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

IE Bulletin No. 79-12
Date: May 31, 1979

SHORT PERIOD SCRAMS AT BWR FACILITIES

Summary:

Reactor scrams, resulting from periods of less than 5 seconds, have occurred recently at three BWR facilities. In each case the scram was caused by high flux detected by the IRM neutron monitors during an approach to critical. These events are similar in most respects to events which were previously described by IE Circular 77-07 (copy enclosed). The recent recurrences of this event indicate an apparent loss of effectiveness of the earlier Circular. Issuance of this Bulletin is considered appropriate to further reduce the number of challenges to the reactor protective system high IRM flux scram.

Description of Circumstances:
The following is a brief account of each event.

1. Oyster Creek - On December 14, 1978, the reactor experienced a scram as control rods were being withdrawn for approach to critical, following a scram from full power which had occurred about 15 hours earlier. The moderator temperature was 380 degrees F and the reactor pressure was 190 psig. Because of the high xenon concentration the operators had not made an accurate estimate of the critical rod pattern. The operator at the controls was using the SRM count rate, which had changed only slightly, (425 to 450 cps) to guide the approach. Control rod 10-43 (first rod in Group 9) was being withdrawn in "notch override" to notch position 10, when the reactor became critical on an estimated 2.8 second period. The operator was attempting to reinsert the rod when the scram occurred. Failure of the "emergency rod in" switch to maintain contact, due to a bent switch stop, apparently contributed to the problem.

2. Browns Ferry Unit 1 - On January 18, 1979, the reactor experienced a scram during the initial approach to critical following refueling. The operator was continuously withdrawing in "notch override" the first control rod in Group 3 (a high worth rod) because the SRM count rate had led him to believe that the reactor was very subcritical. A short reactor period, estimated at 5 seconds, was experienced. The operator was attempting to reinsert control rods when the scram occurred.

3. Hatch Unit 1 - On January 31, 1979, the reactor experienced a scram during an approach to critical. Control rod 42-15 (fifth rod in Group 3) was being continuously withdrawn in "notch override" when the scram occurred, with a period of less then 5 seconds. The temperature was about 200 degrees F with effectively zero xenon.

As indicated above, these short period trips occurred under a wide variety of circumstances. They did have several things in common, however. In none of these cases was an accurate estimate of the critical position made prior to the approach to critical. In each case a rod was being pulled in a high worth region. Finally, in each case the operator, believing that the reactor was very subcritical, was pulling a rod on continuous withdrawal.

Action to be Taken by Licensees:

For all GE BWR power reactor facilities with an operating license:

1. Review and revise, as necessary, your operating procedures to ensure that an estimate of the critical rod pattern be made prior to each approach to critical. The method of estimating critical rod patterns should take into account all important reactivity variables (e.g., core xenon, moderator temperature, etc.).

2. Where inaccuracies in critical rod pattern estimates are anticipated due to unusual conditions, such as high xenon, procedures should require that notch-step withdrawal be used well before the estimated critical position is reached and all SRM channel indicators are monitored so as to permit selection of the most significant data.

3. Review and evaluate your control rod withdrawal sequences to assure that they minimize the notch worth of individual control rods, especially those withdrawn immediately at the point of criticality. Your review should ensure that the following related criteria are also satisfied:

   a. Special rod sequences should be considered for peak xenon conditions.

   b. Provide cautions to the operators on situations which can result in high notch worth (e.g. first rod in a new group will usually exhibit high rod worth).

4. Review and evaluate the operability of your "emergency rod in" switch to perform its function under prolonged severe use.

5. Provide a description of how your reactor operator training program covers the considerations above (i.e., items 1 thru 3).

6. Within 60 calendar days of the date of issue of this Bulletin, report in writing to the Director of the appropriate NRC Regional Office, describing your action(s) taken, or to be taken, in response to each of the above items. A copy of your report should be sent to the United States Nuclear Regulatory Commission, Office of Inspection and Enforcement, Division of Reactor Operations Inspection, Washington, D.C. 20555.

For all BWR facilities with a construction permit and all other power reactor facilities with an operating license or construction permit, this Bulletin is for information only and no written response is required.

Approved by GAO B180225 (R0072); clearance expires 7/31/80. Approval was given under a blanket clearance specifically for identified generic problems.

