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SUBJECT: Forwards revised Relief Request SPT-4 from ASME Section XI
 ISI & sys pressure test programs. Util requesting one time
 relief request for addl exams of CRD cap screws during
 current (Unit 2 Cycle 7) refueling outage.

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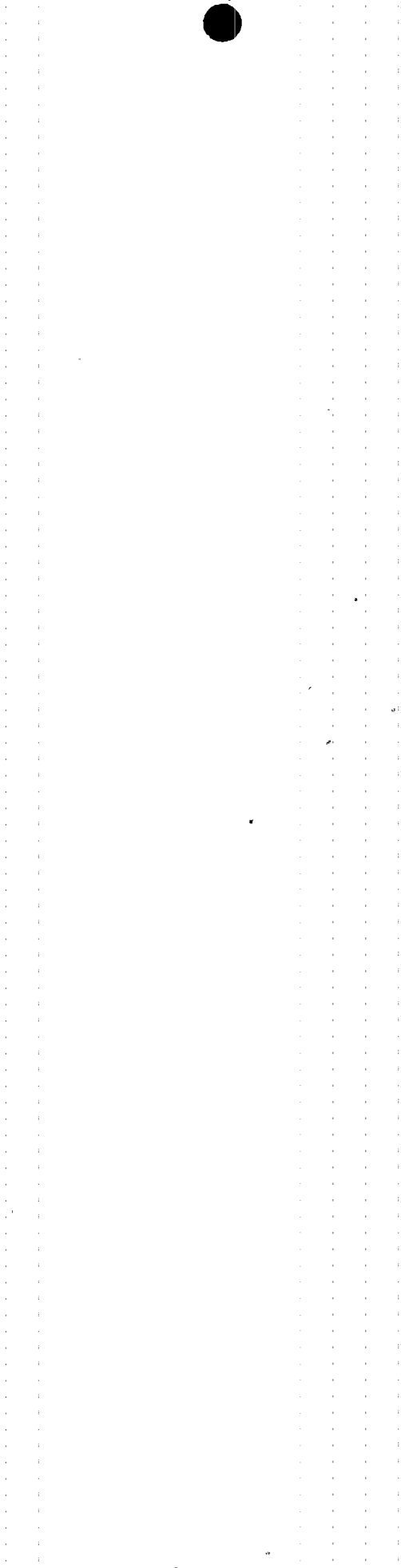
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Tennessee Valley Authority, Post Office Box 2000, Decatur, Alabama 35609

10 CFR 50.55a(3)(i and ii)

November 02, 1994

U.S. Nuclear Regulatory Commission
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Gentlemen:

In the Matter of) Docket No. 50-260
Tennessee Valley Authority)

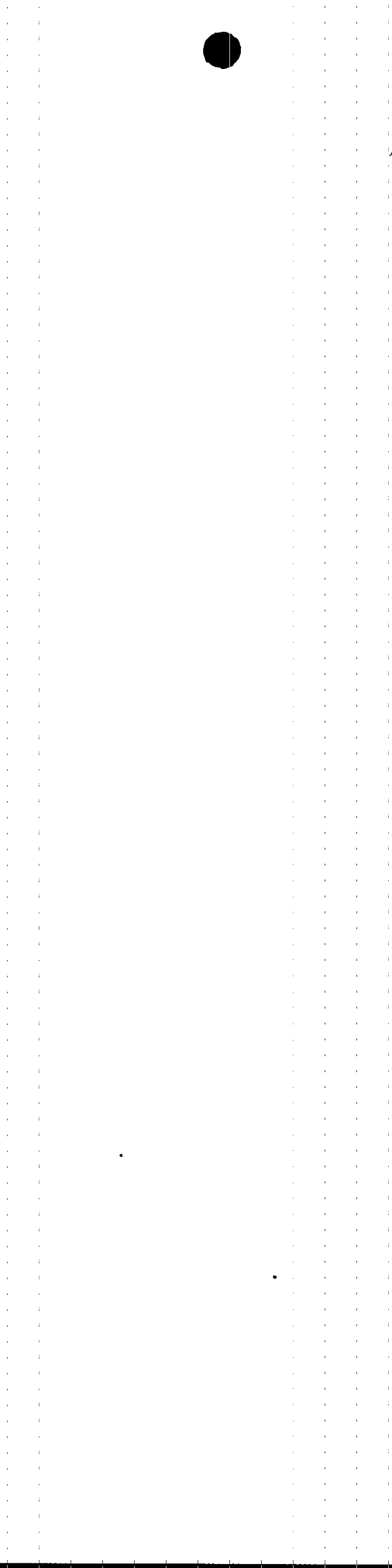
**BROWNS FERRY NUCLEAR PLANT (BFN) - AMERICAN SOCIETY OF
MECHANICAL ENGINEERS (ASME) SECTION XI INSERVICE INSPECTION
AND SYSTEM PRESSURE TEST PROGRAMS - REVISED RELIEF REQUEST
SPT-4**

