Attachment 1

EVALUATION OF LICENSEE'S RESPONSES TO IE BULLETIN 79-06B

NORTHEAST NUCLEAR ENERGY COMPANY, ETAL.

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-336

Introduction

By letter dated April 14, 1979, we transmitted I&E Bulletin No. 79-06B to Northeast Nuclear Energy Company (NNECO or the licensee). This bulletin specified actions to be taken by the licensee to avoid occurrence of an event similar to that which occurred at Three Mile Island, Unit No. 2 (TMI-2) on March 28, 1979. By letter dated April 24, 1979, NNECO provided their responses in conformance with the requirements of the Bulletin for the Millstone Nuclear Power Station, Unit No. 2 (Millstone-2). NNECO supplemented this response, by letters dated May 24 and 31, 1979, providing clarification and elaboration of certain of the items in response to our expressed concerns.

Our evaluation of these responses is given below.

Evaluation

In this evaluation, the paragraph numbers correspond to the bulletin action items and to the licensee's response to each action item.

1. NNECO initially reviewed the serious consequences of the TMI-2 accident with the majority of their operational personnel in

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specialized training sessions presented by the Operations Supervisor. In addition, similar presentations were made to the operators and plant management by an NRC staff team consisting of I&E and Operator Licensing Branch (OLB) representatives on April 20 and 21, 1979. NNECO provided the same training (excluding the NRC portion) to any operational personnel who missed the initial lectures prior to the Millstone-2 plant startup from the refueling outage. We find that the licensee has been responsive to the training requested by the reference bulletin.

2. NNECO states that operating procedures have been revised to require operator verifications of conditions which could lead to voiding. Subsequent communications have confirmed that the procedure revisions are complete, including review by the Plant Operations Review Committee, and that specific values of key parameters, to be monitored by the operators to assure that the Reactor Cooling System (RCS) remains subcooled, are provided.

   a. NNECO states that the parameters to be checked to determine the status of possible core voiding, in accordance with the revised operating procedures, are pressurizer pressure and hot leg temperature to determine the amount of RCS subcooling and core delta-temperature, steam generator delta-pressure and Reactor Coolant Pump (RCP) motor current and vibration to determine the status of RCS flow. A number of control room alarms are available to warn the operating staff of off-normal conditions that could lead to core voiding. NNECO has revised emergency procedures for natural circulation operations to direct the operator to monitor the degree of subcooling using the hot leg or in-core thermocouples versus the saturation temperature for the existing pressurizer pressure. Guidance is provided related to the use of steam dump/atmospheric dump operation in conjunction with auxiliary feedwater flow to establish a core flow producing at least a 10°F temperature gradient across the core. Direction is also provided to monitor the potential for voiding by verifying a stable or decreasing core delta-temperature of less than 50°F. The thermocouples in the in-core neutron detector strings may be used for monitoring the core in both forced and natural circulation modes. We find the licensee's response in regards to the recognition of possible void formation during forced or natural cooling mode of operation acceptable.

b. To assist the operators in taking appropriate actions to prevent void formation, NNECO states that routine and non-routine operations and the resultant procedures have been reviewed. For some plant operations, procedure changes were necessary and have been implemented. The revised procedures caution against over-feeding a steam generator during water recovery so as to prevent loss of pressurizer pressure and level control. According to subsequent conversations with NNECO, the procedures to be used in the event of a Loss of Coolant Accident (LOCA), Main Steam Line Rupture or Steam Generator Tube Rupture were revised to contain RCS pressure versus temperature curves indicating saturation, and $50^{\circ}F$ subcooled conditions. We find that the licensee has adequately addressed the operator actions required to prevent void formation.

c. The licensee states that the appropriate operator action required to enhance core cooling in the event core voiding occurs is to restore pressurizer pressure and level and reinstate RCS cooling using the steam generators. Level is re-established using the normal chemical and volume control system (CVCS) charging pumps or the ECCS high pressure safety injection (HPSI) system pumps, depending on RCS integrity. Core cooling, provided by RCS flow through the steam generators, will be maintained by the operation of at least one RCP per loop according to the revised emergency procedure (see Section 6-c). NNECO states that the recovery of RCS pressure and continued core cooling will assure void collapse. We find that the licensee has adequately addressed this concern of the bulletin.

3. In the design of Millstone-2, the automatic initiation of safety injection (SI) also results in initiation of the containment isolation actuation signal (CIAS). The Millstone-2 Technical Specifications (TS) setpoint values for these actuations are RCS pressure decreasing to 1600 psia or containment pressure above 5 psig. The same setpoints are used for both SI and CIAS NNECO states that all containment penetrations which are not required for engineered safety features operation or core cooling, and which are not isolated by locked closed containment isolation valves, are isolated by a CIAS. TS 3/4.6.3 gives the operability and surveillance requirements for the automatic containment isolation valves. We find that the existing containment isolation system meets the intention of the bulletin requirements.

4. NNECO does not believe that it is necessary or desirable to station an individual (with no other assigned concurrent duties and in direct and continuous communication with the control room) to promptly initiate adequate auxiliary feedwater to the steam generators during accidents at Millstone-2. They state that

because of: (1) immediate actions required by the reactor trip procedure to verify feedwater flow status; (2) complete control of the auxiliary feedwater system from the control room panel where main feedwater flow is controlled; (3) possible interference in the movement of control operators by an unlicensed individual; (4) fifteen minutes available before auxiliary feedwater is required; and (5) past experience with recovery from feedwater system problems, the requirement of this bulletin item is not justified. Although the staff agrees with many of the points raised by NNECO there is still a concern with successful auxiliary feedwater initiation for those plants which do not have automatic start. We believe that it is prudent to have an operator available in the control room able to devote his immediate attention to the feedwater control, with no other concurrent responsibilities, during transients requiring such action. NNECO has documented, in the letter dated 5/31/79, that a licensed operator who has direct responsibility for control and operation of all main and auxiliary feedwater systems will be in the main control room at all times. They, also, provide a backup in case the licensed operator is not available. NNECO further committed to document that the operator assigned to this function will at the time of a transient requiring such action take immediate control of the main and auxiliary feedwater systems, with no other concurrent responsibilities, until the steam generator levels return to a stable condition. We find this response to the bulletin request acceptable.

