UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

September 30, 1996

NRC GENERIC LETTER 96-06: ASSURANCE OF EQUIPMENT OPERABILITY AND CONTAINMENT INTEGRITY DURING DESIGN-BASIS ACCIDENT CONDITIONS

Addressees

All holders of operating licenses for nuclear power reactors, except for those licenses that have been amended to possession-only status.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this generic letter to

(1) notify addressees about safety-significant issues that could affect containment integrity and equipment operability during accident conditions,

(2) request that all addressees submit certain information relative to the issues that have been identified and implement actions as appropriate to address these issues, and

(3) require that all addressees submit a written response to the NRC relative to implementation of the requested actions.

Background

As a result of recent NRC inspection activities, licensee notifications, and event reports, several safety-significant issues have been identified that have generic implications and warrant action by the NRC to assure that these issues have been adequately addressed and resolved. In particular, the following issues are of concern:

(1) Cooling water systems serving the containment air coolers may be exposed to the hydrodynamic effects of waterhammer during either a loss-of-coolant accident (LOCA) or a main steamline break (MSLB). These cooling water systems were not designed to withstand the hydrodynamic effects of waterhammer and corrective actions may be needed to satisfy system design and operability requirements.

(2) Cooling water systems serving the containment air coolers may experience two-phase flow conditions during postulated LOCA and MSLB scenarios. The heat removal assumptions for design-basis accident scenarios were based on single-phase flow conditions. Corrective actions may be needed to satisfy system design and operability requirements.

(3) Thermally induced overpressurization of isolated water-filled piping sections in containment could jeopardize the ability of accident-mitigating systems to perform their safety functions and could also lead to a breach of containment integrity via bypass leakage. Corrective actions may be needed to satisfy system operability requirements.

The sections that follow contain additional background information about each of these issues.

Waterhammer

On February 13, 1996, the Pacific Gas and Electric Company (PG&E, the licensee for Diablo Canyon Units 1 and 2), determined that component cooling water, which is circulated through the containment air coolers, could flash to steam in the cooler unit cooling coils during a design-basis LOCA with a concurrent loss of offsite power (LOOP) or with a delayed sequencing of equipment. This condition was reported to the NRC in Licensee Event Report (LER) 1-96-005, dated April 26, 1996.

The Diablo Canyon units have five containment air coolers in each containment, these are typically used during normal plant operation to prevent excessive containment temperatures. The containment air coolers are also automatically initiated engineered safety features that are relied upon to help maintain containment integrity by performing their heat removal function during postulated accident conditions. The air coolers in the Diablo Canyon units transfer heat from the containment to the respective unit's component cooling water system (a closed-loop system).

PG&E reported that, during a postulated design-basis LOCA with a concurrent LOOP, the component cooling water pumps and the air cooler fans will temporarily lose power (an expected condition). The component cooling water flow stops almost immediately, while the fans coast down over a period of minutes. The first air cooler fan will restart on slow speed approximately 22 seconds after the LOOP and the component cooling water pumps will restart 4 to 8 seconds later. In this scenario, the high-temperature containment atmosphere will be forced across the containment air cooler's cooling coils for up to 30 seconds with no forced component cooling water flow through the coolers. PG&E determined that the stagnant component cooling water in the containment air coolers may boil and create a substantial steam volume in the component cooling water system. As the component cooling water pumps restart, the pumped liquid may rapidly condense this steam volume and produce a waterhammer. The hydrodynamic loads introduced by such a waterhammer event could be substantial, challenging the integrity and function of the containment air coolers and the associated component cooling water system, as well as posing a challenge to containment integrity. As corrective action, PG&E has installed a nitrogen pressurization system on the component cooling water head tank to increase the margin to boiling.

On June 20, 1996, Westinghouse Electric Corporation issued Nuclear Safety Advisory Letter NSAL-96-003, "Containment Fan Cooler Operation During a Design Basis Accident," to alert its customers to the potential safety issue that was

identified by PG&E (Westinghouse is the reactor vendor for the Diablo Canyon units). In NSAL-96-003, Westinghouse recommended that licensees review their containment cooling systems to determine if their safety-related containment air coolers are susceptible to waterhammer.

On July 22, 1996, the Connecticut Yankee Atomic Power Company (CYAPC, the licensee for the Haddam Neck nuclear power plant) declared all four of the containment air coolers at the Haddam Neck plant inoperable and initiated a plant shutdown in accordance with Technical Specification requirements. The containment air coolers at the Haddam Neck plant are the only components that are credited for post-accident containment heat removal, and station service water (an open-loop system) is the cooling medium for the containment air coolers. The containment air coolers were declared inoperable after CYAPC completed its review relative to Westinghouse NSAL-96-003. The licensee's analysis predicted hydrodynamic loads in the service water system from waterhammer that exceeded piping and support structural limits.

On August 12, 1996, the staff issued Information Notice (IN) 96-45, "Potential Common-Mode Post-Accident Failure of Containment Coolers," to alert addressees to the potential failure mode of the containment air coolers and their associated cooling water systems. IN 96-45 discussed the information that was reported by PG&E and CYAPC relative to the Diablo Canyon and Haddam Neck plants, respectively, and attached a copy of Westinghouse letter NSAL-96-003.

Two-Phase Flow in Safety-Related Piping and Components

In July 1996, the NRC issued Inspection Report 50-213/96-201, "Special Inspection of Engineering and Licensing Activities at Haddam Neck-Connecticut Yankee." Among other things, the report identified an issue relative to two-phase flow in the station service water system. The inspection team reviewed the service water system flow models, calculations, and operational data and found that some steam may be produced in the service water system as the service water flows through the containment air coolers during design-basis accident conditions. However, the licensee's service water system model and calculations only assumed single-phase flow conditions (liquid phase only) and did not consider two-phase flow conditions (both steam and liquid present). The licensee is currently evaluating the system to determine whether or not corrective actions are needed.

On July 23, 1996, the Wisconsin Electric Power Company submitted information regarding two-phase flow in the service water system at the Point Beach nuclear plant during a design-basis LOCA. The licensee's preliminary evaluations concluded that after the cooling water is heated via heat transfer from the containment air coolers, some steam could be formed at the air cooler outlet throttle valves. This two-phase mixture (steam and water) would result in a higher frictional pressure drop in the service water return piping and would ultimately affect the service water flow and the heat removal capabilities of the containment air coolers. Steam formation due to low pressure and high temperature in the service water system could reduce the service water flow rates through the containment air coolers to values below those needed to

satisfy design-basis heat removal requirements. The licensee is completing more detailed analyses to determine if immediate corrective action is warranted.

On August 20, 1996, the Public Service Electric and Gas Company (the licensee for Salem 1 and 2) notified the NRC of a condition that is not bounded by the existing design basis for the Salem nuclear power plants (EN 30900). The licensee reported that because the service water isolation valves for the nonsafety-related turbine loads do not start to close until approximately 30 seconds into the emergency loading sequence, the service water system may not be able to supply sufficient flow for the containment cooling function during accident conditions. The licensee determined that the initial heat transfer rates through the containment air coolers could result in additional "flow restrictions" in the air cooler tubes, further decreasing the flow of service water through the containment air coolers as a result of the higher frictional pressure drop caused by two-phase flow. At the time of the licensee's notification, the Salem units were shut down for refueling.

Overpressurization of Isolated Piping Sections

On July 3, 1996, Duquesne Light Company (the licensee for Beaver Valley Units 1 and 2) notified the NRC that during surveillance testing of a component cooling water inlet valve to the RHR heat exchanger on Unit 1, the motor-operated butterfly valve located inside the containment would not open (EN 30833). The licensee found that pressure in the piping section between this valve and a closed manual butterfly valve located outside the containment measured slightly higher than the system design pressure. After the pressure in this isolated section of piping was relieved by opening a drain valve, the remotely operated butterfly valve was opened without any trouble. The licensee concluded that pressure in the isolated section of piping increased when the trapped water was heated up by increased ambient temperatures. The section of piping was isolated in the spring when the unit was shut down and ambient temperatures were much lower than temperatures that existed in the summer after the plant was returned to power operation and ambient temperatures reached about 32 °C [90 °F].

On July 19, 1996, the Maine Yankee Atomic Power Company (MYAPC, licensee for the Maine Yankee nuclear plant) notified the NRC of a condition that was outside the plant design basis (EN 30769). The primary component cooling water (PCCW) system at the Maine Yankee plant has a nonsafety-related subdivision that serves the containment fan coolers (not needed for accident mitigation), and a safety-related subdivision that serves ECCS equipment. The nonsafety-related subdivision of PCCW has a swing-check valve at the containment inlet (supply) penetration, and an air-operated valve at the containment outlet (return) penetration. During a design-basis LOCA, the containment isolation logic initiates closure of the air-operated outlet valve, thereby stopping the flow of water. The licensee has determined that heat from the containment accident environment could cause the PCCW in the containment fan coolers between the inlet check valve and closed air-operated outlet valve to expand, rupturing this portion of the PCCW system. Water from the PCCW system is then able to flow through the supply check valve for the containment fan

coolers and out the rupture, rendering the PCCW system inoperable and jeopardizing safety-related equipment that is cooled by the safety-related division of the PCCW system. Upon recognizing this postulated scenario, the licensee promptly shut down the Maine Yankee plant. To correct this, the licensee plans to install a pressure relief valve on each of the six containment fan cooler PCCW branch lines downstream of the supply check valves.

On August 20, 1996, the staff issued Information Notice (IN) 96-49, "Thermally Induced Pressurization of Nuclear Power Facility Piping," to alert addressees to the potential for safety-related piping to become overpressurized during accident conditions. IN 96-49 discusses the information reported by Duquesne Light Company and MYAPC relative to Beaver Valley Unit 1 and the Maine Yankee plant, respectively.

Discussion

The issues discussed in this generic letter pertain to situations that may not be bounded by the applicable system design capabilities and for which corrective actions may be needed to satisfy equipment design and operability requirements. The sections that follow contain additional discussion about each of these issues.

Waterhammer

At many plants, containment air coolers satisfy a significant safety function by removing heat from the containment and reducing post-accident containment pressure. The hydrodynamic loads imposed by waterhammer can be substantial, challenging the integrity and function of the containment air coolers and the associated cooling water system, as well as posing a challenge to containment integrity. Waterhammer in cooling water systems associated with nonsafety-related containment air coolers can also challenge containment integrity by creating a containment bypass flow path, and interfacing safety-related systems can be affected. During this accident scenario, the steam that is produced in the containment air coolers may accumulate in other parts of the cooling water system, restricting flow as well as causing waterhammer damage. Plant vulnerability to the postulated waterhammer scenario depends on a number of factors, such as piping configuration, how long it takes for the flow of cooling water to stop, the coastdown rate of the fans in the containment fan coolers, the operating pressure and pressure decay rate of the cooling water system, how long it takes to establish forced cooling water flow, the containment temperature profile, and other site-specific parameters.

The postulated failure scenario is applicable to both LOCA and MSLB events that involve a loss of offsite power, a loss of cooling water flow to the containment air coolers (e.g., one train of cooling water inoperable), or the sequencing of equipment that can affect the containment cooling function. Steam formation and waterhammer in cooling water systems associated with safety-related and nonsafety-related containment air coolers may not require a loss of offsite power for this scenario to be valid.

Two-Phase Flow in Safety-Related Piping and Components

Two-phase flow (i.e., both steam and liquid) in cooling water systems associated with the containment air coolers can significantly interfere with the ability of the containment air coolers to remove heat under design-basis accident conditions, and can interfere with the cooling of other safety-related components. These cooling water systems were designed assuming single-phase flow conditions (i.e., liquid only) and containment heat transfer analyses are based on this assumption. Two-phase flow is a much more complex situation to deal with analytically than single-phase flow and involves additional hydrodynamic loading considerations as well as flow, heat transfer, systems interaction and erosion considerations. Additionally, the steam that is formed during two-phase flow can accumulate in the cooling water system, restricting flow and resulting in a waterhammer as discussed above.

Overpressurization of Isolated Piping

Because of its thermal expansion, water heated while it is trapped in isolated piping sections is capable of producing extremely high pressures. This phenomenon is typically a design consideration. Piping design codes as far back as U.S.A. Standard (USAS) B31.1 (1967), have explicitly recognized the need to consider the effects of heating fluid that is trapped in an isolated section of piping. The potential for thermally induced expansion of fluid trapped in valve bonnets was one reason for issuing Generic Letter (GL) 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves." In addition, several information notices (INs) have been issued discussing the pressurization of water trapped in valve bonnets, including IN 95-14, "Susceptibility of Containment Sump Recirculation Gate Valves to Pressure Locking," IN 95-18, "Potential Pressure-Locking of Safety-Related Power-Operated Gate Valves," IN 95-30, "Susceptibility of LPCI and Core Spray Injection Valves to Pressure Locking," and IN 96-08, "Thermally Induced Pressure Locking of a HPCI Gate Valve."

The potential for systems to fail to perform their safety functions as a result of thermally induced overpressurization is dependent on many factors. These factors include leak tightness of valve seats, bonnets, packing glands and flange gaskets; piping and component material properties, location and geometry; ambient and post-accident temperature response; pipe fracture mechanisms; heat transfer mechanisms; relief valves and their settings; and system isolation logic and setpoints. Engineering design and modification evaluations, which include systematic evaluation of heat input to systems and components with consideration of factors such as those just noted, can detect conditions which may influence system operability under normal operating, transient, and accident conditions.

Under the "single-failure concept," failure due to overpressurization does not preclude consideration of additional active and passive failures in the same and other systems in evaluating plant response to a postulated accident. If relief valves are installed to prevent overpressure conditions, consideration must be given to the effects of stuck-open relief valves and associated environmental flooding and radiation hazards.

Requested Action(s)

Addressees are requested to determine:

(1) if containment air cooler cooling water systems are susceptible to either waterhammer or two-phase flow conditions during postulated accident conditions;

(2) if piping systems that penetrate the containment are susceptible to thermal expansion of fluid so that overpressurization of piping could occur.

In addition to the individual addressee's postulated accident conditions, these items should be reviewed with respect to the scenarios referenced in the generic letter.

With regard to waterhammer, addressees may find Volumes 1 and 2 of NUREG/CR-5220, "Diagnosis of Condensation-Induced Waterhammer," dated October 1988, informative and useful in evaluating potential waterhammer conditions.

If systems are found to be susceptible to the conditions discussed in this generic letter, addressees are expected to assess the operability of affected systems and take corrective action as appropriate in accordance with the requirements stated in 10 CFR Part 50 Appendix B and as required by the plant Technical Specifications. GL 91-18, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability," dated November 7, 1991, contains guidance on the review of licensee operability determinations and licensee resolution of degraded and nonconforming conditions.

Requested Information

Within 120 days of the date of this generic letter, addressees are requested to submit a written summary report stating actions taken in response to the requested actions noted above, conclusions that were reached relative to susceptibility for waterhammer and two-phase flow in the containment air cooler cooling water system and overpressurization of piping that penetrates containment, the basis for continued operability of affected systems and components as applicable, and corrective actions that were implemented or are planned to be implemented. If systems were found to be susceptible to the conditions that are discussed in this generic letter, identify the systems affected and describe the specific circumstances involved.

Required Response

Within 30 days of the date of this generic letter, addressees are required to submit a written response indicating: (1) whether or not the requested actions will be completed, (2) whether or not the requested information will be submitted and (3) whether or not the requested information will be submitted within the requested time period. Addressees who choose not to

complete the requested actions, or choose not to submit the requested information, or are unable to satisfy the requested completion date, must describe in their response any alternative course of action that is proposed to be taken, including the basis for establishing the acceptability of the proposed alternative course of action and the basis for continued operability of affected systems and components as applicable.

Address the required written reports to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, under oath or affirmation, under the provisions of Section 182a, Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). In addition, send a copy to the appropriate regional administrator.

Backfit Discussion

Title 10 of the Code of Federal Regulations (10 CFR) Part 50 (Appendix A) and plant licensing safety analyses require and/or commit that the addressees design safety-related components and systems to offer adequate assurance that those systems can perform their safety functions. Specifically, 10 CFR Part 50, (Appendix A, Criterion 38) specifies a "system to remove heat from the reactor containment. The safety function of this system is to rapidly reduce pressure and temperature in the containment following any loss-of-coolant accident and to maintain them at acceptably low levels." Additionally, Criterion 44 of Appendix A specifies a "system to transfer heat from structures, systems, and components important to safety. The system safety function shall be to transfer the combined heat load of these structures, systems and components under normal operating and accident conditions." The heat load values as defined in final safety analysis reports are based on single-phase flow assumptions for the containment air cooler cooling water systems. The potential for waterhammer and two-phase flow raises concerns that these systems will not meet their design-basis requirements as specified in 10 CFR Appendix A, Criteria 38 and 44. Further, 10 CFR Part 50 Appendix A, Criteria 1 and 4 specify that safety-related systems be designed to offer adequate assurance that those systems can perform their safety functions under accident conditions. Accordingly, licensees are required to ensure that the containment air coolers and their associated cooling water systems that may be affected by waterhammer or by two-phase flow are capable of performing their required safety functions and that containment integrity will be maintained.

Licensees are also required either by their commitment to USAS B31.1 or the American Society of Mechanical Engineers (ASME) Code for piping design or by virtue of 10 CFR 50.55a, which endorses various editions of the ASME Boiler and Pressure Vessel Code, to comply with design criteria which specify that piping systems which have the potential to experience pressurization due to trapped fluid expansion shall either be designed to withstand the increased pressure or shall have provisions for relieving the excess pressure. The potential for overpressurization raises concerns that these piping systems will not meet their design code criteria.