In accordance with 10 CFR 50.55a(3)(i and ii), Enclosure 1 contains a revised Request for Relief from the specified Section XI inspection and pressure testing requirements of the 1986 Edition of the ASME Boiler and Pressure Code. TVA is requesting a one time relief request for additional examinations of Control Rod Drive (CRD) cap screws during the current (Unit 2 Cycle 7) refueling outage. This relief request applies to Paragraph IWB-2430(a) of the Section XI Code. In lieu of performing an expanded sample during this outage, TVA will replace the remaining 1,128 cap screws that are susceptible to stress corrosion cracking during the Unit 2 Cycle 8 refueling outage. This Relief Request is based on the BFN history of CRD leakage, the technical evaluations and testing of the existing CRD cap screws, and the reduction in expected radiation exposure associated with deferring additional inspections until after Unit 2 is chemically decontaminated during the Cycle 8 Outage.

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November 02, 1994

TVA is also requesting a permanent relief request to the IWA-5250(a) requirements for a visual examination of CRD cap screws when CRD leakage is detected during operational system pressure testing. TVA will instead evaluate and document the acceptability of the CRD leakage. If the leakage is determined to be acceptable, no further inspections will be performed prior to start-up. During the next refueling outage, the CRDs that exhibited leakage during the operational system pressure testing will have their cap screws inspected in accordance with the applicable Code requirements.

TVA is also requesting a one time relief request for the examination of the CRD cap screws that will be replaced during the Unit 2 Cycle 8 refueling outage. This relief request applies to Paragraph IWB-2500 in the Section XI Code. Since TVA will be replacing these 1,128 cap screws, performing an inservice inspection of the cap screws being replaced is of little practical benefit and is offset by the increase in personnel dose exposure and costs that are associated with these examinations. None of the 1,128 cap screws, which are susceptible to stress corrosion cracking, will be considered for re-use.

TVA identified the impractical aspects of the specified ASME Section XI requirements during Unit 2, Cycle 7 refueling outage. Review of this submittal is requested prior to startup, which is currently scheduled for November 14, 1994. TVA will inform the NRC Project Manager of any changes to this milestone in the outage schedule. This letter supersedes the relief request submitted on October 3, 1994.

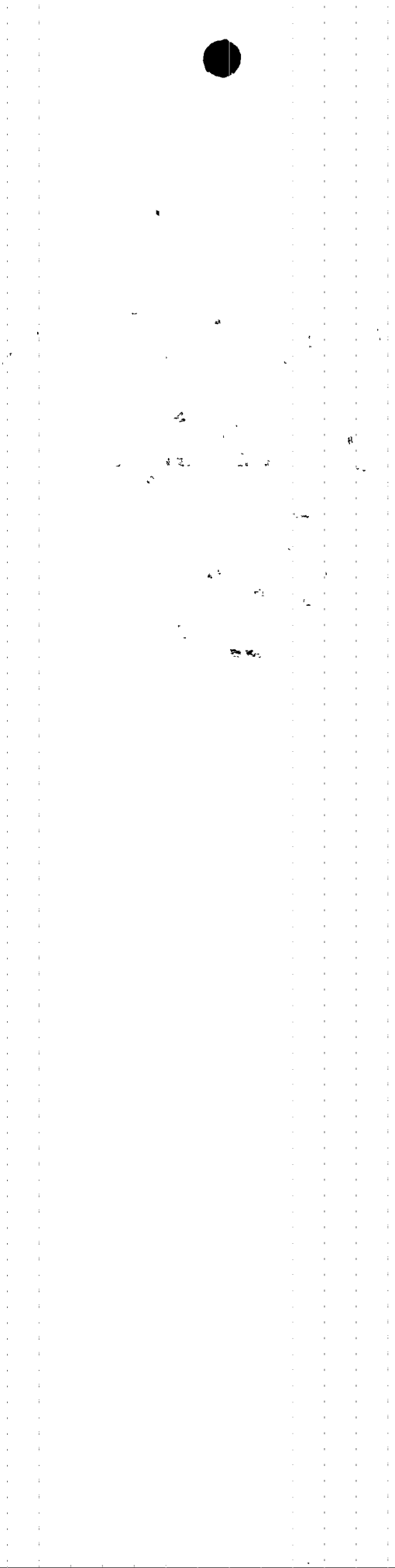
A summary of the commitments contained in this letter is provided as Enclosure 2. If you have any questions, please telephone me at (205) 729-2636.

