OMB No.: 3150-0011
NRCB 89-02

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

July 19, 1989

NRC BULLETIN NO. 89-02: STRESS CORROSION CRACKING OF HIGH-HARDNESS TYPE 410 STAINLESS STEEL INTERNAL PRELOADED BOLTING IN ANCHOR DARLING MODEL S350W SWING CHECK VALVES OR VALVES OF SIMILAR DESIGN

Addressees:

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

Purpose:

The purpose of this bulletin is to request addressees to identify, disassemble and inspect certain types of swing check valves which may contain Type 410 stainless steel (SS) bolting material. If the Type 410 SS bolting material is of sufficiently high hardness that it is susceptible to stress corrosion cracking (SCC), or has failed, addressees are requested to take appropriate actions.

Description of Circumstances:

The occurrences discussed below have raised concerns about the use of Anchor Darling swing check valves, Model S350W, and valves of similar design with internal preloaded bolting material of ASTM Specification A193 Grade B6 Type 410 SS.

Diablo Canyon, Unit 2 - In October 1988, the licensee performed a scheduled preventive maintenance on a swing check valve in the residual heat removal (RHR) system. This valve had been successfully stroked by hand several times before the mechanic detected slight movement of the retaining block. Further investigation showed that both retaining block studs shown in Figure 1 were broken. The retaining block studs (bolts) retain the blocks that hold the valve disk assembly in place to the valve body as shown in Figure 1. The valve was an 8 inch pressure isolation valve in piping attached to the reactor coolant system hot leg. One bolt was broken at the block to valve body interface and the other bolt was broken inside the retaining block. There were signs of significant corrosion product buildup on the failed bolts. The valve was manufactured by Anchor Darling. Details of this failure are given in NRC Information Notice 88-85, "Broken Retaining Block Studs on Anchor Darling Check Valves," dated October 14, 1988.

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D.C. Cook, Units 1 & 2 - At D.C. Cook, Unit 2, during maintenance on an 8 inch Anchor Darling swing check valve installed in the low-pressure emergency core cooling system (ECCS), the licensee performed an inspection of the valve internals. One of the two internal preloaded bolts was found broken and the other cracked. As a result of this finding, the licensee inspected the corresponding check valve in the redundant low-pressure ECCS train. Again, one of the two internal preloaded bolts was found broken and the other cracked. This discovery prompted the licensee to expand the inspection to Anchor Darling check valves of the same design as those in which the degraded studs were found. This included 12 valves, all classified as pressure isolation valves, in the ECCS and RHR systems at this plant. The licensee identified one accumulator outlet check valve with a cracked bolt.

Following the licensee's decision to initiate inspection of Unit 2 check valves, Unit 1 went from power operation to hot shutdown (Mode 4) because of an unrelated event. The licensee decided to inspect the four Unit 1 check valves that were accessible in Mode 4 and found one broken bolt in each of the two check valves installed in the low-pressure ECCS. The licensee provided details of these failures in a letter dated October 28, 1988.

J.A. FitzPatrick Plant - Licensee Event Report 87-003 identifies broken bolts from the High Pressure Coolant Injection (HPCI) Terry turbine throttle valve lifting beam. The bolts of Type 410 stainless steel with hardness in the upper Rc30 range failed from intergranular stress corrosion cracking.

Discussion:

These occurrences raise questions concerning the operability and reliability of Anchor Darling Model S350W swing check valves with Type 410 SS retaining block studs and valves of similar design with internal preloaded bolting. The internal bolts of these valves are of ASTM specification A193 B6 Type 410 martensitic stainless steel with a tempering temperature of 1100°F and a specified minimum tensile strength, but no maximum specified tensile strength. The licensees determined the preliminary cause of the Type 410 SS material failure to be SCC. Three parameters determine the susceptibility of Type 410 SS to SCC: heat treatment, environment, and stress magnitude. Metal hardness is related to the heat treatment performed on the material and Rockwell hardness values below Rc26 are indicative of heat treatments that are generally less susceptible to SCC. Before Winter 1974, hardness control was exercised only through meeting the tempering temperature requirement in ASME SA193-B6. The maximum hardness requirement of the ASTM A193-B6 was incorporated into the ASME Code in Winter 1974. The current hardness requirements would probably have been met had the material actually been tempered at the required 1100°F for the appropriate time. The susceptibility of martensitic steel (B6 and B7) to SCC increases for hardness values exceeding Rc26. The B6 bolting material with limitation on maximum hardness is designated as B6X. One of the two broken bolts examined for the preliminary results indicated that the hardness was Rc36. This value is significantly higher than the desired range and is more susceptible to SCC than a properly heat-treated bolt.

