UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 13, 1997

Mr. David A. Lochbaum
Union of Concerned Scientists
1616 P Street, NW., Suite 310
Washington, DC 20036-1495

50-259

Dear Mr. Lochbaum:

Your letter to the Chairman of the U.S. Nuclear Regulatory Commission (NRC), dated May 2, 1997, in which you requested information pertaining to problems with boiling water reactor (BWR) control rod rapid insertion systems, usually referred to as "scram" systems, has been referred to me for response. In your letter you requested copies of all available documentation regarding concerns expressed in a memorandum dated July 11, 1973, by S. H. Hanauer of the Atomic Energy Commission; copies of all available analyses produced by, or on behalf of, the NRC regarding the safety implications of a partial failure to scram at the Browns Ferry Nuclear Plant (BFN) Unit 3 on June 28, 1980; and a copy of an analysis of a postulated generator load rejection from rated power with incomplete control rod insertion for BFN by the General Electric Company (GE), and the NRC's assessment of that analysis.

SAFETY CONCERNS ASSOCIATED WITH PIPE BREAKS AFFECTING THE BWR SCRAM SYSTEM

The July 11, 1973, Hanauer memorandum expresses a concern that a pipe rupture could damage nearby scram discharge piping, thus interfering with the ability of affected control rods to insert into the reactor. This scenario differs from the BFN Unit 3 event of June 28, 1980, which was caused by an undetected accumulation of water in one of the two scram discharge volumes. The staff did not locate documents addressing the Hanauer memorandum in the 1973 time-frame. However, the NRC has determined that the scenario given in the Hanauer memorandum has been adequately addressed as part of BWR design. A summary of the basis for this conclusion follows.

Appendix A of Title 10 of the Code of Federal Regulations provides the general design criteria for nuclear power plants. General Design Criterion (GDC) 4, "Environmental and Dynamic Effects Design Basis," requires that important safety systems be protected against the dynamic effects of pipe ruptures, or that it be demonstrated that the probability of pipe rupture which could affect a safety system is extremely low. The requirements of GDC 4 were incorporated into NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 3.6.2, and were implemented for plants licensed using NUREG-0800. Assessment of the effects of pipe breaks on structures, systems, and components was incorporated in the Systematic Evaluation Program for selected older facilities as Topics III-5.A and III-5.B.

GDC 4 is also applied to the scram discharge piping. On April 10, 1981, the NRC staff requested plant-specific information addressing concerns associated with scram system pipe breaks. The NRC designated this issue as Generic Safety Issue (GSI) 40. Subsequently, Generic Letter (GL) 81-34 and GL 81-35 (Enclosures 1 and 2) were sent to BWR licensees and applicants, respectively.







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stating that responses conforming to the guidance contained in NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping" (Enclosure 3), would satisfy the information request of April 10, 1981.

The staff's generic safety evaluation for this issue was transmitted to all BWR applicants and licensees on January 6, 1986. This evaluation concluded that breaks in scram discharge piping need not be postulated.

As discussed above, the NRC requires that licensees design or evaluate piping systems in a manner which precludes a pipe rupture from affecting the scram discharge piping. Therefore, the scenario postulated in the Hanauer memorandum is beyond the licensing basis for the scram discharge piping system, since GDC 4 requires this system to be appropriately protected against adverse effects from piping ruptures.

BWR SCRAM DISCHARGE VOLUME SYSTEMS

The NRC conducted extensive assessments of the safety implications of the BFN Unit 3 event of June 28, 1980. The NRC designated the safety implications of the partial failure to scram at BFN Unit 3 on June 28, 1980, as GSI 41. Inspection and Enforcement (IE) Bulletin 80-17 and three supplements thereto were issued to inform licensees of the deficiencies identified, and to request that corrective actions be developed and implemented.

Shortly after the event, the NRC Office for Analysis and Evaluation of Operational Data (AEOD) initiated an independent study, including the BFN Unit 3 scram system design and operation and the special scram tests and inspections that were performed at the plant site. The principal purpose of this study was to provide an independent assessment of the event, to determine the lessons learned, and to recommend corrective actions to prevent recurrence. The AEOD review focused, for the most part, on the scram system design and the adequacy of the design features that protect against loss of scram capability.

The first AEOD assessment (AEOD Report C001, Enclosure 4) of the BFN Unit 3 partial failure to scram concluded that the problem was caused by the presence of water in the east scram discharge header. The analysis of the scram discharge volume (SDV) and scram discharge volume instrument volume (SDVIV) design configuration, together with its vent and drain characteristics, led AEOD to conclude that several actual and postulated mechanisms existed that could cause the SDV to fill undetected and without protection against such filling.