5. This bulletin item relates to the operation of the power operated relief valves (PORVS) on the pressurizer.

   a. NNECO response states that indications that plant operators may utilize to determine that a PORV is open are available in the control room. They consist of a temperature indicator on the PORV common discharge header and quench tank level, temperature and pressure indication. We find such instrumentation satisfies the concern expressed in the bulletin and appropriate direction is provided by the emergency procedures.

   b. NNECO states that "the emergency procedure for reactor trip has been revised to direct the operator to maintain closed the isolation valve of a stuck open PORV". In response to our questions, NNECO explained that sequential closing and possible reopening of the PORV individual block valves may be necessary to identify the leaking PORV. However, when the leaking PORV is identified, its block valve would not be reopened. The licensee's responses indicate that appropriate procedural control of a possible leaking PORV have been implemented.

6. This bulletin item makes specific requests of licensees to ensure that procedures and training instructions prevent the overriding of engineered safety features during accident conditions.

a. As a result of a reportable occurrence and in response to our November 29, 1978 letter regarding containment purging during plant operation, NNECO's indicated that appropriate procedures were recently revised to include cautions against using equipment overrides. They state, "The cautions only allow override if directed by approved procedures, for equipment or personnel protection, or when equipment is not needed for the operating mode". The licensee has performed another review, in light of the TMI-2 Accident, and found these procedures adequate.

The licensee places special emphasis on securing the containment spray pumps, when not needed, to prevent damage to equipment such as the RCPs. In subsequent communications with NNECO, we learned that the procedure allows these pumps to be secured by overriding an automatic action only if the containment pressure is below 10 psig. In the Millstone-2 design, containment air recirculation units, redundant to the spray pumps, are available during accident conditions to handle containment cooling requirements.

The licensee's response and the above example indicate that procedural controls, preventing the overriding of automatic actions of engineered safety features have been initiated in accordance with the bulletin.

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b. NNECO states that applicable emergency procedures have been revised to provide the specific instruction provided by the bulletin in regards to the continuation of HPSI pump operation after automatic actuation. Although this adequately addresses the requirement of the bulletin, we are providing the following clarification of the intent of paragraph 6.6.(2). "After 50°F of subcooling has been achieved, termination of HPI operation prior to 20 minutes is only permissible if it has been determined that continued operation would result in an unsafe plant condition, e.g., pressure/temperature considerations for the vessel integrity". In addition, NNECO provided instructions regarding charging pumps operation. They state that applicable procedures have been revised and contain the same requirements as proposed by Bulletin 79-06B. We find that the licensee has adequately addressed this item for HPSI and charging pump operation.

c. NNECO's responses say that applicable emergency procedures have been revised to require continued operation of at least one RCP per loop during the HPSI phase following an accident. They agree to leave the RCPs running or will restart the pumps as long as the pump is providing forced flow as indicated by control room indications. We find these statements responsive to the requirements of the bulletin. The following information is provided for clarification of the intent of bulletin paragraph 6.c. "In the event of HPSI initiation with RCP operating, at least one RCP shall remain operating in each loop as long as the pump(s) is providing forced flow and continued operation shall not result in an unsafe plant condition, e.g., loss of seal integrity may result in system failure of greater consequence than the benefit derived from forced flow."

d. The NNECO response states that the applicable emergency procedures have been revised to further minimize operator dependence on pressurizer level. We find that the licensee has adequately addressed this item as presented in the bulletin.

7. The licensee states that all safety related valve positions, positioning requirements and procedural controls, which ensure that the valves remain properly positioned, have been reviewed and are adequate to ensure proper operation of engineered safety features. The administrative procedures for control of maintenance on safety related equipment were revised to specifically assure correct positioning of valves which were

worked on or were used for isolation purposes. The positions of all safety related valves, except for locked valves are visually checked monthly. The positions of locked valves are visually checked prior to each startup and after any system manipulation that require their repositioning. We find the NNECO statements to be an adequate response to this item of the bulletin.

8. NNECO identifies all systems designed to transfer potentially radioactive gases and liquids out of the primary containment and states that all of these systems, which are not part of the engineered safety features, are automatically isolated by a CIAS. In addition, the containment purge valves, which are open only in the refueling and cold shutdown modes of operation, are closed upon detection of high radiation in the containment.

The licensee states that to eliminate the only potential for undesirable pumping, venting or other release, a plant design change has been completed to eliminate the AUTO start feature of the containment sump pump. In the event of a steam generator tube leak, the steam generator blowdown system will process radioactive water from the steam generators to the environment or aerated liquid radwaste. A CIAS or high radiation signal from the blowdown or the steam jet air ejectors will isolate blowdown, preventing an undesired release.

Following a postulated LOCA during the recirculation phase, potentially radioactive water will circulate from the containment sump through the LPSI pumps in the auxiliary building and then back to the reactor. The only potential leakage would be from pump seals, valve packings and other small sources. NNECO states that this operation would not result in any significant release.

NNECO also addressed the subject of administrative controls regarding the use of the manual overrides for the Millstone-2 systems. The subject of manual overrides is part of an ongoing staff review based on responses to our generic letter of November 29, 1978.

We find that the licensee has adequately addressed the bulletin concerns regarding possible release of radioactive gases or liquids from the containment.

9. Bulletin Item 9 relates to the safety-related system maintenance and test procedures.

   a. NNECO states that the administrative procedures have been revised to specify that prior to removal of safety related systems from service the redundant system will be verified operable. We find this concern of the bulletin has been properly addressed.

b. The licensee says that the procedures for maintenance and testing of safety related systems have been reviewed and changes were made to strengthen the requirement to verify operability of safety related systems prior to taking credit for the system(s) to satisfy TS requirements. We find this to be an adequate response to the request.

c. NNECO response states that a licensed operator is required to authorize all maintenance, tests, or surveillance which affect plant systems. Prior to releasing the controlling document, the operator ensures he is aware of the effect of the activity on the system or equipment. Upon completion of the item, the document is returned to the operator for acceptance or for the purpose of returning the system to service.

