LIS ORIGINAL

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

should be considered flexible if the unstiffened distance between the member welded to the plate and the edge of the base plate is greater than twice the thickness of the plate. If the base plate is determined to be flexible, then recalculate the bolt loads using an appropriate analysis which will account for the effects of shear - tension interaction, minimum edge distance and proper bolt spacing. This is to be done prior to testing of anchor bolts. These calculated bolt loads are referred to hereafter as the bolt design loads.

2. Verify that the concrete expansion anchor bolts have the following minimum factor of safety between the bolt design load and the bolt ultimate capacity determined from static load tests (e.g. anchor bolt manufacturer's) which simulate the actual conditions of installation (i.e., type of concrete and its strength properties):

   a. Four - For wedge and sleeve type anchor bolts,

   b. Five - For shell type anchor bolts.

3. Describe the design requirements if applicable for anchor bolts to withstand cyclic loads (e.g. seismic loads and high cycle operating loads).

4. Verify from existing QC documentation that design requirements have been met for each anchor bolt in the following areas:

   (a) Cyclic loads have been considered (e.g. anchor bolt preload is equal to or greater than bolt design load). In the case of the shell type, assure that it is not in contact with the back of the support plate prior to preload testing.

   (b) Specified design size and type is correctly installed (e.g. proper embedment depth).

   If sufficient documentation does not exist, then initiate a testing program that will assure that minimum design requirements have been met with respect to sub-items (a) and (b) above. A sampling technique is acceptable. One acceptable technique is to randomly select and test one anchor bolt in each base plate (i.e. some supports may have more than one base plate). The test should provide verification of sub-items (a) and (b) above. If the test fails, all other bolts on that base plate should be similarly tested. In any event, the test program should assure that each Seismic Category 1 system will perform its intended function.

5. All holders of operating licenses for power reactor facilities are requested to complete items 1 through 4 within 120 days of date of issuance of this Bulletin. A reactor shutdown is not required to be initiated solely for purposes of this inspection above. Maintain documentation of any sampling inspection of anchor bolts required by item 4 on site and available for NRC inspection. Report in writing within 120 days of date of Bulletin issuance, to the Director of the appropriate NRC Regional Office, completion of your verification and describe any discrepancies in meeting items 1 through 4 and, if necessary, your plans and schedule for resolution. For planned action, a final report is to be submitted upon completion of your action. A copy of your report(s) should be sent to the United States Nuclear Regulatory Commission, Office of Inspection and Enforcement, Division of Reactor Operations Inspection, Washington, D.C. 20555. These reporting requirements do not preclude nor substitute for the applicable requirements to report as set forth in the regulations and license.

6. All holders of construction permits for power reactor facilities are requested to complete items 1 though 4 for installed pipe support base plates with concrete anchor bolts within 120 days of date of issuance of this Bulletin. For pipe support base plates which have not yet been installed, document your actions to assure that items 1 though 4 will be satisfied. Maintain documentation of these actions on site and available for NRC inspection. Report in writing within 120 days of date of Bulletin issuance, to the Director of the appropriate NRC Regional Office, completion of your review and describe any discrepancies in meeting items 1 though 4 and, if necessary, your plans and schedule for resolution. A copy of your report should be sent to the United States Nuclear Regulatory Commission, 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.

Enclosure:
List of IE Bulletins
Issued in Last
Twelve Months

IE Bulletin No. 79-02
March 8 , 1979

LISTING OF IE BULLETINS ISSUED IN LAST TWELVE MONTHS

Bulletin No.	Subject	Date Issued	Issued To
78-03	Potential Explosive Gas Mixture Accumulations Associated with BWR Offgas System Operations	2/8/78	All BWR Power Reactor Facilities with an OL or CP
78-04	Environmental Qualification of Certain Stem Mounted Limit Switches Inside Reactor Containment	2/21/78	All Power Reactor Facilities with an OL or CP
78-05	Malfunctioning of Circuit Breaker Auxiliary Contact Mechanism-General Model CR105X	4/14/78	All Power Reactor Facilities with an OL or CP
78-06	Defective Cutler-Hammer, Type M Relays With DC Coils	5/31/78	All Power Reactor Facilities with an OL or 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 1 Material Licensees
78-08	Radiation Levels from Fuel Element Transfer Tubes	6/12/78	All Power and Research Reactor Facilities with a Fuel Element transfer tube and an OL.
78-09	BWR Drywell Leakage Paths Associated with Inadequate Drywell Closures	6/14/79	All BWR Power Reactor Facilities with an OL or CP