Sincerely,



Pedro Salas
Manager of Site Licensing

Enclosures
cc: see page 3



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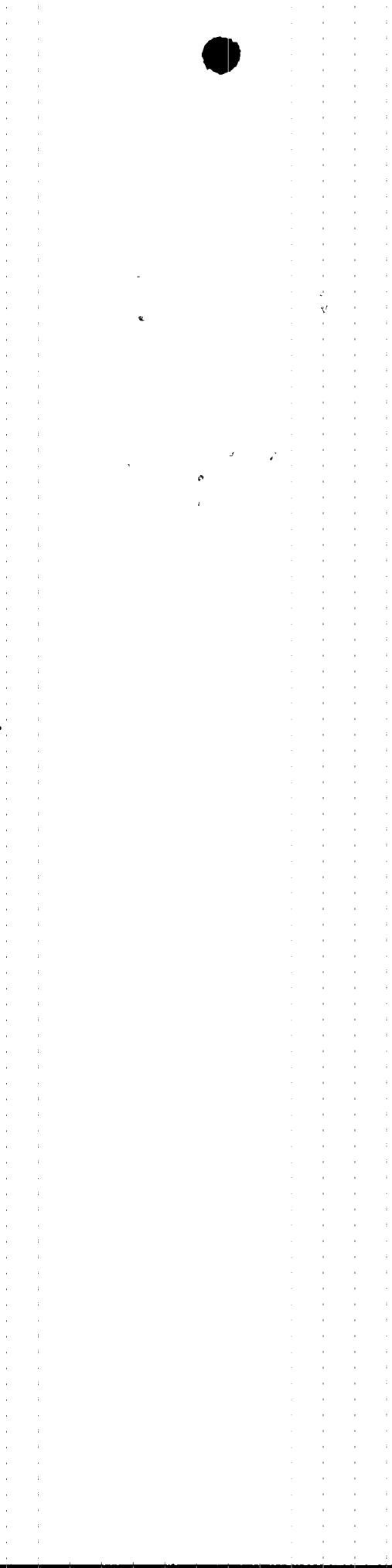
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cc (Enclosures):

