residual heat removal (RHR) heat exchangers. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Background

The NRC staff issued Information Notice 96-45, "Potential Common-Mode Post-Accident Failure of Containment Coolers," on August 28, 1996, to alert licensees to a potential failure mechanism of containment coolers during a design-basis loss-of-coolant accident (LOCA) with a concurrent loss of offsite power or with a delayed sequencing of safety-related equipment. Specifically, if the containment coolers are exposed to forced convection in a post-LOCA environment without cooling water flow, boiling may occur on the stagnant cooling water side of the containment coolers. When cooling water flow is subsequently reinitiated, the steam voids will collapse and may create significant hydrodynamic loads (waterhammer). The waterhammer may threaten the integrity of the containment cooler or the associated cooling water piping. Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," issued on September 30, 1996, also addressed this issue.

Description of Circumstances

A potential waterhammer scenario in the service water side of the RHR heat exchanger was identified at the LaSalle Nuclear Station. The postulated phenomenon is analogous to that identified for containment coolers in Information Notice 96-45 and Generic Letter 96-06.

At LaSalle, a BWR 5 plant, the top of the vertical RHR heat exchanger is the high point of the RHR service water system and is at an elevation that is higher than that of the ultimate heat sink by approximately 9 meters (30 feet) for normal lake levels. The resultant pressure in the heat exchanger tubes in a standby condition is estimated to be between 7000 and 21,000 pascals (1 and 3 psia), with associated saturation temperatures between 43° and 60° C (110° and 140° F). Boiling will occur in these tubes if these temperatures are exceeded. Under low lake levels, the licensee found that voiding would occur in the top of the tubes regardless of temperature.

At LaSalle, the RHR system is lined up so that flow will go through both the RHR heat exchanger and a bypass valve whenever the RHR system is initiated in its safety injection (low pressure coolant injection) mode. The RHR service water flow to the tube side of the RHR heat exchangers does not start automatically on a safety injection signal. Emergency operating procedures direct that the RHR service water pumps be started manually within the first 10 minutes of an accident.

It is postulated that during a LOCA, water from the suppression pool, which could start from the technical specification allowable 41° C (105° F), could quickly be heated to 88°C (190° F). When the relatively hot suppression pool fluid would be pumped through the RHR heat exchanger, it would boil the stagnant low pressure service water. When RHR service water flow would subsequently be manually initiated, the steam voids would collapse and might create significant hydrodynamic loads. The hydrodynamic loads might impair the integrity of the heat exchangers or the associated service water piping. This scenario does not occur during normal operation of the RHR system in the shutdown cooling mode because the RHR service water system is started first to ensure that cooling flow is established before hot fluid is introduced into the heat exchangers.

Both units at the LaSalle Nuclear Station are currently shut down. The licensee continues to assess this issue and will ensure operability of the RHR heat exchangers before restart. The licensee is considering the installation of a keep-fill system on the RHR service water system to keep the heat exchangers pressurized.

Discussion

The RHR heat exchangers provide an important safety function for long term heat removal. The postulated failure scenario could cause a common failure of the RHR heat exchangers, thereby potentially challenging this function. In addition, a structural failure of the RHR heat exchangers could create a containment bypass release path and divert low pressure coolant injection flow. An individual plant vulnerability to these postulated failures depends on a number of factors. The physical configuration of the RHR service water system and RHR heat exchanger, the operational alignment of the heat exchanger isolation valves, the sequencing of the RHR service water pumps, and other site-specific parameters have an effect on facility vulnerability to this potential failure mode.

This information notice requires no specific action or written response. If you have any questions about the information in this notice, please contact one of the technical contacts listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

DBMatthews

for Thomas T. Martin, Director
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Technical contacts: V. Patricia Lougheed, Region III
(630) 829-9760
E-mail: vpl@nrc.gov

James Tatum, NRR
(301) 415-2805
E-mail: jet1@nrc.gov

John Tappert, NRR
(301) 415-1167
E-mail: jrt@nrc.gov

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96-59	Potential Degradation of Post Loss-of-Coolant Recirculation Capability as a Result of Debris	10/30/96	All holders of OLs or CPs for nuclear power reactors
96-58	RCP Seal Replacement with Pump on Backseat	10/30/96	All holders of OLs or CPs for pressurized-water reactors
96-57	Incident-Reporting Requirements Involving Intakes, During a 24-Hour Period That May Cause a Total Effective Dose Equivalent in Excess of 0.05 Sv (5 rems)	10/29/96	All U.S. Nuclear Regulatory Commission licensees
96-56	Problems Associated with Testing, Tuning, or Resetting of Digital Control Systems While at Power	10/22/96	All holders of OLs or CPs for nuclear power reactors

OL = Operating License
CP = Construction Permit

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original signed by D. B. Matthews

for Thomas T. Martin, Director
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Technical contacts: V. Patricia Lougheed, Region III
(630) 829-9760
E-mail: vpl@nrc.gov

James Tatum, NRR
(301) 415-2805
E-mail: jet1@nrc.gov

John Tappert, NRR
(301) 415-1167
E-mail: jrt@nrc.gov

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Tech Editor reviewed 10/29
DOCUMENT NAME: 96-62.IN

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DATE	10/30/96	11/04/96	11/06/96	11/12/96

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Office of Nuclear Reactor Regulation

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(630) 829-9760
E-mailt: vpl@nrc.gov

James Tatum, NRR
(301) 415-2805
E-mail: jetl@nrc.gov

John Tappert, NRR
(301) 415-1167
E-mail: jrt@nrc.gov

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Internet: vpl@nrc.gov

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(301) 415-2805
Internet: jet1@nrc.gov

John Tappert, NRR
(301) 415-1167
Internet: jrt@nrc.gov

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