The Anchor Darling check valves at Diablo Canyon are in lines filled with borated water and are not used during normal plant operation. The interior of the valve with the broken bolt was found to be rusty, while the others were clear. This may indicate that the bolt failure could have been a result of events or conditions present before system operation and that failure of the bolt may not have been a result of these valves being in a borated water environment.

Although the internal preloaded bolts of these check valves do not experience loading from valve operation or from system pressure, stress does result from initial torquing of the bolts. Because excessive preloading aides SCC, it is important that the valve maintenance manuals contain the valve vendor's torquing requirements. In addition, stress on the bolts could be produced by differential thermal expansion of the dissimilar metals (retaining block materials of Type 304 SS and bolts of Type 410 SS will result in a differential growth of approximately 50 percent). However, this consideration does not preclude using replacement stud material of similar thermal expansion as that of Type 410 SS provided that the replacement material is less susceptible to SCC.

Care should be taken when testing check valves after maintenance. Recently (May 20, 1989) Salem Unit 1 experienced a loss of residual heat removal capability while flow testing a check valve in an accumulator line. Related guidance on check valve disassembly and post-maintenance testing where the ASME Code, Section XI requirements are impractical can be found in Generic Letter 89-04, "Guidance on Developing Acceptable Inservice Testing Programs."

<u>Actions Requested</u>:

I. For all licensees of operating reactors:

   A. All licensees of operating reactors are requested to disassemble and inspect all safety-related Anchor Darling Model S350W swing check valves supplied with internal retaining block studs of ASTM specification A193 Grade B6 Type 410 SS. Licensees should review the design of other safety-related check valves to determine if similar designs and material selection to the Anchor Darling Model S350W are used. If so, such valves should be similarly inspected. The inspection by disassembly should be performed as follows:

      1. If any of the internal bolting is to be reused, it should be inspected for cracks using surface inspection techniques (penetrant or magnetic particle). Cracked bolting should be replaced and a failure analysis performed including chemical analysis to confirm material type.

2. If all suspect bolting is to be replaced with bolting of material and hardness specified in I.A.3, surface inspection and failure analysis of the old bolting may not be needed unless an unexpected failure mechanism is evident.

3. Reused and new bolting should be hardness tested for a maximum Rockwell hardness value of Rc26. Any internal preloaded bolting that does not meet the hardness requirements should be replaced by bolts of the same material with a maximum Rockwell hardness of Rc26 or by an alternate material approved by the valve manufacturer.

B. Inspections of Anchor Darling Model S350W swing check valves are requested to be performed at the next refueling outage or scheduled outage of sufficient duration (four weeks or longer) that begins 90 days after receipt of this bulletin. Documentation review to identify similar swing check valves with internal preloaded Type 410 SS bolting in the facility and the inspections are requested to be performed at the next refueling outage that begins 180 days after receipt of this bulletin.

II. For all applicants for Operating Licenses:

A. The Actions Requested are the same as I.A. above.

B. The implementation of the Actions Requested is requested to be complete before fuel loading, or, if fuel loading occurs within 90 days of receipt of this bulletin, at the first refueling outage after receipt of this bulletin.

Reporting Requirements:

Activities performed in response to this bulletin shall be documented and maintained in accordance with plant procedures for safety-related equipment and reported as follows:

1. Addressees who do not have Anchor Darling Model S350W swing check valves with Type 410 SS bolts subject to this bulletin and do not have valves of similar design with preloaded Type 410 SS bolt material shall within 180 days of receipt of this bulletin provide a letter of confirmation to the NRC of these facts.

2. Addressees who do have swing check valves subject to this bulletin shall provide a letter to the NRC within 60 days of completion of the inspections stating the number of valves inspected and the number of valves found to have service induced cracking of bolting.