The actions requested in this generic letter are considered compliance backfits under the provisions of 10 CFR 50.109 and existing NRC procedures to

ensure that containment integrity will be maintained and that safety-related components and piping systems are capable of performing their intended safety functions and satisfying their licensing-basis code criteria, respectively; and that containment integrity and these safety-related piping systems and components will not be adversely affected by the occurrence of waterhammer, two-phase flow, or thermal overpressurization that may occur in safety-related and nonsafety-related systems that penetrate containment. In accordance with the provisions of 10 CFR 50.109 regarding compliance backfits, a full backfit analysis was not performed for this proposed action; but the staff performed a documented evaluation which stated the objectives of and reasons for the requested actions and the basis for invoking the compliance exception. See also 10 CFR 50.54(f) . A copy of this evaluation will be placed in the NRC Public Document Room.

Federal Register Notification

A notice of opportunity for public comment was not published in the Federal Register because of the urgent nature of the generic letter. However, comments on the actions requested and the technical issues addressed by this generic letter may be sent to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001.

Paperwork Reduction Act Statement

This generic letter contains information collections that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These information collections were approved by the Office of Management and Budget, approval number 3150-0011, which expires on July 31, 1997.

The public reporting burden for this collection of information is estimated to average 300 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the collection of information. The U.S. Nuclear Regulatory Commission is seeking public comment on the potential impact of the collection of information contained in the generic letter and on the following issues:

(1) Is the proposed collection of information necessary for the proper performance of the functions of the NRC, including whether the information will have practical utility?

(2) Is the estimate of burden accurate?

(3) Is there a way to enhance the quality, utility, and clarity of the information to be collected?

(4) How can the burden of the collection of information be minimized, including the use of automated collection techniques?

Send comments on any aspect of this collection of information, including suggestions for reducing this burden, to the Information and Records Management Branch, T-6F33, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0011), Office of Management and Budget, Washington, D.C. 20503.

The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid OMB control number.

If you have any questions about this matter, please contact one of the technical contacts listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

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Attachment: List of Recently Issued NRC Generic Letters

LIST OF RECENTLY ISSUED GENERIC LETTERS

Generic Letter	Subject	Date of Issuance	Issued To
96-05	PERIODIC VERIFICATION OF DESIGN-BASIS CAPABILITY OF SAFETY-RELATED MOTOR-OPERATED VALVES	09/18/96	ALL HOLDERS OF OLs (EXCEPT THOSE LICENSES THAT HAVE BEEN AMENDED TO POSSESSION-ONLY STATUS) OR CPs FOR NPRs
96-04	BORAFLEX DEGRADATION IN SPENT FUEL POOL STORAGE RACKS	06/26/96	ALL HOLDERS OF OLs FOR NPRs
95-09, SUPP. 1	MONITORING AND TRAINING OF SHIPPERS AND CARRIERS OF RADIOACTIVE MATERIALS	04/05/96	ALL U.S. NUCLEAR REGULATORY COMMISSION LICENSEES
96-03	RELOCATION OF THE PRESSURE TEMPERATURE LIMIT CURVES AND LOW TEMPERATURE OVER-PRESSURE PROTECTION SYSTEM LIMITS	01/31/96	ALL HOLDERS OF OLs OR CPs FOR NPRs
96-02	RECONSIDERATION OF NUCLEAR POWER PLANT SECURITY REQUIREMENTS ASSOCIATED WITH AN INTERNAL THREAT	01/31/96	ALL HOLDERS OF OLs OR CPs FOR NPRs
89-10, Supp. 7	CONSIDERATION OF VALVE MISPOSITIONING IN PRESSURIZED-WATER REACTORS	01/24/96	ALL HOLDERS OF OLs (EXCEPT THOSE LICENSES THAT HAVE BEEN AMENDED TO A POSSESSION ONLY STATUS) OR CPs FOR NPRs
96-01	TESTING OF SAFETY-RELATED LOGIC CIRCUITS	01/10/96	ALL HOLDERS OF OLs OR CPs FOR NPRs
95-10	RELOCATION OF SELECTED TECHNICAL SPECIFICATIONS REQUIREMENTS RELATED TO INSTRUMENTATION	12/15/95	ALL HOLDERS OF OLs OR CPs FOR NPRs

OL = OPERATING LICENSE
CP = CONSTRUCTION PERMIT
NPR = NUCLEAR POWER REACTORS

Send comments on any aspect of this collection of information, including suggestions for reducing this burden, to the Information and Records Management Branch, T-6F33, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0011), Office of Management and Budget, Washington, D.C. 20503.

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original signed by B.K. Grimes

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Attachment: List of Recently Issued NRC Generic Letters

Attachment filed in Jacket

Tech Editor has reviewed and concurred on 08/30/96
*SEE PREVIOUS CONCURRENCES

DOCUMENT NAME: 96-06.GL

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