IE Bulletin No. 79-02
March 8, 1979

LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS

Bulletin No.	Subject	Date Issued	Issued To
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	7/21/78	BWR Power Reactor Facilities for action: Peach Bottom 2 and 3, Quad Cities 1 and 2, Hatch 1, Monticello and Vermont Yankee
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
78-13	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 or CP
79-01	Environmental Qualification of Class IE Equipment	2/8/79	All Power Reactor Facilities with an OL or CP

SSINS: 6820
Accession No:
7908150164

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

August 20, 1979

IE Bulletin No. 79-02
Revision No. 1
(Supplement No. 1)

PIPE SUPPORT BASE PLATE DESIGNS USING CONCRETE EXPANSION ANCHOR BOLTS

Description of Circumstances:

The supplement to IE Bulletin No. 79-02 is intended to establish criteria for the evaluation of interim acceptability of plant operation with less than the design factors of safety for piping supports due to as-built problems, under design, base plate flexibility, or anchor bolt deficiencies.

In the reviews for system operability of the Duane Arnold and Crystal River facilities, criteria have been developed by the NRC staff that defines pipe support operability. The criteria has been applied in lieu of other analysis or evaluation. Specifically, the licensees identified problems with pipe supports in which the original design factors of safety were not met but some lesser margin was available. The design margins of four or five are intended to be final design and installation objectives but systems may be classed as operable on an interim basis with some lesser margin providing a program of restoration to at least the Bulletin factors of safety has been developed. Facilities which fall outside the operability criteria are considered to probably require a Technical Specification exception and will require review on a case by case basis.

Action to be Taken by Licensees:

For the following two cases, plant operation may continue or may begin:

a. For the support as a unit, the factor of safety compared to ultimate strengths is less than the original design but equal to or greater than two.

b. For the anchor bolts the factor of safety is equal to or greater than two and for the support steel the original design factor of safety compared to ultimate strengths is met.

The above criteria may be applied provided that the affected systems are upgraded to design margins of safety expeditiously for normally accessibile supports and by the next refueling for nonaccessible supports. Accessibility is as defined in Bulletin No. 79-14 where "normally accessible" refers to those areas of the plant which can be entered during reactor operation.

1. Any support not satisfying the criteria should be classed as inoperable and the Technical Specification action statement met unless it can be shown that the system can function in a design basis seismic event without the support.

2. Repairs to supports should result in return to the design factor of safety.

3. Operations may be continued while repairs to upgrade the system from a factor of safety equal to or greater than two with respect to design loads are performed. Consideration must be given to the effect of the repair process on support function and system operability. In other words the time the support is not functional should be limited to T.S. action statement times or the support must be determined not to cause the system to be unable to perform its function in a seismic event. The licensee should also exercise care not to take several supports on a given system out of service at the same time or cause both trains of one safeguards system to be made inoperable at the same time. Control over workmen on safety related systems during plant operation requires a high degree of control by the licensee.

4. There are no special reporting requirements for this supplement to the Bulletin; however, the reporting requirements as set forth in the regulations and licenses must be met.