The NNECO response of May 24, 1979 states that the requirements for authorizing equipment maintenance, tests, or surveillance are entrusted to individuals qualified for Shift Supervisor or Supervising Control Operator positions, and that it is a corporate objective to have such personnel qualified at the NRC Senior Reactor Operator level. The status of all safety-related equipment and Technical Specification requirements are maintained in the Shift Supervisor's log. Each oncoming shift reviews the log to keep cognizant of the status of safety-related equipment. We find this to be an adequate response on the operating personnel notification requirements of the bulletin.

10. NNECO responds that a revision to the administrative procedure on communications and outside assistance has been approved. This revision incorporates the required notifications and establishment of communication channels requested in the bulletin.

The NNECO response requests more specific guidance on "Immediate notification" circumstances and notes t. t the bulletin statement is a general statement subject to interpretation. We agree that the bulletin statement is, of necessity, a general statement and was prepared in light of our knowledge of the early sequence of events at TMI-2 prior to NRC notification. We leave it to the licensee to likewise review the TMI-2 events and, using that as guidance together with his experience in routine operations and the recognition of non-routine events, promulgate his own interpretation of prompt NRC notification, keeping in mind NRC's role in these matters. However, we conclude that should a question arise in regard to NRC notification, the licensee should plan to err on the side of providing prompt notification.

11. NNECO has reviewed the operating modes and procedures used to deal with significant amounts of hydrogen gas that could be generated and collect in the RCS or released to the containment. They describe these methods that they use for degassing the primary coolant system (the radwaste degasifier, pressurizer steam space vent, and volume control tank gas space purge). They also described two methods for hydrogen removal from containment (hydrogen recombiner and containment purge).

    Their response indicates an understanding of this concern expressed by our bulletin. We find this response acceptable.

CONCLUSION

Based on our review of the information provided by the licensee to date, we conclude that the licensee has correctly interpreted IE Bulletin No. 79-06B. The actions taken demonstrate his understanding of the concerns arising from the Three Mile Island incident in reviewing their implications on his own operations, and provide added assurance for the protection of the public health and safety during plant operation.

Dated: June 7, 1979

Attachment 2

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

February 8, 1979

IE Bulletin No. 79-01

ENVIRONMENTAL QUALIFICATION OF CLASS IE EQUIPMENT

Description of Circumstances:

The intent of IE Circular 78-08 was to highlight to all licensees important lessons learned from environmental qualification deficiencies reported by individual licensees. In this regard, licensees were requested to examine installed safety-related electrical equipment and determine that proper documentation existed which provided assurance that this equipment would function under postulated accident conditions. The scope of IE Circular 78-08 was much broader than other previously issued Bulletins and Circulars (such as IEB 78-04 and IEB 78-02) which addressed specific component failures. The intent of this Bulletin is to raise the threshold of IE Circular 78-08 to the level of a Bulletin; i.e., action requiring a licensee response.

Inspections conducted to date by the NRC of licensees' activities in response to IE Circular 78-08 have identified one component which licensees have found to be unqualified for service within the LOCA environment. Specificially, unqualified stem mounted limit switches (SMLS), other than those identified in previously issued IE Bulletin 78-04, were found to be installed on safety-related valves inside containment at both Duane Arnold and Quad Cities 1 and 2 Nuclear Generating Stations. The unqualified switches are identified as NAMCO Models SL2-C-11, S3CML, SA1-31, SA1-32, D1200j, EA-700 and EA-770 switches. According to the manufacturer, these switches are designed only for general purpose applications and are not considered suitable devices for service in the LOCA environment. Consequently, switches are being replaced at the above power plants with qualified components.

Also, NRC inspection of component qualification has identified equipment which does not have documentation indicating it is qualified for the LOCA environment. The inspections have also identified that the licensees' re-review and resolution of problem areas are not receiving the level of attention from all licensees which the NRC believes is warranted. Because of the protracted schedule for completion of the re-review, we are now requesting the power reactor facilities with operating licenses to expedite completion of their re-review program originally requested by IE Circular 78-08 dated May 31, 1978.

Action to Be Taken By Licensees of All Power Reactor Facilities (Except Those 11 SEP Plants Listed on Enclosure 3) With An Operating License:

1. Complete the re-review program described in IE Circular 78-08 within 120 days of receipt of this Bulletin.

2. Determine if the types of stem mounted limit switches described above are being used or planned for use on safety-related valves which are located inside containment at your facility. If so, provide a written report to the NRC within the time frame specified and to the address specified in Item 4 below.

3. Provide written evidence of the qualification of electrical equipment required to function under accident conditions.* For those items not having complete qualification data available for review, identify your plans for determining qualification, either by testing or engineering analysis, or combination of these, or by replacement with qualified equipment. Include your schedule for completing these actions and your justification for continued operation.

   Submit this information to the Director, Division of Reactor Operations Inspection, Office of Inspection and Enforcement, Nuclear Regulatory Commission, Washington, D.C. 20555 with a copy to the appropriate NRC Regional Office within 120 days of receipt of this Bulletin.

4. Report any items which are identified as not meeting qualification requirements for service intended to the Director, Division of Operating Reactors, Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission, Washington, D.C. 20555 with a copy to the appropriate NRC Regional Office within 24 hours of identification. If plant operation is to continue following identification, provide justification for such operation. Provide a detailed written report within 14 days of identification to NRR, with a copy to the appropriate NRC Regional Office.

* This written evidence should include: 1) component description; 2) description of the accident environment; 3) the environment to which the component or equipment is qualified; 4) the manner of qualification which should include test methods such as sequential, synergistic, etc., and 5) identification of the specific supporting qualification documentation.

No additional written response to this IE Bulletin is required other than those responses described above. NRC inspectors will continue to monitor the licensees' progress in completing the requested action described above. If additional information is required, contact the Director of the appropriate NRC Regional Office.

