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NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REGULATION

REACTOR INTERNALS BOLTING PROBLEMS

SACRAMENTO MUNICIPAL UTILITY DISTRICT

RANCHO SECO NUCLEAR GENERATING STATION

DOCKET NO. 50-312

1.0 INTRODUCTION

Since 1981, cracking of reactor vessel internals bolting has been found in a number of operating B&W plants during their routine Inservice Inspection (ISI) using ultransonic examination. These plants are Oconee 1, 2, & 3; Rancho Seco; Crystal River 3 (CR-3) and Arkansas 1 (ANO-1). The cracked bolts are made of ASTM A-286 steel and are used in the following locations:

a) Upper Core Barrel (UCB) to Core Support Shield joint
b) Lower Core Barrel (LCB) to Lower Grid assembly joint
c) Lower Thermal Shield (LTS) to Lower Grid Assembly joint
d) Surveillance Specimen Holder Tube (SSHT) to Thermal Shield Joint

Of these bolts, only the UCB and LCB bolts have core support significance. Cracked bolts have been randomly distributed around the joint without specific pattern with the location of cracks at the head-to-shank transition region of the bolt (see Enclosure 1). This area is a region of high stress concentration due to geometric discontinuity.

2.0 BACKGROUND

As a result of failures discovered at Oconee in July of 1981 in the bolts fastening the lower portion of the reactor vessel internals thermal shield to the lower grid assembly, a program for reactor vessel internal bolt inspection and repair was initiated. This program resulted in the detection in March 1983 at Rancho Seco by ultrasonic testing of abnormal indications in 19 upper core barrel bolts. Subsequently, in April 1983 ultrasonic tests (UT) at Crystal River Unit 3 showed abnormal indications in 4 lower core barrel bolts and a number of upper core barrel bolts. In May 1983, a special B&W Owners Group Task Force met with the NRC staff and subsequently submitted a report (BAW-1784) containing justification for continued operation for Davis-Besse 1, Oconee, and ANO-1. The basis was an analysis showing that, in the unlikely event of most or all of the upper or lower core barrel bolts failing, the safety of the plant will not be compromised. Sacramento Municipal Utility District (SMUD) in their letter of June 24, 1983 referenced the May 6 meeting and BAW-1784 and gave additional details on the repairs to the Rancho Seco Plant to justify its return to full power operation. While plant specific corrective measures were progressing, the bolting Task Force of the B&W Owners Group proceeded with a long range plan for generic resolution of these bolting problems.

3.0 EVALUATION

The bolting used in the upper core barrel to core support shield, the upper thermal shield to upper core barrel, the lower thermal shield to lower grid assembly, the lower core barrel to lower grid assembly, and the lower grid assembly to flow distributor are made of a nickel/chrome steel (A286) designated as SA 453 GR660, Condition A material (solution treated at 1650°F for 2 hours and age hardened at 1325°F 16 hours). All bolts originally installed in these joints were hot headed. In addition, the thermal shield bolts were heavily cold worked before head forming or threading. The bolts in these five joints generally have welded locking clips to capture the bolt and prevent bolt rotation. The yield stress of this material is in the range of 100 to 134 ksi.

One important aspect of the staff's review was the design of the reactor internal structural elements, based on defined material strength of those elements. In evaluating the causes of failures and the need for design modifications, stress levels, including those caused by stress concentrations at local areas under various loads, and loading combinations, are the primary concern. The findings of this aspect of the review based upon information provided by the licensees and the task force are the following:

1. Reactor internals are designed to withstand normal operation and accident loads. In order to assure that no separation of bolted elements will occur at a joint under these loads, high prestress ("preload") is imposed on each bolt during installation. Although Section III of the ASME Code has provided rules on the construction of core support structures, there is little guidance provided on preload, including the amount of preload and the techniques to apply the load accurately. According to the information provided, the peak stresses at the head-to-shank transition region of the initially installed bolts can be as high as 124, 108, 159 and 126 ksi in UCB, LCB, and LTS bolts, respectively, including the combined stress concentrations due to local geometric discontinuities in that area. The yield stress of A-286 steel is in the range of 100 to 134 ksi. Based on the facts that (a) the cracks were found at locations coinciding with high local stress locations, (b) the peak stresses are higher than yield stress in most cases, and (c) they are constantly subjected to flow-induced cyclic stresses, the degradation is considered to be stress assisted intergranular cracking.

   The SMUD letter of June 24, 1983, indicates that the replacement bolts in Rancho Seco have reduced the preload, such that the peak stresses after load combination and concentration effects are within yield.

   In their letter of November 3, 1983 the licensee stated that the LTS bolts were replaced with bolts which have less stress concentration effects than the original ones. We conclude that such design changes are responsive to the problems and hence acceptable. The random distribution of cracked bolt locations in a joint may be partially attributed to the high preloads and inaccurate methods of establishing initial preload.