IE Bulletin No. 79-02
Revision 1, Supplement No. 1
August 20, 1979

LISTING OF IE BULLETINS
ISSUED IN LAST SIX MONTHS

Bulletin No.	Subject	Date Issued	Issued To
79-18	Audibility Problems Encountered on Evacuation of Personnel from High-Noise Areas	8/7/79	All OL's for action All CP's for information
79-17	Pipe Cracks in Stagnant Borated Water Systems at PWR Plants	7/26/79	All PWR's with operating license
79-16	Vital Area Access Controls	7/26/79	All Holders of and applicants for Power Reactor Operating Licenses who anticipate loading fuel prior to 1981
79-15 (Supp. 1)	Deep Draft Pump Deficiencies	7/18/79	All Power Reactor Licensees with a CP and/or OL
79-15	Deep Draft Pump Deficiencies	7/11/79	All Power Reactor Licensees with a CP and/or OL
79-14 (Correction)	Seismic Analyses for As-Built Safety-Related Piping System	7/27/79	All Power Reactor Facilities with an OL or a CP
79-14 (Rev. 1)	Seismic Analyses for As-Built Safety-Related Piping System	7/18/79	All Power Reactor Facilities with an OL or a CP
79-14	Seismic Analyses for As-Built Safety-Related Piping System	7/2/79	All Power Reactor facilities with an OL or a CP
79-13	Cracking in Feedwater System Piping	6/25/79	All PWRs with an OL for action. All BWRs with a CP for information

LISTING OF IE BULLETINS
ISSUED IN LAST SIX MONTHS

Bulletin No.	Subject	Date Issued	Issued To
79-12	Short Period Scrams at BWR Facilities	5/31/79	All GE BWR 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 a CP
79-10	Requalification Training Program Statistics	5/11/79	All Power Reactor Facilities with an OL
79-09	Failures of GE Type AK-2 Circuit Breaker in Safety Related Systems	4/17/79	All Power Reactor Facilities with an OL or CP
79-08	Events Relevant to BWR Reactors Identified During Three Mile Island Incident	4/14/79	All BWR Power Reactor Facilities with an OL
79-07	Seismic Stress Analysis of Safety-Related Piping	4/14/79	All Power Reactor Facilities with an OL or CP
79-06C	Nuclear Incident at Three Mile Island - Supplement	7/26/79	To all PWR Power Reactor Facilities with an OL
79-06B	Review of Operational Errors and System Mis-alignments Identified During the Three Mile Island Incident	4/14/79	All Combustion Engineer-ing Designed Pressurized Water Power Reactor Facilities with an Operating License
79-06A (Rev 1)	Review of Operational Errors and System Mis-alignments Identified During the Three Mile Island Incident	4/18/79	All Pressurized Water Power Reactor Facilities of Westinghouse Design with an OL
79-06A	Review of Operational Errors and System Mis-alignments Identified During the Three Mile Island Incident	4/14/79	All Pressurized Water Power Reactor Facilities of Westinghouse Design with an OL

LISTING OF IE BULLETINS
ISSUED IN LAST SIX MONTHS

Bulletin No.	Subject	Date Issued	Issued To
79-06	Review of Operational Errors and System Mis-alignments Identified During the Three Mile Island Incident	4/11/79	All Pressurized Water Power Reactors with an OL except B&W facilities
79-05C	Nuclear Incident at Three Mile Island - Supplement	7/26/79	To all PWR Power Reactor Facilities with an OL
79-05B	Nuclear Incident at Three Mile Island	4/21/79	All B&W Power Reactor Facilities with an OL
79-05A	Nuclear Incident at Three Mile Island	4/5/79	All B&W Power Reactor Facilities with an OL
79-05	Nuclear Incident at Three Mile Island	4/1/79	All Power Reactor Facilities with an OL and 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
79-03	Longitudinal Welds Defects In ASME SA-312 Type 304 Stainless Steel Pipe Spools Manufactured by Youngstown Welding and Engineering Co.	3/12/79	All Power Reactor Facilities with an OL or CP
79-02 (Rev. 1)	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts	6/21/79	All Power Reactor Facilities with an OL or a CP
79-02	Pipe Support Base Plate Designs Using Concrete Using Concrete Expansion Anchor Bolts	3/8/79	All Power Reactor Facilities with an OL or a CP

LISTING OF IE BULLETINS
ISSUED IN LAST SIX MONTHS

Bulletin No.	Subject	Date Issued	Issued To
79-01A	Environmental Qualification of Class 1E Equipment (Deficiencies in the Environmental Qualification of ASCO Solenoid Valves)	6/6/79	All Power Reactor Facilities with an OL or CP
79-01 (Correction)	Environmental Qualification of Class IE Equipment (Deficiencies in the Environmental Qualification of ASCO Solenoid Valves)	2/28/79	All Power Reactor Facilities with an OL or CP
79-01	Environmental Qualification of Class IE Equipment (Deficiencies in the Environmental Qualification of ASCO Solenoid Valves)	2/8/79	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