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

attachment 3

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

March 8, 1979

IE Bulletin No. 79-02

PIPE SUPPORT BASE PLATE DESIGNS USING CONCRETE EXPANSION ANCHOR BOLTS

Description of Circumstances:

While performing inservice inspections during a March-April 1978 refueling outage at Millstone Unit 1, structural failures of piping supports for safety equipment were observed by the licensee. Subsequent licensee inspections of undamaged supports showed a large percentage of the concrete anchor bolts were not tightened properly.

Deficiency reports, in accordance with 10 CFR 50.55(e), filed by Long Island Lighting Company on Shoreham Unit 1, indicate that design of base plates using rigid plate assumptions has resulted in underestimation of loads on some anchor bolts. Initial investigation indicated that nearly fifty percent of the base plates could not be assumed to behave as rigid plates. In addition, licensee inspection of anchor bolt installations at Shoreham has shown over fifty percent of the bolt installations to be deficient.

Vendor Inspection Audits by NRC at Architect Engineering firms have shown a wide range of design practices and installation procedures which have been employed for the use of concrete expansion anchors. The current trends in the industry are toward more rigorous controls and verification of the installation of the bolts.

The data available on dynamic testing of the concrete expansion anchors show fatigue failures can occur at loads substantially below the bolt static capacities due to material imperfections or notch type stress risers. The data also show low cycle dynamic failures at loads below the bolt static capacities due to joint slippage.

Action to be Taken by Licensees and Permit Holders:

For pipe support base plates that use concrete expansion anchor bolts in Seismic Category I systems as defined by Regulatory Guide 1.29, "Seismic Design Classification" Revision 1, dated August 1973 or as defined in the applicable FSAR.

1. Verify that pipe support base plate flexibility was accounted for in the calculation of anchor bolt loads. In lieu of supporting analysis justifying the assumption of rigidity, the base plates

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

March 12, 1979

IE Bulletin No. 79-03

LONGITUDINAL WELD DEFECTS IN ASME SA-312 TYPE 304 STAINLESS STEEL PIPE SPOOLS MANUFACTURED BY YOUNGSTOWN WELDING AND ENGINEERING COMPANY

Description of Circumstances:

On September 27, 1978, the Arizona Public Service Company reported that defects had been discovered in longitudinal welds in ASME Section III class 2 pipe supplied for the Palo Verde Nuclear Generating Station (PVNGS). On November 17, 1978, the Southern California Edison Company reported similar defects in pipe supplied for the San Onofre Nuclear Generating Station, Units 2 and 3.

Pullman Power Products of Los Angeles, California supplies safety-related fabricated piping spools of various diameters for the PVNGS. The defects were discovered by Pullman in ASME SA-312 type 304 stainless steel pipe supplied to Pullman by Youngstown Welding and Engineering Company of Youngstown, Ohio. The pipe is manufactured by rolling plate into cylinders and then fusion welding the longitudinal seam without filler metal.

Pullman discovered defects in the longitudinal welds while radiographing their circumferential shop welds. Further radiographic examination of the longitudinal welds revealed rejectable porosity and lack of fusion.

Pullman then performed ultrasonic examination of the full length of the longitudinal welds and discovered indications exceeding the acceptance criteria of ASME Section III. Further ultrasonic examination revealed indications in other piping subassemblies where pipe was supplied by Youngstown. Two indications ver[illegible]ied by radiography were identified as porosity and measured 0.350 inch [illegible] 0.125 inch in one case and 0.300 inch by 0.125 inch in another cas[illegible] in pipe with a nominal wall thickness of 0.375 inch.

The additional examinations revealed that of 103 spools and four pipe supports shipped to PVNGS, 44 spools and one pipe support were found to contain ultrasonic indications exceeding those permitted by the ASME Code. Of 65 partially fabricated piping spools, 30 were found to be similarly defective. The acceptance criteria for the pipe supplied by Youngstown includes 100 percent ultrasonic examination of the longitudinal welds in accordance with ASME Section III. The documentation provided with the pipe indicated that the required ultrasonic examination had been performed by Youngstown but the rejectable indications were not identified.

A special inspection was performed at Youngstown by NRC inspectors during the week of January 22, 1979. It was determined that the apparent cause of the identified defects was inadequate control of welding parameters although no specific ASME Code violations could be identified. Youngstown has recently hired a consultant to reevaluate the fusion welding parameters and revised their welding procedures to provide better control of welding current, voltage and travel speed for all material thickness ranges.

Ultrasonic examinations of the pipe welds were performed by a subcontractor to Youngstown. The reason why this subcontractor's ultrasonic testing did not detect indications exceeding ASME Code acceptance criteria was not determined. The piping was known to have been tested in the heat treated condition, prior to the removal of surface oxides. However, a comparison of attenuation of the pipe in as heat treated vs. heat treated and pickled condition did not reveal a discernible difference.

The NRC inspectors could not determine a definite time period during which the welding and ultrasonic testing problems are thought to have existed. All type 304 or 316 SA 312 pipe manufactured before mid-November, 1978 may have been shipped in similar condition. As a large supplier, Youngstown is known to have supplied piping for nuclear applications to the Dravo Corporation, Chicago Bridge and Iron, Flowline Corporation and ITT Grinnell Industrial Piping Inc. In addition, piping was also supplied to material warehousing operations including Albert Pipe Supply, Guyon Alloys Inc., and Allegheny Ludlum Steel Corporation which may have eventually been used in safety-related nuclear applications.

Action to be Taken by the Licensees and Permit Holders:

For all power reactor facilities with an operating license or a construction permit:

1. Determine whether ASME SA-312, type 304 or other welded (without filler metal) pipe manufactured by Youngstown Welding and Engineering Company is in use or planned for use in safety-related systems at your facility.

2. For those safety-related systems where the subject piping is in use or planned for use, identify the application of the piping including system, pipe location, pipe size and design pressure/temperature requirements.