2. BAW-1784 indicated that the surveillance by monitoring changes of neutron noise signals at regular intervals will detect core motion due to bolting failure. We feel that such measures will not be effective for detecting individual cracked bolts but will detect gross core motion caused by a significant degradation of the joint constraint at the core barrel due to rupture of a great number of bolts.

3. We agree with the licensee's conclusion that loose parts will not be generated due to ruptured core barrel bolts since the bolt heads would be held in place by the locking clips.

4. We agree that in the unlikely event of a complete severance of all core barrel bolts, the guide lugs have adequate design margin to limit the core drop to 0.54 inches. The maximum stress in the guide lugs under such a postulated event is within the ASME Section III Code limit. Lateral and rotation motion would be restrained by the bolt shanks and by the guide lugs and guide blocks. Some damage to peripheral fuel assemblies might occur, as a result of contact between the core baffle plate and the upper end fitting of the fuel assemblies. With the core in the dropped position, an additional 5 percent bypass flow is predicted. This is within the existing thermal margin limits. Control rod penetration into the fuel assembly guide tubes (6.5 inches) is sufficient to assure control rod insertion. Normal and accident loads were assessed. Simultaneous large break LOCA, and combined LOCA with SSE, occurring with a core drop were evaluated. It was concluded that a dropped core could withstand both normal and accident loads with the core maintained in a coolable geometry.

5. We agree that the number of bolts actually installed are far greater than the number of bolts needed for normal operation (i.e., only 8 symmetrically located UCB bolts are needed out of 120 bolts actually installed). Thus the design margin for normal operation is adequate.

The other aspect of the staff's evaluation of this problem as it relates specifically to Rancho Seco involved the material engineering aspects of the problem. Table 1 shows the history of testing and replacement of bolts at Rancho Seco. In summary, the following corrective action has been taken:

1. All 120 upper core barrel bolts were replaced with bolts fabricated from A-286 material which:

   A. Were manufactured by machining the heads. No hot heading was allowed. This eliminated any potential problems in the heat effected zone which remains after hot heading.

B. Manufactured using UNR thread tolerances to ensure rounded thread radii.

C. Peened under the head and in the head to shank transition region to minimize local stresses.

D. Incorporated a locking device to ensure an isolated broken bolt does not become a loose part.

E. Were installed with torque values and pre-stress methods that resulted in a peak stress of 69,000 psi which is well below the measured yield stress of 100,000-134,000 psi for the A-286 material.

2. All 96 lower thermal shield bolts were replaced with bolts fabricated from A-286 material which:

   A. Were manufactured from 1 3/4" diameter forged bar stock without hot heading. This elimates any potential problem in the heat effected zone which remains after hot heading.

   B. The design of the head to shank zone was improved by utilizing a composite under head radii to reduce the stress concentration factor.

   C. The loading on the bolts was reduced by lowering the bolt torgue from 305-335 ft-lbs to 235-265 ft-lbs.

3. None of lower core barrel bolts, flow distributer bolts, guide block bolts, and upper thermal shield bolts were replaced. Four lower core barrel bolts were removed for testing and archival material. These were not replaced.

4. A different lubricant (Fel Pro N 5000) was used on all replaced bolts.

One of the sketches presented of a typical B&W bolt failure indicated torsional failure components thus indicating a higher preload than that which was anticipated in the design of the joint. The coefficient of friction of the thread lubricant used has a very large effect upon the preload resulting from a given installation torque. For this reason, the coefficient of friction for Neolube, the thread lubricant used, was questioned.

The staff had available information indicating that Neolube has a very low coefficient of friction (0.03 to .09). B&W uses a value of approximately 0.14 and submitted lab data essentially verifying this value. Coefficient of friction tests performed by Brookhaven also verified the B&W value. These tests were run on dry Neolube as it was assumed that the bolts are installed dry.

The tentative conclusion of the B&W Owners Group Task Force in regards to the failure mechanism involved is that all bolt failures to date are attributed to intergranular stress assisted cracking in the bolt head-to-shank transition region. Although the Task Force has stated that no evidence of corrosion has been observed, the staff has not been informed of any specific effort to determine if corrosion is a contributor to the failures. The licensee, as a member of the task force, supports this tentative conclusion regarding the bolt failures at Rancho Seco and has provided information to update the BAW-1784 information specific to Rancho Seco.

Based on the considerations discussed above, the licensee has stated that Rancho Seco can proceed without undue risk to the health and safety of the public. Although this issue does not constitute an immediate safety concern, the licensee plans to actively pursue further evaluation of this issue and to participate in the ongoing efforts of the B&W Owners Group Task Force.

The need for and extent of future inspection and/or repairs will be based on the results of the Task Force investigations. The licensee will continue to keep the NRC staff informed of developments on this issue.

4.0 CONCLUSION

Based on a review and the evaluation made above, the staff concluded that continued operation of the Rancho Seco Plant for the next fuel cycle is justified because (1) the rate of occurrence of bolt cracking is apparently slow, (2) the knowledge that even failure of all of the bolts would not lead to a situation compromising safe shutdown of the plant, (3) SMUD has, during the recent refueling outage, completed an extensive inspection program and has replaced all bolts having indications of possible cracking, and (4) SMUD has committed to participate in the ongoing B&W Owners Group investigation of this problem. The staff has reasonable assurance that continued operation, in light of the reactor vessel internal bolt problem, will not endanger the health and safety of the public.