Enclosures:
1. IE Circular No. 77-07
2. List of IE Bulletins
   Issued in Last
   Twelve Months

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SHORT PERIOD DURING REACTOR STARTUP

DESCRIPTION OF CIRCUMSTANCES:

Recent events of concern to the NRC occurred at the Monticello and Dresden BWRs involving inadvertent high reactivity insertions causing short periods during reactor startup.

At Dresden Unit No. 2 on December 28, 1976 during a reactor startup following a scram from unrelated causes about 9 hours earlier, a rod withdrawal of one notch resulted in a rapid power rise associated with a reactor period of about one second and caused an Intermediate Range Monitor (IRM) Hi-Hi flux scram. The IRM was on its most sensitive scale. The moderator was essentially without voids and the reactor water temperature was 338°F. A similar event occurred at this facility on August 17, 1972.

At Monticello on February 23, 1977, following a reactor scram about 10 hours earlier from unrelated causes, a reactor period of about one second was experienced during startup before the reactor tripped on IRM Hi-Hi flux. The IRM was on its most sensitive scale and the short period resulted from the withdrawal of a control rod one notch. The reactor moderator had few voids and the water temperature was 480°F.

The two most recent events were similar in the following respects:

1. Prior to the earlier, unrelated scram, both plants had been operating at or near full power with axial flux peaking in the bottom portion of the core.

2. The time from the earlier scrams to the subsequent startups maximized the xenon concentrations in the core.

3. High worth rod locations were similar and both plants were using the same generic control rod pattern (identified as B1).

4. Prior to the IRM scram at both facilities, dramatic indications of high notch worth had been seen with rod withdrawals resulting in periods ranging from 10 to 30 seconds, which were terminated by reinsertion of the rod.

Review of the events showed that all of the systems including the Reactor Protections System functioned as required. Analyses indicate that the combination of essentially no voids in the moderator and high xenon concentration accounted for the conditions that resulted in the control rod notch acquiring an unusually high differential reactivity worth which approximated one-half percent delta K/K at Monticello. This excessive worth of rod notch was the result of essentially no voids in the moderator and peak xenon conditions which necessitated the withdrawal of significantly more control rods than is normally required to reach criticality. The resultant flux distribution at criticality magnified the normal axial peaking at the top of the core due to the heavy xenon concentrations at the bottom. Additionally, the radial contribution to flux peaking was enhanced due to the withdrawal of peripheral rods.

A review of NRC records showed that after the earlier event at Dresden Unit No. 2 on August 17, 1972, corrective measures were taken for the subsequent startup consisting of notchwise withdrawal of the group of rods. This corrective action was taken only for that operating cycle.

Evaluation of these events indicates that essentially trouble-free startups can be accomplished by avoiding the peak xenon with no moderator voids condition or possibly by the use of a rod pattern developed for these particular conditions.

These events indicate a need for all licensees of operating BWRs to review their startup procedures and practices to assure that their operating staff has adequate information to perform reactor startups avoiding such short periods in the event that the above-described conditions of peak xenon with no moderator voids exist at the time of startup. Operators should be made aware that extremely high rod notch worths can be encountered under these conditions. The procedures should include requirements for a thorough assessment following the occurrence of a short period before any further rod withdrawals are made. These considerations should be included in the operator training and requalification training programs.

No written response to this Circular is required. If you need additional information regarding this matter contact the Director of the cognizant NRC Regional Office.

LISTING OF IE BULLETINS ISSUED IN LAST TWELVE MONTHS

Bulletin No.	Subject	Date Issued	Issued To
78-06	Defective Cutler-Hammer, Type M Relays	5/31/78	All Power Reactor Facilities with an Operating License (OL) or Construction Permit (CP)
78-07	Protection afforded by Air-Line Respirators and Supplied-Air Hoods	6/12/78	All Power Reactor Facilities with an OL, all class E and F Research Reactors with an OL, all Fuel Cycle Facilities with an OL, and all Priority I Material Licensees
78-08	Radiation Levels from Fuel Element Transfer Tubes	6/12/78	All Power, Test and Research Reactor Facilities with an OL having Fuel Element Transfer Tubes
78-09	BWR Drywell Leakage Paths Associated with Inadequate Drywell Closures	6/14/78	All BWR Power Reactor Facilities with an OL (for action) or CP (for information)
78-10	Bergen-Paterson Hydraulic Shock Suppressor Accumulator Spring Coils	6/27/78	All BWR Power Reactor Facilities with an OL or CP
78-11	Examination of Mark I Containment Torus Welds	7/24/78	BWR Power Reactor Facilities with an OL for action: Peach Bottom 2 and 3, Quad Cities 1 and 2, Hatch 1, Monticello and Vermont Yankee. All other BWR Power Reactor Facilities with an OL for information

LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS (CONTINUED)

Bulletin No.	Subject	Date Issued	Issued To
78-12	Atypical Weld Material in Reactor Pressure Vessel Welds	9/29/78	All Power Reactor Facilities with an OL or CP
78-12A	Atypical Weld Material in Reactor Pressure Vessel Welds	11/24/78	All Power Reactor Facilities with an OL or CP
78-12B	Atypical Weld Material in Reactor Pressure Vessel Welds	3/19/79	All Power Reactor Facilities with an OL or CP
78-13	Failures In Source Heads of Kay-Ray, Inc., Gauges Models 7050, 7050B, 7051, 7051B, 7060, 7060B, 7061 and 7061B	10/27/78	All General and Specific Licensees with the subject Kay-Ray, Inc. Gauges
78-14	Deterioration of Buna-N Components In ASCO Solenoids	12/19/78	All GE BWR Facilities with an OL (for action), and all other Power Reactor Facilities with an OL or CP (for information)
79-01	Environmental Qualification of Class IE Equipment	2/8/79	All Power Reactor Facilities with an OL, except the 11 Systematic Evaluation Program Plants (for action), and all other Power Reactor Facilities with an OL or CP (For Information)
79-02	Pipe Support Base Plate Design Using Concrete Expansion Anchor Bolts	3/8/79	All Power Reactor Facilities with an OL or CP

LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS (CONTINUED)

Bulletin No.	Subject	Date Issued	Issued to
79-03	Longitudinal Weld Defects in ASME SA-312 Type 304 Stainless Steel Pipe Spools Manufactured by Youngstown Welding and Engineering Company	3/12/79	All Power Reactor Facilities with an OL or CP
79-04	Incorrect Weights for Swing Check Valves Manufactured by Velan Engineering Corporation	3/30/79	All Power Reactor Facilities with an OL or CP
79-05	Nuclear Incident at Three Mile Island	4/1/79	All Babcock and Wilcox Power Reactor Facilities with an OL, Except Three Mile Island 1 and 2 (For Action), and All Other Power Reactor Facilities With an OL or CP (For Information)
79-05A	Nuclear Incident at Three Mile Island - Supplement	4/5/79	Same as 79-05
79-06	Review of Operational Errors and System Misalignments Identified During the Three Mile Incident	4/11/79	All Pressurized Water Power Reactor Facilities with an OL Except B&W Facilities (For Action), All Other Power Reactor Facilities with an OL or CP (For Information)

LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS (CONTINUED)

Bulletin No.	Subject	Date Issued	Issued to
79-06A	Same Title as 79-06	4/14/79	All Westinghouse Designed Pressurized Power Reactor Facilities with an OL (For Action), and All Other Power Reactor Facilities with an OL or CP (For Information)
79-06A (Revision 1)	Same Title as 79-06	4/18/79	All Westinghouse Designed Pressurized Power Reactor Facilities with an OL (For Action), and All Other Power Reactor Facilities with an OL or CP (For Information)
79-06B	Same Title as 79-06	4/14/79	All Combustion Engineering Designed Pressurized Power Reactor Facilities with an OL (For Action), and All Other Power Reactor Facilities with an OL or CP (For Information)
79-07	Seismic Stress Analysis of Safety-Related Piping	4/14/79	All Power Reactor Facilities with an OL or CP

LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS (CONTINUED)

Bulletin No.	Subject	Date Issued	Issued to
79-08	Events Relevant to Boiling Water Power Reactors Identified During Three Mile Island Incident	4/14/79	All BWR Power Reactor Facilities with an OL (For Action), All Other Power Reactor Facilities with an OL or CP (For Information)
79-09	Failures of GE Type AK-2 Type Circuit Breaker in Safety Related Systems	4/17/79	All Power Reactor Facilities with an OL or CP
79-10	Requalification Training Program Statistics	5/11/79	All Power Reactor Facilities with an OL
79-11	Faulty Overcurrent Trip Device in Circuit Breakers for Engineered Safety Systems	5/22/79	All Power Reactor Facilities with an OL or CP