3. Develop a program for volumetric examination of the longitudinal welds including acceptance criteria for the piping identified in Item 2 above. Describe planned corrective actions if acceptance criteria are not met. If a sampling program is utilized explain the basis for the sample size.

4. For facilities with an operating license, a report of the above actions, including the date(s) when they will be completed shall be submitted within 30 days of receipt of this Bulletin.

5. For facilities with a construction permit, a report of the above actions, including the date(s) when they will be completed shall be submitted within 60 days of receipt of this Bulletin.

Reports shoul be submitted to the Director of the appropriate NRC Regional Office and a copy should be forwarded to the NRC Office of Inspection and Enforcement, Division of Reactor Construction Inspection, Washington, D.C., 20555.

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

Attachment 5

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

March 30, 1979

IE Bulletin No. 79-04

INCORRECT WEIGHTS FOR SWING CHECK VALVES MANUFACTURED BY VELAN ENGINEERING CORPORATION

Description of Circumstances:

North Anna No. 1, Beaver Valley No. 1 and Salem No. 1 have reported to the NRC that they had been provided incorrect weights for the six inch swing check valves provided by Velan Engineering Corporation. The six inch valve weight provided on the drawing was 225 pounds, whereas the actual weight has been determined to be 450 pounds. In addition to the 6 inch valves, drawings for 3 inch valves have specified 60 pounds weight while the measured weight by the manufacturer was 85 pounds and drawings for 4 inch valves have specified 100 pounds weight while the measured weight was 135 pounds. The manufacturer presently estimates the following maximum weights for swing check valves.

Nominal Valve Size	Maximum Weight (lbs) for High Pressure (1500 psi)	
	Up to 1973	After 1973
3 inches	85	100
4 inches	135	150
6 inches	450	525
8 inches	750	1200
10 inches	1200	1200

The NRC staff has indications that in some cases, incorrect valve weights derived from engineering drawings were used in piping stress analyses. The staff is not aware of a significant difference in the actual weight and the weight provided on drawings for the 8 and 10 inch valves.

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

July 26, 1979

IE Bulletin Nos. 79-05C & 79-06C

NUCLEAR INCIDENT AT THREE MILE ISLAND - SUPPLEMENT

Description of Circumstances:

Information has become available to the NRC, subsequent to the issuance of IE Bulletins 79-05, 79-05A, 79-05B, 79-06, 79-06A, 79-06A (Revison 1) and 79-06B, which requires modification to the "Action To Be Taken By Licensees" portion of IE Bulletins 79-05A, 79-06A and 79-06B, for all pressurized water reactors (PWRs).

Item 4.c of Bulletin 79-05A required all holders of operating licenses for Babcock & Wilcox designed PWRs to revise their operating procedures to specify that, in the event of high pressure injection (HPI) initiation with reactor coolant pumps (RCPs) operating, at least one RCP per loop would remain operating. Similar requirements, applicable to reactors designed by other PWR vendors, were contained in Item 7.c of Bulletin 79-06A (for Westinghouse designed plants) and in Item 6.c of Bulletin 79-06B (for Combustion Engineering designed plants).

Prior to the incident at Three Mile Island Unit 2 (TMI 2), Westinghouse and its licensees generally adopted the position that the operator should promptly trip all operating RCPs in the loss of coolant accident (LOCA) situation. This Westinghouse position, has led to a series of meetings between the NRC staff and Westinghouse, as well as with other PWR vendors, to discuss this issue. In addition, more detailed analyses concerning this matter were requested by the NRC. Recent preliminary calculations performed by Babcock & Wilcox, Westinghouse and Combustion Engineering indicate that, for a certain spectrum of small breaks in the reactor coolant system, continued operation of the RCPs can increase the mass lost through the break and prolong or aggravate the uncovering of the reactor core.

The damage to the reactor core at TMI 2 followed tripping of the last operating RCP, when two phase fluid was being pumped through the reactor coolant system. It is our current understanding that all three of the nuclear steam system suppliers for PWRs now agree that an acceptable action under LOCA symptoms is to trip all operating RCPs immediately, before significant voiding in the reactor coolant system occurs.

Action To Be Taken By Licensees:

In order to alleviate the concern over delayed tripping of the RCPs after a LOCA, all holders of operating licens[illegible] following actions:

DUPLICATE DOCUMENT 9C6035

Entire document previously entered into system under:

ANO 7908020171

No. of pages: 6

attachment 7

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

April 14, 1979

IE Bulletin No. 79-07

SEISMIC STRESS ANALYSIS OF SAFETY-RELATED PIPING

Description of Circumstances:

In the course of evaluation of certain piping designs, significant discrepancies were observed between the original piping analysis computer code used to analyze earthquake loads and a currently acceptable computer code developed for this purpose. This problem resulted in the Nuclear Regulatory Commission order to shutdown five power reactors whose design had involved the use of the suspect computer codes (IE Information Notice No. 79-06). The difference in predicted piping stresses between the two computer codes is attributable to the fact that the piping analysis code used for a number of piping systems uses an algebraic summation of the loads predicted separately by the computer code for both the horizontal components and for the vertical component of seismic events. This is an incorrect treatment of such loads and was not recognized as such at the time the original analyses were performed. Such codirectional loads should not be algebraically added (with predicted loads in the negative direction offsetting predicted loads in the positive direction) unless certain more complex time-history analyses are performed. Rather, to properly account for the effects of earthquakes on systems important to safety, as required by "Design Bases for Protection Against Natural Phenomena," General Design Criterion 2 of Appendix A to 10 CFR Part 50, such loads should be combined absolutely or, as is the case in the newer codes, using techniques such as the square root of the sum of the squares. These combinations of loads conform to current industry practice.

The inappropriate analytical treatment of load combinations discussed above becomes significant for piping runs in which the horizontal seismic excitation can have both horizontal and vertical components of response on piping systems, and the vertical seismic excitation also has both horizontal and vertical components of response. It is in these runs that the predicted earthquake loads may differ significantly.