The documentation of the valve inspection to be maintained by the licensee shall summarize the inspection findings and include the items listed below:

a. The number and location of subject swing check valves inspected.

b. The number of subject swing check valves where broken and/or cracked retaining block bolts are found.

c. The extent of any cracking found, the nondestructive examination methods used and the acceptance criteria employed.

d. The number of subject bolts that were replaced and type of material used.

e. The results of any failure analysis.

3. Licensees unable to meet the above schedules shall submit a report to the staff with technical justification and alternative schedules as appropriate within 30 days after the need for scheduler relief is realized.

Although not requested by this bulletin, addressees are encouraged to work collectively to address the technical concerns associated with this issue, as well as to share information regarding valves similar to Anchor Darling Model S350W.

The letters required above shall be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, under oath or affirmation under the provisions of Section 182a, Atomic Energy Act of 1954, as amended and 10 CFR 50.54(f). In addition, a copy shall be submitted to the appropriate Regional Administrator.

This request is covered by Office of Management and Budget Clearance Number 3150-0011 which expires December 31, 1989. The estimated average burden hours is 60 person-hours per valve, including assessment of the new recommendations, searching data sources, gathering and analyzing the data, and preparing the required letters. These estimated average burden hours pertain only to these identified response-related matters and do not include the time of actual implementation of physical changes consistent with the requested actions. Send comments regarding this burden estimate or any other aspect of this collection of information, including suggestions for reducing this burden, to the Records and Reports Management Branch, Division of Information Support Services, Office of Information Resources Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555; and to the Paperwork Reduction Project (3150-0011), Office of Management and Budget, Washington, D.C. 20503.

The radiation dose that would be incurred by the actions in this bulletin is strongly dependent on the location of the valves in question. The limited experience to date indicates that the dose can range from less than 0.1 person-rem per valve to about 2.5 person-rem per valve depending on the location of the valve within the plant system.

If you have any questions about this matter, please contact one of the technical contacts listed below or the Regional Administrator of the appropriate regional office.

Charles E. Rossi

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

Technical Contacts: Thomas McLellan, NRR
(301) 492-3218

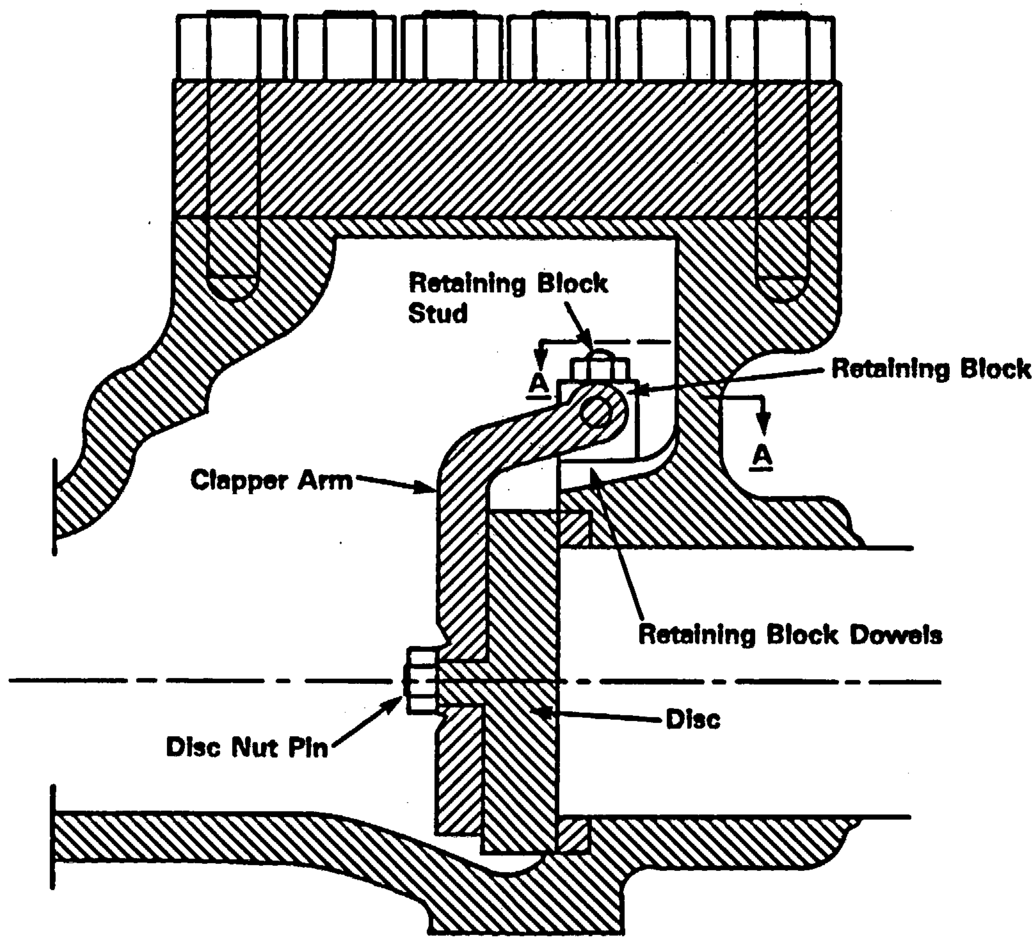
C. David Sellers, NRR
(301) 492-0930

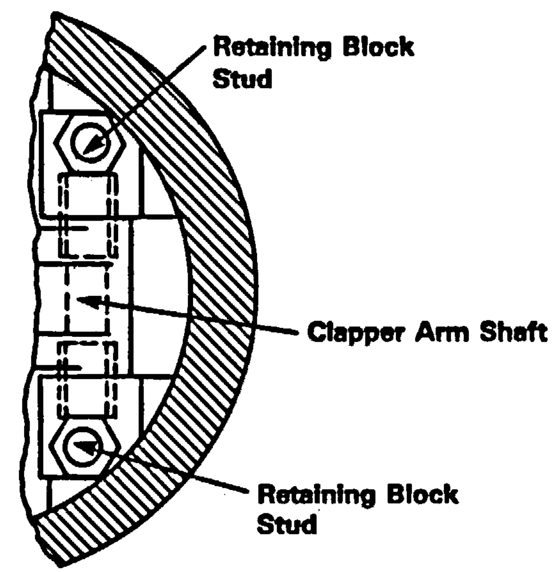
N. Prasad Kadambi, NRR
(301) 492-1153

Attachments:
1. Figure 1. D.C. Cook 1 and 2 and Diablo Canyon 2
2. List of Recently Issued NRC Bulletins

Figure 1 NRC Bulletin 89-02

D.C. COOK 1 AND 2 AND DIABLO CANYON 2





Section A-A

(Not to Scale)

Attachment 1
NRCB 89-02
July 19, 1989
Page 1 of 1

LIST OF RECENTLY ISSUED
NRC BULLETINS

Bulletin No.	Subject	Date of Issuance	Issued to
89-01	Failure of Westinghouse Steam Generator Tube Mechanical Plugs	5/15/89	All holders of OLs or CPs for PWRs.
88-08, Supplement 3	Thermal Stresses in Piping Connected to Reactor Coolant Systems	4/11/89	All holders of OLs or CPs for light-water-cooled nuclear power reactors.
88-07, Supplement 1	Power Oscillations in Boiling Water Reactors	12/30/88	All holders of OLs or CPs for BWRs.
88-11	Pressurizer Surge Line Thermal Stratification	12/20/88	All holders of OLs or CPs for PWRs.
88-10	Nonconforming Molded-Case Circuit Breakers	11/22/88	All holders of OLs or CPs for nuclear power reactors.
88-05, Supplement 2	Nonconforming Materials Supplied by Piping Supplies, Inc. at Folsom, New Jersey and West Jersey Manufacturing Company at Williamstown, New Jersey	8/3/88	All holders of OLs or CPs for nuclear power reactors.
88-08, Supplement 2	Thermal Stresses in Piping Connected to Reactor Coolant Systems	8/4/88	All holders of OLs or CPs for light-water-cooled nuclear power reactors.
88-09	Thimble Tube Thinning in Westinghouse Reactors	7/26/88	All holders of OLs or CPs for W-designed nuclear power reactors that utilize bottom mounted instrumentation.
88-08, Supplement 1	Thermal Stresses in Piping Connected to Reactor Coolant Systems	6/24/88	All holders of OLs or CPs for light-water-cooled nuclear power reactors.

OL = Operating License
CP = Construction Permit

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