5.0 ACKNOWLEGEMENT

The following staff members were principal contributors to this Safety Evaluation: D. Sellers, S. Hou, E. Throm, and S. Miner

ENCLOSURE 1

LOCATION OF FRACTURE

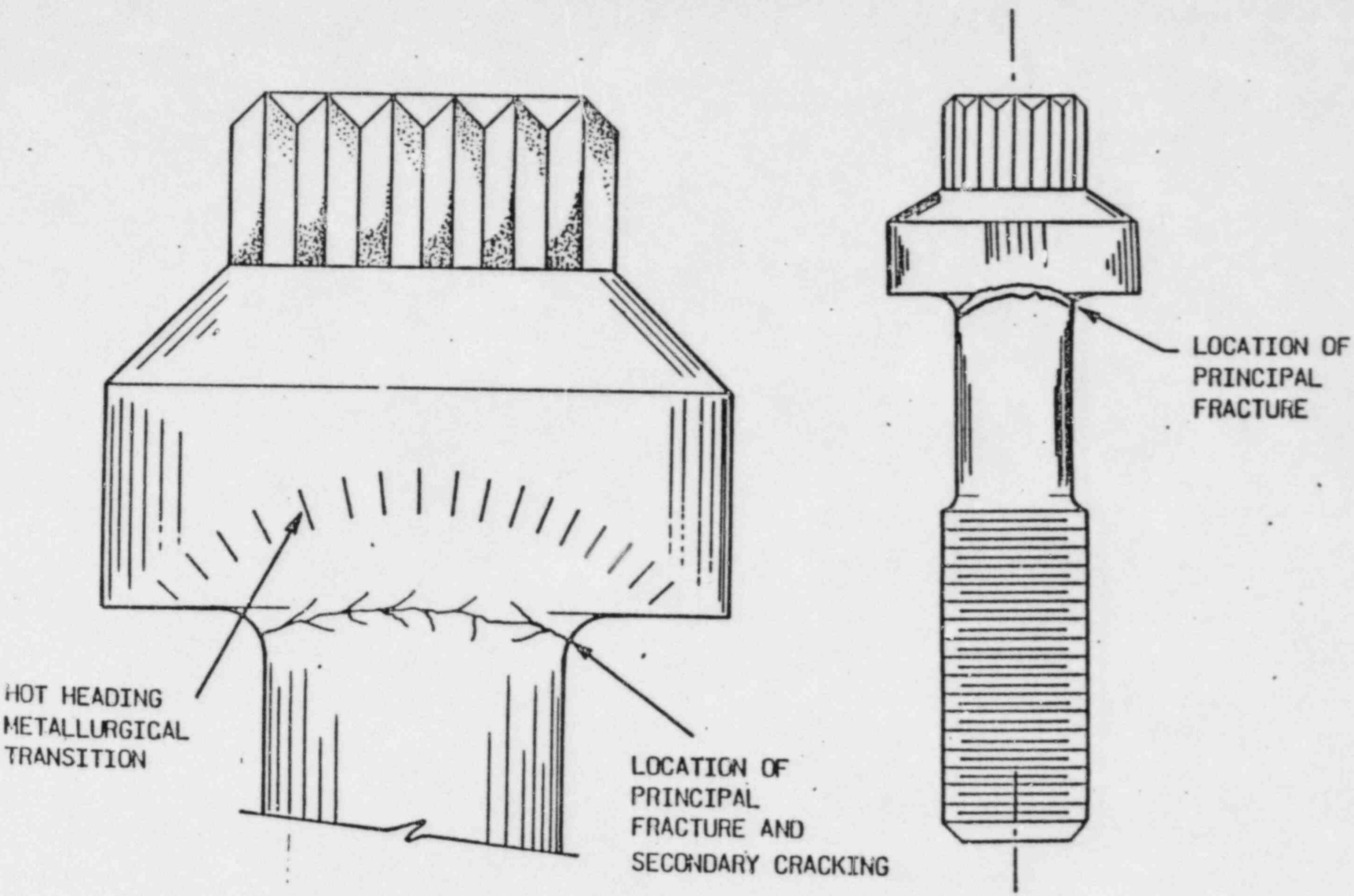

TABLE 1

REACTOR VESSEL INTERNAL BOLTS

RESULTS OF ULTRASONIC TESTS - RANCHO SECO

BOLT LOCATION	TOTAL NO. OF BOLTS	NO. OF BOLTS TESTED	NO. OF INDICATIONS	NO. OF BOLTS REPLACED
Upper Core Barrel	120	120	19	120
Lower Core Barrel	120	108	0	0
Flow Distributor to Lower Grid	96	93	0	0
Upper Thermal Shield (Upper Restraint Block)	60	60	0	0
Lower Thermal Shield	96	96	77	96