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Although the greatest differences in predicted loads would tend to be limited to localized stresses in pipe supports and restraints or in weld attachments to pipes, there could be a substantial number of areas of high stress in piping, as well as a number of areas in which there is potential for damage to adjacent restraints or supports. Any of these situations could have significant adverse effects on the ability of the piping system to withstand seismic events.

The NRC staff has not yet determined that all of the piping systems important to safety that were designed using a piping analysis computer code which contains the algebraic summation error, have been identified. Certain information is needed in order to make this determination.

Action To Be Taken By All Licensees and Permit Holders:

For all power reactor facilities with an operating license or a construction permit:

(1) Identify which, if any, of the methods specified below were employed or were used in computer codes for the seismic analysis of safety related piping in your plant and provide a list of safety systems (or portions thereof) affected:

Response Spectrum Model Analysis:

a. Algebraic (considering signs) summation of the codirectional spatial components (i.e., algebraic summation of the maximum values of the codirectional responses caused by each of the components of earthquake motion at a particular point in the mathematical model).

b. Algebraic (considering signs) summation of the codirectional inter model responses (i.e., for the number of modes considered, the maximum values of response for each mode summed algebraically)

Time History Analysis:

a. Algebraic summation of the codirectional maximum responses or the time dependent responses due to each of the components of earthquake motion acting simultaneously when the eartquake directional motions are not statistically independent.

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(2) Provide complete computer program listings for the dynamic response analysis portions of the codes which employed the techniques identified in Item 1 above.

(3) Verify that all piping computer programs were checked against either piping benchmark problems or compared to other piping computer programs. You are requested to identify the benchmark problems and/or the computer programs that were used for such verifications or describe in detail how it was determined that these programs yielded appropriate results (i.e., gave results which corresponded to the correct performance of their intended methodology).

(4) If any of the methods listed in item 1 are identified, submit a plan of action and an estimated schedule for the re-evaluation of the safety related piping, supports, and equipment affected by these analysis techniques. Also provide an estimate of the degree to which the capability of the plant to safely withstand a seismic event in the interim is impacted.

The responses for Items 1, 2 and 3 above, should include all subsequent piping system additions and modifications. Any re-evaluation required, in conformance with Item 4, should incorporate the "as built" conditions.

Licensees of all operating power reactor facilities should submit the information identified in Items 1 through 4, above, within 10 days of the date of this letter. Holders of construction permits for power reactor facilities should submit this information within 45 days of the date of this letter.

Reports should be submitted to the Director of the appropriate NRC Regional Office and a copy should be forwarded to the NRC Office of Inspection and Enforcement, Division of Reactor Operations Inspection, Washington, D.C., 20555.

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

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

April 17, 1979

IE Bulletin No. 79-09

FAILURES OF GE TYPE AK-2 CURCUIT BREAKER IN SAFETY RELATED SYSTEMS

Description of Circumstances:

Twelve failures of General Electric (GE) type AK-2 (i.e., AK-2A-15, 25, 50, 75, or 100) Circuit Breakers installed in safety-related systems have been reported since 1975. The failures occurred at the following facilities:

	Date	Facility	System
1.	9/16/78	Arkansas-1	Control Rod Drive System
2.	9/25/78	Arkansas-1	Control Rod Drive System
3.	10/17/78	Arkansas-1	Control Rod Drive System
4.	1/22/78	Crystal River-3	Control Rod Drive System
5.	8/7/75	Oconee Unit-3	Control Rod Drive System
6.	1/18/79	Oconee Unit-3	Control Rod Drive System
7.	1/22/79	Oconee Unit-1	Control Rod Drive System
8.	1/31/79	Oconee Unit-1	Control Rod Drive System
9.	4/25/75	TMI/1	Control Rod Drive System
10.	11/26/78	Oyster Creek-1	Containment Spray Pump
11.	11/30/78	Oyster Creek-1	Service Water Pump No. 1
12.	11/30/78	Oyster Creek-1	Service Water Pump No. 2

It is significant to note that during a loss-of-off-site power test on November 30, 1978, at Oyster Creek, both service water pump circuit breakers failed to trip, as required. The undervoltage relays which monitor voltage level on each emergency bus functioned properly but could not actuate the trip mechanism via the undervoltage trip device within each circuit breaker. These failures, in turn, created a potential overload condition on each emergency diesel generator unit by allowing simultaneous starting of multiple high horse power motors during sequential loading phase of the test.

The causes for failure were attributed to either binding within the linkage mechanism of the undervoltage (UV) trip device and trip shaft assembly or out-of-adjustment conditions in the same linkage mechanism. Babcock and Wilcox (B&W) and GE determined that the binding and out-of-adjustment resulted from inadequate preventive maintenance programs

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May 11, 1979

IE Bulletin No. 79-10

REQUALIFICATION TRAINING PROGRAM STATISTICS

Description of Circumstances:

Because of the apparent failure of the operators at Three Mile Island to recognize certain plant conditions and take appropriate action to effectively cool the core and contain fission products after release from the core, the NRC Commissioners are evaluating licensee's requalification training programs. As one element in this evaluation, the NRC is interested in obtaining statistics about the failure rate on the annual requalification examinations. The information requested below along with other information will then be used to evaluate the effectiveness of the operator requalification training program.

Action to be Taken by Licensees:

For all power reactor facilities with an operating license.

1. Provide both the total number and percentage of operators who have failed the annual requalification examination.

2. Provide the percentage of those operators who take the annual requalification examination and are required to attend lectures on categories of material for which they received a grade of less than 80 percent. Also provide the total number of supplemental lectures attended (e.g., 3 operators had to attend 2 lectures, 1 operator had to attend 3 lectures, etc.).

3. Provide both the total number and percentage of operators under the requalification program that participated in accelerated training because they either scored less than 70 percent overall on the annual written examination or had an unsatisfactory performance on the oral examination.