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ENCLOSURE 1

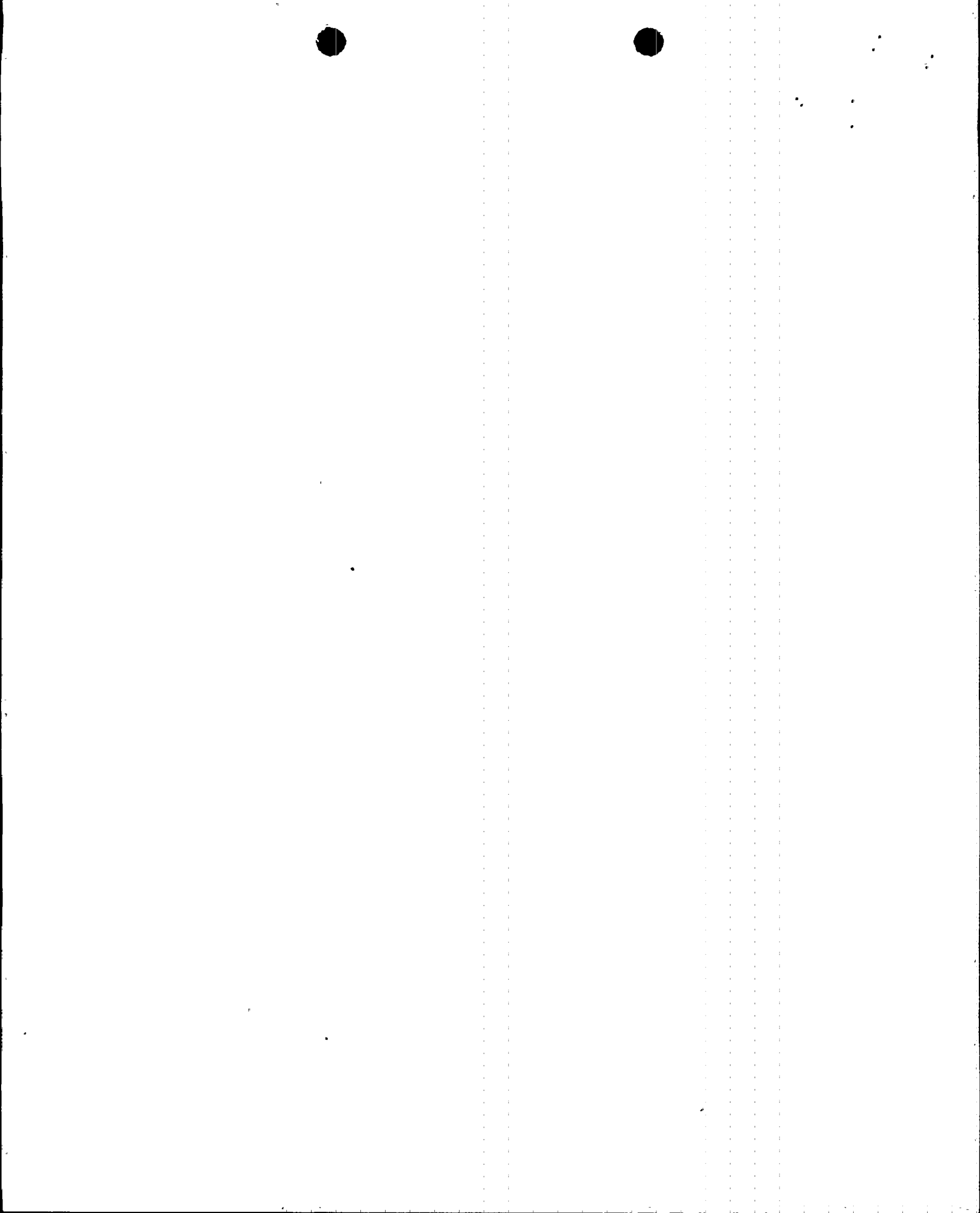
TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNIT 2

REVISED REQUEST FOR RELIEF SPT-4

System: Control Rod Drive (CRD) Hydraulic (85)
Drawing: 2-47E820-2
Components: CRD cap screws (185 CRDs per unit, 8 cap screws per CRD housing-to-flange connection)
Class: 1
Function: The CRD cap screws connect the CRD to the reactor pressure vessel (RPV) CRD nozzle flange and provides part of the primary coolant pressure boundary.

Impractical
Test
Requirements:

- I. IWA-5250(a) - The source of leakage detected during the conduct of a system pressure test shall be located and evaluated by the Owner for corrective measures as follows:
 - (2) if leakage occurs at a bolted connection, the bolting shall be removed, VT-3 visually examined for corrosion, and evaluated in accordance with IWA-3100.
- II. IWB-2430(a) - Examinations performed in accordance with Table IWB-2500-1 that reveal indications exceeding the acceptance standards of Table IWB-3410-1 shall be extended to include additional examinations at this outage. The additional examinations shall include the remaining welds, areas, or parts included in the inspection item listing and scheduled for this and the subsequent period. If the examinations for that inspection item are not scheduled in the subsequent period, the most immediate period containing scheduled examination shall be taken as the subsequent period.



III. IWB-2500(a) - Components shall be examined and tested as specified in Table IWB-2500-1. Table IWB-2510-1, Examination Category B-G-2, Item No. B7.80 - Surface examination (Visual, VT-1) of CRD housing bolts, studs, and nuts when disassembled. The acceptance standard is IWB-3517. