July 24, 1981

Docket No. 50-133

Pacific Gas and Electric Company
P. O. Box 7442
San Francisco, California 94106

Attention: Mr. Philip A. Crane, Jr.
Assistant General Counsel

Gentlemen:

The enclosed circular is forwarded to you for appropriate action. No written response to this circular is required. If you have any questions related to this matter, please contact this office.

Sincerely,

R.C. Haynes

FOR/ R. H. Engelken
Director

Enclosure:
IE Circular No. 81-11

cc w/enclosure:
J. D. Shiffer, PG&E
E. Weeks, PG&E (Humboldt Bay)
W. A. Raymond, PG&E

OFFICE	RV/dot						
SURNAME	CREWS	ENGELKEN					
DATE	7/24/81	7/24/81					

SSINS No.: 6830
Accession No.:
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IEC 81-11

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

July 24, 1981

IE Circular No. 81-11: INADEQUATE DECAY HEAT REMOVAL DURING REACTOR SHUTDOWN

Background:

Following several losses of decay heat removal capability at operating pressurized water reactors (PWRs), IE Bulletin 80-12 "Decay Heat Removal System Operability" (issued May 1, 1980) requested PWR licensees to take certain actions intended to reduce the probability of loss of decay heat removal. All operating PWRs were requested to amend the Technical Specifications for their facilities with respect to reactor decay heat removal capability by letter from D. Eisenhut, Division of Licensing, on June 11, 1980. IE Bulletin 80-12 was issued to boiling water reactor (BWR) licensees for information with the expectation that the information would be evaluated for applicability and subsequent action taken as determined necessary. However, events involving inadequate decay heat removal at operating BWRs now indicate the need for BWR licensees to provide additional controls related to decay heat removal.

Description of Circumstances:

1. Brunswick - Temporary Loss of Shutdown Cooling

   On December 8, 1980, unplanned heatup of the reactor coolant occurred at Brunswick Unit 2 when the unit was in cold shutdown (<212°F) with all rods inserted. The heatup occurred while the service water cooling for the "A" loop of the residual heat removal (RHR) system was isolated longer than expected for repair of a service water leak. Shutdown cooling was not lined up to loop "B" (1) because it was expected that loop "A" would be returned to service before 212°F was reached and (2) because of the length of time required to line up the "B" loop for operation. During the repair, the recirculation pumps were off, an RHR pump was running, and the control rod drive pump was supplying water to the reactor pressure vessel (RPV) while the reactor water cleanup (CU) system was rejecting water for level control. The reactor coolant temperature monitored at the CU inlet (from a recirculation loop) indicated <212°F during the repair. The reactor head vents were reported to be opened during this period, with no evidence of steaming. However, average coolant temperature at the time of completion of repair approached 212°F with an observed maximum of 217°F. Shutdown cooling was initiated and primary coolant temperature decreased to a normal temperature within approximately 30 minutes. Primary containment could not be quickly established due to cables going through the personal access hatch and the torus hatch being removed.

   A similar event occurred at Brunswick Unit 2 on the following day. With the primary containment and reactor head vents reported open, the conventional and nuclear service water systems were secured to repair a conventional service water pump discharge check valve. The primary coolant temperature initially was less than 120°F. Approximately two hours after the service water systems were secured, the RHR pumps in the A loop were secured to reduce coolant heat input from the pumps.

Repairs took longer than anticipated, and when the conventional and nuclear service water systems were returned to service, the primary coolant temperature at the vessel bottom head drain was 147°F. Approximately fifteen minutes later shutdown cooling was initiated using the B loop of the RHR. There were indications of heatup of the coolant to approximately 212°F; however, there was no evidence of steaming through the open reactor heat vents. Primary coolant temperature decreased to a normal temperature within approximately three hours.

2. Dresden Unit 3 - Unplanned Repressurization

On December 20, 1980, the Dresden Nuclear Power Station Unit 3 was in the cold shutdown condition. Numerous maintenance and modification outages were in progress which resulted in the shutdown and/or isolation of all systems which communicate with the reactor vessel, and which normally provide cooling and recirculation of the primary coolant. Subsequently, one of three loops of the shutdown cooling system (SDC) was put in service to maintain reactor water temperature at approximately 150°F. The reactor water level was maintained at the normal operating level (instead of flooding up) to limit vessel safe end thermal stresses.

Because the design of the SDC does not allow for throttling of the cooling water flow to the SDC heat exchangers, it is standard practice to throttle SDC flow to the recirculation loop to maintain vessel temperature when in cold shutdown. As the decay heat load decreased the unit operators reduced SDC flow until insufficient vessel flow existed to provide mixing of the primary coolant, and accurate temperature measurements by the recirculation pump and SDC pump suction temperature instruments. Because the operators monitored only the recirculation pump and SDC temperatures, a slow heatup and repressurization of the reactor vessel to 175 psig occurred over a six hour period of time.