In the second study (AEOD Report C002, Enclosure 5), AEOD evaluated the procedures and equipment at BFN Units 1, 2, and 3 to determine their adequacy in providing assurance that the SDV will not fill with water and interfere with a successful scram. This study found the instrumentation and procedures in place after the BFN Unit 3 event to respond to the loss of control air scenario to be inadequate. However, AEOD concluded that interim surveillance efforts to detect the presence of water in the SDV, described in IE Bulletin 80-17, were adequate for continued interim operation if this study's recommendations on degraded control air pressure were implemented. IE

Bulletin 80-17, Supplement 3, was issued in response to the concerns raised by AEOD about degraded air pressure in the control air system as a mechanism that could rapidly fill the SDV.

In addition, shortly before the BFN Unit 3 event, the NRC had issued IE Bulletin 80-14 regarding issues with scram discharge volume instrumentation.

On December 9, 1980, the NRC issued the staff's generic Safety Evaluation Report (SER) for the BWR Scram Discharge System to address concerns raised by IE Bulletins 80-14 and 80-17. This SER (Enclosure 6) includes a detailed discussion of the cause of the BFN Unit 3 event and subsequent corrective actions.

Licensees were requested to take short-term actions to ensure the continuous safe operation with inadequate SDV/SDVIV hydraulic coupling until permanent design changes were made. The NRC staff's evaluation of each licensee's short-term actions is discussed in Appendix B to the generic SER.

The long-term action plan involved the evaluation of the scram system against criteria based on functional, safety, operational, design, and surveillance requirements for the system. These criteria were developed by the BWR Owners Subgroup, with the NRC staff's additional requirement to address potential common-cause failures of scram level instrumentation.

Completion of licensees' actions to address GSI 41 was documented in NUREG-1435, "Status of Safety Issues at Licensed Power Plants," Volume 3 (Enclosure 7), for most BWRs. NUREG-1435 notes that implementation of corrective actions was incomplete only for BFN Unit 3. Implementation of corrective actions for BFN Unit 3 for GSI 41 was documented in TVA's October 27, 1995, letter prior to that unit's restart from its extended recovery outage.

ANTICIPATED TRANSIENT WITHOUT SCRAM - ANALYSIS OF GENERATOR LOAD REJECTION

Your questions regarding analysis of a Browns Ferry Unit 3 generator load rejection with a failure to scram fall within the scope of the NRC's actions taken to address an anticipated transient without scram (ATWS). An ATWS is an expected operational transient, such as a loss of feedwater, a loss of condenser vacuum, or generator load rejection, accompanied by a failure of the reactor trip system to shut down the reactor. The NRC staff concluded that, under some conditions, core damage and release of radioactivity could result from an ATWS event unless additional mitigation features were added.

During the 1970s, ATWS and the manner in which this potential phenomenon should be considered in the design of nuclear power plants were discussed extensively by the NRC and the nuclear industry. The NRC published NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors, Staff Report," to summarize technical considerations related to ATWS. The NRC staff's technical findings for the ATWS issue were reported in Volume 4 of NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors, Resolution of Unresolved Safety Issue [USI] TAP A-9," which includes a summary of analyses performed by GE for a variety of ATWS events for each BWR product line. The excerpt from NUREG-0460, Volume 4 (Enclosure 8), is provided to address your interest in analysis of generator load rejection with a failure to scram. The staff did not locate a BFN Unit 3 plant-specific analysis of a generator load rejection with a partial failure to scram similar to the BFN Unit 3 event of June 28, 1980; however, the results presented in NUREG-0460 for the turbine trip without bypass bounds this case. Therefore, the staff does not plan to request additional information from GE on this topic.

USI A-9 was resolved in June 1984 with the publication of 10 CFR 50.62, which specified improvements needed to reduce the likelihood that the reactor protection system would fail to shut down the reactor following anticipated transients, improvements to mitigate the consequences of an ATWS event, and an implementation schedule. For BWRs, this rule required an alternate rod injection system, a standby liquid control system, and an automatic recirculation pump trip for conditions indicative of an ATWS.

I trust this information is responsive to your request.

Sincerely,

Samuel J. Collins

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Generic Letter 81-34
2. Generic Letter 81-35
3. NUREG-0803
4. AEOD Report C001
5. AEOD Report C002
6. Generic Safety Evaluation Report
7. NUREG-1435, Volume 3
8. NUREG-0460, Volume 4 (Excerpt)

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Original signed by
Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Generic Letter 81-34
2. Generic Letter 81-35
3. NUREG-0803
4. AEOD Report C001
5. AEOD Report C002
6. Generic Safety Evaluation Report
7. NUREG-1435, Volume 3
8. NUREG-0460, Volume 4 (Excerpt)

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Union of Concerned Scientists

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Office Director

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