4. Provide the same information required by 1 thru 3 on Senior Operators.

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May 22, 1979

IE Bulletin No. 79-11

FAULTY OVERCURRENT TRIP DEVICE IN CIRCUIT BREAKERS FOR ENGINEERED SAFETY SYSTEMS

Discussion:

We have received information from Westinghouse and an NRC licensee relating to the potential failure of a circuit breaker in an engineered safety system of a nuclear power plant. This circuit breaker had a defect in one of its three time delay dashpots which resulted in a reduced time delay for overcurrent protection. The defect was a small hairline crack in the end cap of the dashpot. Further investigation by this licensee disclosed that 7 out of 17 spare dashpot end caps and 2 non-engineered safety feature breakers also had similar defects. The circuit breaker is a Westinghouse type DB-75. Westinghouse type DB-50 breakers also use the same type of dashpot and end cap. DB-50 and -75 breakers are used extensively in PWR's, and some BWR's may also have the same breakers.

Similar make and model circuit breakers, when used for scram purposes, do not require the overcurrent trip feature and thus are not of concern. The end cap crack defect, if severe enough, could result in premature tripping of the circuit breaker because of insufficient time delay in overcurrent protection; i.e., the motor starting (inrush) current could cause the breaker to trip inadvertently and thus prevent the motor start.

The defects reported by the licensee in April 1979, occurred in the replacement end caps which were provided to solve the problem described in IE Bulletin 73-1. The subject of Bulletin 73-1 was end caps made of a black phenolic material. As a result of that Bulletin, the black end caps were replaced with a new type made of fibre-filled polyester material cal[illegible] "navy-gray". Prior to the April 1979 report, there have been no reports of [illegible]uspect "navy-gray" end caps either from scheduled testing or unusual behavior in service. The manufacturer of the "navy-gray" end caps believes the crack defects may be linked to a raw material batch problem. That is, the molding ingredient materials used may have neared the end of their shelf life before use. It is not believed the end caps, after fabrication, have a significant shelf life limit, due to the low residual stress and low crack propagation probabilities.

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WASHINGTON, D.C. 20555

May 31, 1979

IE Bulletin No. 79-12

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.

ATTACHMENT 12

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

June 25, 1979

IE Bulletin No. 79-13

CRACKING IN FEEDWATER SYSTEM PIPING

Description of Circumstances:

On May 20, 1979, Indiana and Michigan Power Company notified the NRC of cracking in two feedwater lines at their D. C. Cook Unit 2 facility. The cracking was discovered following a shutdown on May 19 to investigate leakage inside containment. Leaking circumferential cracks were identified in the 16-inch feedwater elbows adjacent to two steam generator nozzle elbow welds. Subsequent radiographic examination revealed crack indications in all eight steam generator feedwater lines at this location on both Units 1 and 2.

On May 25, 1979, a letter was sent to all PWR licensees by the Office of Nuclear Reactor Regulation which informed licensees of the D. C. Cook failures and requested specific information on feedwater system design, fabrication, inspection and operating histories. To further explore the generic nature of the cracking problem, the Office of Inspection and Enforcement requested licensees of PWR plants in current outages to immediately conduct volumetric examination of certain feedwater piping welds.

As a result of these actions, several other licensees with Westinghouse steam generators reported crack indications. Southern California Edison reported on June 5, 1979, that radiographic examination revealed indications of cracking in feedwater nozzle-to-piping welds on two of three steam generators of San Onofre Unit 1. On June 15, 1979, Carolina Power and Light reported that radiography showed crack indications in similar locations at their H. B. Robinson Unit 2. Duquesne Power and Light confirmed on June 18, 1979, that radiography has shown cracking in their Beaver Valley Unit 1 feedwater piping to vessel nozzle weld. Public Service Electric and Gas Company reported on June 20, 1979 that Salem Unit 1 also has crack indications. Wisconsin Public Service company decided on June 20, 1979 to cut out a feedwater nozzle to pipe weld which contained questionable indication, for metallurgical examination. As of June 22, 1979 and since May 25, 1979 seven other PWR facilities have inspected the feedwater nozzle-to-pipe welds without finding cracking indications.

The feedwater nozzle-to-pipe configurations for D.C. Cook and for San Onofre are shown on the attached figures 1 and 2. A typical feedwater pipe-to-nozzle weld joint detail showing the principal crack locations for D.C. Cook and San Onofre are shown on the attached figure 3.

On March 17, 1977, during heat-up for
Unit 1, a leak was discovered in the
the 16-inch diameter feedwater piping
nondestructive examination of all noz

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July 2, 1979

IE Bulletin No. 79-14

SEISMIC ANALYSES FOR AS-BUILT SAFETY-RELATED PIPING SYSTEMS

Description of Circumstances:

Recently two issues were identified which can cause seismic analysis of safety-related piping systems to yield nonconservative results. One issue involved algebraic summation of loads in some seismic analyses. This was addressed in show cause orders for Beaver Valley, Fitzpatrick, Maine Yankee and Surry. It was also addressed in IE Bulletin 79-07 which was sent to all power reactor licensees.

The other issue involves the accuracy of the information input for seismic analyses. In this regard, several potentially unconservative factors were discovered and subsequently addressed in IE Bulletin 79-02 (pipe supports) and 79-04 (valve weights). During resolution of these concerns, inspection by IE and by licensees of the as-built configuration of several piping systems revealed a number of nonconformances to design documents which could potentially affect the validity of seismic analyses. Nonconformances are identified in Appendix A to this bulletin. Because apparently significant nonconformances to design documents have occurred in a number of plants, this issue is generic.

The staff has determined, where design specifications and drawings are used to obtain input information for seismic analysis of safety-related piping systems, that it is essential for these documents to reflect as-built configurations. Where subsequent use, damage or modifications affect the condition or configuration of safety-related piping systems as described in documents from which seismic analysis input information was obtained, the licensee must consider the need to re-evaluate the seismic analyses to consider the as-built configuration.

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July 11, 1979

IE Bulletin No. 79-15

DEEP DRAFT PUMP DEFICIENCIES

Description of Circumstances:

On October 20, 1978, Commonwealth Edison Company reported that manufacturing deficiencies had been identified in new high pressure core spray, low pressure core spray, and residual heat removal pumps manufactured by Ingersoll-Rand (I-R) Company, Cameron Pump Division.

