UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

August 10, 1993

NRC INFORMATION NOTICE 93-62: THERMAL STRATIFICATION OF WATER IN BWR REACTOR VESSELS

Addressees

All holders of operating licenses or construction permits for boiling water reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to alert addressees that loss of forced circulation through the reactor vessel coupled with isolation from the main condenser may allow cold water to stratify in the bottom of the reactor vessel and cause temperatures to be lower than allowable. 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.

Description of Circumstances

Hatch Unit 1

On August 27, 1992, at Unit 1 of the Hatch Nuclear Power Plant, a high radiation signal from a main steamline radiation monitor initiated a Group 1 isolation. The main steam isolation valves closed and the reactor automatically scrammed from 100-percent power. The resulting low water level in the reactor vessel caused the recirculation pumps to trip thereby terminating forced circulation through the reactor vessel. There was no circulation by the reactor water cleanup system because the licensee had previously isolated that system for testing. The reactor vessel water level was restored by the steam-driven feedwater pumps and the reactor core isolation cooling (RCIC) system.

After steam was no longer available to the feedwater pump turbines, water level was maintained primarily by the RCIC system. Relatively cool water was added to the reactor vessel by injection from the RCIC system into the feedwater sparger and from the control rod drive system into the lower region of the reactor vessel. Initially, the operators used the temperature of the coolant in the drain line from the bottom head of the reactor vessel to monitor the temperature in the reactor vessel. Later, the operators realized that the temperature in the drain line would not be meaningful because the reactor water cleanup system had been secured and there was no flow in the drain line. Because of this, the operators changed to another point on the reactor vessel. That location was subsequently determined to be above the reactor vessel bottom head and not representative of the minimum temperature in the vessel. A vendor review of the event determined that at one point the bottom head temperatures were 8°C [15°F] lower than were allowed by the pressure-temperature limits in the technical specifications.

The operators could not restart the reactor recirculation pumps because the difference in temperature between the reactor dome and the reactor bottom was greater than the technical specification limit of 62.8°C [145°F]. After the reactor was depressurized and while the coolant was still stratified, the operators started one residual heat removal pump in the shutdown cooling mode. The temperature at the drain line increased 82°C [220°F] in 10 minutes from its initial 50°C [90°F] temperature; this change exceeded the technical specification limit of 37.8°C [100°F] change in one hour.

Peach Bottom Unit 3

On October 15, 1992, at Peach Bottom Unit 3, a half-isolation of the primary containment occurred after operators had performed a surveillance test of low-pressure switches on the main steamline. While plant personnel were checking the relays to determine the cause of the half-isolation signal, a second half-isolation signal was received. The main steam isolation valves closed and the reactor scrammed from 100-percent power. High-pressure coolant injection and RCIC automatically initiated and, in conjunction with the main safety relief valves, were used to control water level and system pressure. During recovery from the transient, a second reactor scram from high pressure occurred. Because of a delay in resetting the first scram and limited flow through the reactor drain line, thermal stratification of the reactor coolant occurred in the vessel. The operators did not consider the temperature of the drain line to be representative of the bottom head temperature because the reduced flow rate through the drain line caused too low an indicated temperature due to heat losses from the drain line.

The operators could not restart the recirculation pumps because the temperature difference between the reactor dome and the drain line was greater than the 62.8°C [145°F] allowed by the technical specifications for restarting a recirculation pump. Therefore, the operators proceeded to depressurize and cool the reactor before restoring forced circulation. Although other bottom head metal temperature indications were available, they were not actively monitored by the operators because of procedural deficiencies and lack of training. A subsequent review of recorded temperature data determined that the temperature difference between the reactor dome and the bottom head drain line was about 135°C [240°F] and that the pressure-temperature limits for the reactor vessel bottom head had been violated. With the reactor pressure about 4.14 MPa [600 psig], the reactor vessel metal temperature had decreased to about 64°C [115°F], nearly 22°C [40°F] lower than the pressure-temperature limit.

Discussion

These events demonstrate that isolation of the reactor vessel from the main condenser with loss of recirculation flow can be initiated by a variety of causes. Isolation of the reactor vessel causes the operators to take manual control to restore proper water level and system pressure. A restart of the recirculation pumps may be delayed because of procedural restrictions. General Electric has issued a number of communications to licensees regarding the loss of forced circulation in the reactor vessel. In those communications, General Electric addressed issues such as potential operating difficulties, concerns about reactor vessel temperature monitoring, and the potential for thermal stratification within the reactor vessel. Also, plant technical specifications contain pressure-temperature limitations to ensure the integrity and safe operation of the reactor coolant system.

Once thermal stratification occurs, any rapid circulation of water could result in a large step change in the temperature of the water adjacent to the reactor bottom head penetrations. This step change may violate the technical specification limits for rate of temperature change. A temperature differential within the reactor vessel may be reduced by increasing coolant flow out of the bottom head drain and reducing cold water flow through the control rod drive system, which enters the bottom region of the reactor vessel.

Another concern is maintaining operation within brittle fracture temperature limits. Once temperature differences develop in the reactor vessel that restrict restoring forced circulation, operator actions that affect pressure-temperature limits are critical.

Correct and timely operator response to the above conditions depends upon proper actions being specified in plant procedures and appropriate training being provided to operators for those actions.

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

Brian Grimes

Brian K. Grimes, Director
Division of Operating Reactors Support
Office of Nuclear Reactor Regulation

Technical contact: J. Carter, NRR
(301) 504-1153

Attachment:
List of Recently Issued NRC Information Notices

LIST OF RECENTLY ISSUED
NRC INFORMATION NOTICES

Information Notice No.	Subject	Date of Issuance	Issued to
93-61	Excessive Reactor Coolant Leakage Following A Seal Failure in A Reactor Coolant Pump or Reactor Recirculation Pump	08/09/93	All holders of OLs or CPs for nuclear power reactors
93-60	Reporting Fuel Cycle and Materials Events to the NRC Operations Center	08/04/93	All fuel cycle and material licensees.
93-59	Unexpected Opening of Both Doors in An Airlock	07/26/93	All holders of OLs or CPs for nuclear power reactors.
93-58	Nonconservatism in Low-Temperature Overpressure Protection for Pressurized-Water Reactors	07/26/93	All holders of OLs or CPs for pressurized-water reactors.
93-57	Software Problems Involving Digital Control Console Systems at Non-Power Reactors	07/23/93	All holders of OLs or CPs for test and research reactors and nuclear power reactors.
93-56	Weakness in Emergency Operating Procedures Found as Result of Steam Generator Tube Rupture	07/22/93	All holders of OLs or CPs for pressurized water reactors.
93-55	Potential Problem with Main Steamline Break Analysis for Main Steam Vaults/Tunnels	07/21/93	All holders of OLs or CPs for pressurized water reactors.
93-54	Motor-Operated Valve Actuator Thrust Variations Measured with A Torque Thrust Cell and A Strain Gage	07/20/93	All holders of OLs or CPs for nuclear power reactors.

OL = Operating License
CP = Construction Permit

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Once thermal stratification occurs, any rapid circulation of water could result in a large step change in the temperature of the water adjacent to the reactor bottom head penetrations. This step change may violate the technical specification limits for rate of temperature change. A temperature differential within the reactor vessel may be reduced by increasing coolant flow out of the bottom head drain and reducing cold water flow through the control rod drive system, which enters the bottom region of the reactor vessel.

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Original signed by
Brian K. Grimes

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*See previous concurrence

OFC	OEAB:DORS	SC/OEAB:DORS	PUB:ADM	C/OEAB:DORS
NAME	JCarter*	RDennig*	Tech Ed*	AChaffee*
DATE	06/07/93	06/15/93	04/22/93	06/18/93

OFC	PDII-3:ADR2	PDI-2:ADR1	*OGCB:DORS	C/OGCB:DORS	D/DORS
NAME	*KJabbour	JShea*	JBirmingham	*GMarcus	BGrimes
DATE	07/19/93	07/15/93	07/19/93	07/23/93	08/6/93

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Correct and timely operator response to the above conditions depends upon proper actions being specified in plant procedures and operators receiving appropriate training being provided to operators for those actions.

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Technical specifications contain thermal-pressure limitations to ensure the integrity and safe operation of the reactor coolant system.

Once thermal stratification exists, any rapid circulation of water, such as would occur with operation of a recirculation pump, could result in a step change in temperature of the water adjacent to the reactor bottom head penetrations. Recommended methods for reducing the temperature differential include increasing reactor coolant flow out through the bottom head drain and reducing the cold CRD flow that enters the lower region of reactor vessel. Another concern is that brittle fracture temperature limits must not be exceeded. Once temperature differences develop in the reactor vessel that prevent restoring forced circulation, operators must consciously take pressure-temperature considerations into account before they act. These actions are given in plant operating and surveillance procedures, and timely action is ensured by operator training.

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Issue coordinated with both PMs and SRIs. Comments incorporated, but their concurrence being held until after OGCB review/comments are incorporated.

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