IN 87-65

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

December 23, 1987

NRC INFORMATION NOTICE NO. 87-65: PLANT OPERATION BEYOND ANALYZED CONDITIONS

Addressees:

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

Purpose:

This information notice is being provided to alert addressees to potential problems resulting from operating a plant beyond its analyzed basis. The safety concerns of the particular circumstances described in this information notice are high temperature inside containment and insufficient post-LOCA cooling of safety systems. 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 do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

Arkansas 1 (ANO-1). During normal operation on August 7, 1987, it was found that the containment temperatures were significantly higher than the temperatures assumed in the accident analyses in the final safety analysis report (FSAR, including updates) and equipment qualification program. In the FSAR, a design temperature of 110°F was assumed for safety analysis of containment integrity and 120°F was assumed for equipment qualification during normal service life. Measured temperatures ranged from 103°F to 165°F with one local "hot spot" of 183°F about the "A" steam generator. The licensee had observed such temperatures since plant startup in 1974.

Crystal River 3. During an inspection, it was found that the temperature of the ultimate heat sink (UHS), the Gulf of Mexico, was above the value of 85°F assumed in the FSAR analysis for heat removal capability after a loss-of-coolant accident (LOCA). The Technical Specifications (TS) permit a UHS temperature of 105°F. The plant has been operating within the TS limit but beyond the design-basis temperature of 85°F assumed in the accident analysis.

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Discussion:

Because ANO-1 had been operating at elevated containment temperatures for extended periods, the NRC staff had several concerns:

1. The plant had been operating beyond its analyzed basis with regard to post-accident (LOCA) containment performance because the initial conditions assumed in the analysis were exceeded.

2. The higher temperature implies accelerated aging of equipment required for post-accident safe shutdown in accordance with regulation 10 CFR 50.49 on equipment qualification.

3. The higher temperature may cause deterioration of the concrete structure.

In response to the NRC staff concerns, the licensee submitted an analysis of the safety implications of the elevated containment temperatures and identified both near term and long term actions to justify continued operation.

In general, the FSAR contains design bases, operational limits, and analyses of structures, systems, and components for ensuring the safety of the facility. It is a statement by the applicant/licensee of how it intends to comply with NRC requirements. This statement is reviewed by the NRC to form the bases for the operating license. The analysis of containment performance following a design-basis accident (for example, a LOCA) depends on certain assumed initial conditions. Exceeding these conditions may invalidate the analysis and thereby raise concerns regarding the maintenance of containment integrity following an accident.

In accordance with the "10°C rule," which may be used to calculate qualified life, an increase of 10°C (18°F) over the initially assumed temperature reduces the qualified life by 50 percent. Under these circumstances, equipment that is relied on in the event of a design-basis accident may not reliably perform its safety function when required.

In the case of Crystal River 3, the concern was consistency between the FSAR and the TS. Regulation 10 CFR 50.36 requires that the TS be derived from the analyses in the safety analysis report. Since the plant has been operating beyond the assumed design-basis temperature for the UHS, the adequate transfer of post-accident heat loads from safety-related structures, systems, and components was in question.

LIST OF RECENTLY ISSUED
NRC INFORMATION NOTICES 1987

Information Notice No.	Subject	Date of Issuance	Issued to
87-64	Conviction for Falsification of Security Training Records	12/22/87	All nuclear power reactor facilities holding an OL or CP and all major fuel facility licensees.
87-35, Supp. 1	Reactor Trip Breaker Westinghouse Model DS-416, Failed to Open on Manual Initiation From the Control Room	12/16/87	All holders of OLs or CPs for nuclear power reactors.
87-63	Inadequate Net Positive Suction Head in Low Pressure Safety Systems	12/9/87	All holders of OLs or CPs for nuclear power reactors.
87-62	Mechanical Failure of Indicating-Type Fuses	12/8/87	All holders of OLs or CPs for nuclear power reactors.
87-61	Failure of Westinghouse W-2-Type Circuit Breaker Cell Switches.	12/7/87	All holders of OLs or CPs for nuclear power reactors.
87-60	Depressurization of Reactor Coolant Systems in Pressurized-Water Reactors	12/4/87	All holders of OLs or CPs for PWRs.
86-108, Supp. 2	Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion	11/19/87	All holders of OLs or CPs for nuclear power reactors.
87-59	Potential RHR Pump Loss	11/17/87	All holders of OLs or CPs for nuclear power reactors.

OL = Operating License
CP = Construction Permit

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Charles E. Rossi

Charles E. Rossi, Director
Division of Operational Events Assessment
Office of Nuclear Reactor Regulation

Technical Contacts: C. Li, NRR
(301) 492-9414

Vern Hodge, NRR
(301) 492-8196

Attachment: List of Recently Issued NRC Information Notices

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The draft of this information notice was transmitted to DOEA by DEST in a memorandum from L. Shao dated 10/26/87.

*SEE PREVIOUS CONCURRENCES

*OGCB:DOEA:NRR	*PPMB:ARM	*SAD/DEST:NRR	*C/OGCB:DOEA:NRR	D/DOEA:NRR
CVHodge	TechEd	AThadani	CHBerlinger	CERossi
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