Each of these pumps is a vertical turbine pump with impellers located in bowls in a sump or a self contained barrel. The motor (prime mover) is located at the highest pump elevation to take into account maximum flooding at the site or space considerations. The suction is at the lower end of the pump while the discharge head is just below the driver. Bearings supporting the vertical shaft segments (usually 5 to 10 segments) are either self lubricated, force fed (lubricated by fluid being pumped), or oil lubricated and maintained within their own isolated system. These pumps are designated as "Deep Draft". Figures 1&2 show typical outlines of such pumps.

The internal deficiencies, identified through dimensional and visual inspections were as follows:

Low Pressure Core Spray Pumps (I-R Model No. 29APKD-5) (Date of Manufacture - Feburary 1973)

. Loose impeller bolts and bolts improperly staked
. Loose key - keyway fit
. Excessive runout on pump shaft
. Bearing showed wear
. Bearing clearance exceeded recommended tolerance
. Coupling thread galled
. Wear ring clearance out-of-specification
. Impeller-to-shaft clearance out of specification
. Cracks found in second-and-third-stage impellers
. Stuffing box bushings were severely galled

High Pressure Core Spray Pumps (I-R Model No. 12X20KD) (Date of Manufacture - September 1972)

. Bearing clearance exceeded recommended tolerance
. Wear ring clearance out-of-specification
. Bearings showed wear

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July 26, 1979

IE Bulletin No. 79-16

VITAL AREA ACCESS CONTROLS

Description of Circumstances:

An attempt to damage new fuel assemblies occurred recently at an operating nuclear reactor facility. During a routine fuel inspection, the licensee discovered that a chemical liquid had been poured over 62 of 64 new fuel assemblies. Analysis indicates that the chemical liquid was sodium hydroxide, a chemical stored and used onsite.

The licensee stores new fuel assemblies in dry storage wells on the same elevation as the spent fuel pool within the Fuel Building, a vital area. Access to the building is controlled by use of a coded keycard which electronically unlocks the alarmed personnel portals. The licensee issues coded keycards to both licensee and contractor personnel after the successful completion of a background screening program. In addition, licensee site management certifies monthly that each individual has the need for a coded keycard in order to perform required duties. Further access within this building is not limited by other barriers or controls.

As a result of this incident, an initial licensee audit determined that several hundred licensee and contractor personnel had access to this area during the period when the attempt to damage the fuel was made. The audit also revealed that one coded keycard reader at a vital area portal was inaccurately recording access data at the alarm station. Also discovered during this audit were indications of frequent "tailgating" on access through the portals. Tailgating occurs when more than one person passes through a portal on one person's authorized access. Their passage is therefore not recorded, and unauthorized persons could gain entry in this manner. Tailgating does not include authorized access controlled by an escort.

Discussion of Applicable Requirements:

10 CFR 73.55(a) requires the licensees to protect against industrial sabotage committed by an insider in any position. 10 CFR 73.55(d)(7) states that access to Vital Areas shall be positively controlled and limited to individuals who are authorized access to vital equipment and who require such access to perform their duties. Specific commitments implementing this regulation are described in each licensee's approved Security Plan.

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July 26, 1979

IE Bulletin No. 79-17

PIPE CRACKS IN STAGNANT BORATED WATER SYSTEMS AT PWR PLANTS

Description of Circumstances:

During the period of November 1974 to February 1977 a number of cracking incidents have been experienced in safety-related stainless steel piping systems and portions of systems which contain oxygenated, stagnant or essentially stagnant borated water. Metallurgical investigations revealed these cracks occurred in the weld heat affected zone of 8-inch to 10-inch type 304 material (schedule 10 and 40), initiating on the piping I.D. surface and propagating in either an intergranular or transgranular mode typical of Stress Corrosion Cracking. Analysis indicated the probable corrodents to be chloride and oxygen contamination in the affected systems. Plants affected up to this time were Arkansas Nuclear Unit 1, R. E. Ginna, H.B.Robinson Unit 2, Crystal River Unit 3, San Onofre Unit 1, and Surry Units 1 and 2. The NRC issued Circular 76-06 (copy attached) in view of the apparent generic nature of the problem.

During the refueling outage of Three Mile Island Unit 1 which began in February of this year, visual inspections disclosed five (5) through-wall cracks at welds in the spent fuel cooling system piping and one (1) at a weld in the decay heat removal system. These cracks were found as a result of local boric acid build-up and later confirmed by liquid penetrant tests. This initial identification of cracking was reported to the NRC in a Licensee Event Report (LER) dated May 16, 1979. A preliminary metallurgical analysis was performed by the licensee on a section of cracked and leaking weld joint from the spent fuel cooling system. The conclusion of this analysis was that cracking was due to Intergranular Stress Corrosion Cracking (IGSCC) originating on the pipe I.D. The cracking was localized to the heat affected zone where the type 304 stainless steel is sensitized (precipitated carbides) during welding. In addition to the main through-wall crack, incipient cracks were observed at several locations in the weld heat affected zone including the weld root fusion area where a miniscule lack of fusion had occurred. The stresses responsible for cracking are believed to be primarily residual welding stresses in as much as the calculated applied stresses were found to be less than code design limits. There is no conclusive evidence at this time to identify those aggressive chemical species which promoted this IGSCC attack. Further analytical efforts in this area and on other system welds are being pursued.

NUREG-0510

ATTACHMENT 24

IDENTIFICATION OF UNRESOLVED SAFETY ISSUES RELATING TO NUCLEAR POWER PLANTS

Report to Congress

Office of Nuclear Reactor Regulation

U. S. Nuclear Regulatory Commission

attachment 17

NUREG-0560

STAFF REPORT ON THE GENERIC ASSESSMENT OF FEEDWATER TRANSIENTS IN PRESSURIZED WATER REACTORS DESIGNED BY THE BABCOCK & WILCOX COMPANY

Office of Nuclear Re
U. S. Nuclear Regul

Available from
National Technical Information Service
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