Upon discovering the repressurization, SDC flow was increased, and a second SDC loop was placed in service to expedite the return to cold shutdown. The indicated recirculation suction temperature rose to approximately 225°F, indicating that the entire vessel contents did not heat up to the saturation temperature at 175 psig (377°F).

During the repressurization event the containment personnel access doors were open, resulting in violation of the Technical Specification limiting condition for operation for primary containment integrity. Had the Technical Specification been revised to conform to current BWR standard Technical Specifications the LCO's for the High pressure coolant injection system and isolation condenser systems would also have been exceeded.

Post event evaluations of the circumstances leading up to the repressurization, and the chronology of the event itself, establish that the licensee did not evaluate the potential for adverse effects on plant safety resulting from procedure changes removing the vessel floodup requirement, and the effect of removing from service those systems which normally cool and recirculate the reactor coolant. The potential for inaccurate response of normally used instrumentation was apparently not considered by the licensee, and redundant instrumentation which could have provided warning that the event was in progress was not utilized by operations personnel.

The licensees of the above facilities have committed to make administrative and procedural changes to provide personnel additional guidance when operating in the shutdown cooling mode. Additional information regarding these events and corrective actions is contained in LERs 2-80-107, 2-80-112 (Brunswick 2), and LER 80-047/01T-0 (Dresden 3).

There have been recent events at other BWRs involving the loss of systems providing normal decay heat removal, and appropriate action has been taken by operating personnel to put alternate cooling in service. These events indicate the need for timely operator response and the need to have backup systems available.

Recommended Action for Licensees of BWRs with an Operating License:

1. Review your existing procedures and administrative controls that relate to decay heat removal during reactor shutdown. Analyze these procedures for adequacy of monitoring and responding to events involving lost or degraded decay heat removal. Special emphasis should be placed on conditions involving low core recirculation or cooling flow, or when maintenance or refueling activities degrade the decay heat removal capability.

2. Administrative controls should provide the following:

   a. Assure that redundant or diverse decay heat removal methods are available during all modes of plant operation. (Note: When in a refueling mode with water in the refueling cavity and the head removed, an acceptable means could include one decay heat removal train and a readily accessible source of water to replenish any loss of inventory). (Note: Only one power source needs to be operable in order to consider the decay heat removal system operable while in modes 4 and 5).

   b. For those cases where single failures or other actions result in only one decay heat removal train being available, provide an additional alternate means of decay heat removal or provide an expeditious means for the restoration of the lost train or method.

   c. Implement administrative controls during periods of low flow or no flow to ensure that the maximum coolant temperature remains below the saturation temperature. Consideration should be given to maintaining water level in the reactor vessel sufficiently high to enable natural circulation at all times.

d. Require monitoring of the reactor coolant temperature and pressure at a specified frequency.

3. Any changes needed in the existing procedures or administrative controls as a result of Items 1 and 2 above should be implemented within 120 days of the date of this circular.

No written response to this circular is required. If you need additional information regarding this subject, please contact the appropriate Regional Office.

Attachment:
Recently issued IE Circulars

Attachment
IEC 81-11
July 24, 1981

RECENTLY ISSUED
IE CIRCULARS

Circular No.	Subject	Date of Issue	Issued to
81-12	Inadequate Periodic Test Procedure of PWR Protection System	7/22/81	All power reactor facilities with an OL or CP
81-10	Steam Voiding in the Reactor Coolant System During Decay Heat Removal Cooldown	7/2/81	All power reactor facilities with an OL or CP
81-09	Containment Effluent Water That Bypasses Radioactivity Monitor	7/10/81	All power reactor facilities with and OL
81-08	Foundation Materials	5/29/81	All power reactor facilities with an OL or CP
81-07	Control of Radioactively Contaminated Material	5/14/81	All power reactor facilities with an OL or CP
81-06	Potential Deficiency Affecting Certain Foxboro 20 to 50 Milliampere Transmitters	4/14/81	All power reactor facilities with an OL or CP
81-05	Self-Aligning Rod End Bushings for Pipe Supports	3/31/81	All power reactor facilities with an OL or CP
81-04	The Role of Shift Technical Advisors and Importance of Reporting Operational Events	4/30/81	All power reactor facilities with an OL or near-term OL
81-03	Inoperable Seismic Monitoring Instrumentation	3/2/81	All power reactor facilities with an OL or CP
81-02	Performance of NRC-Licensed Individuals While on Duty	2/9/81	All power reactor facilities (research & test) with an OL or CP

OL = Operating Licenses
CP = Construction Permit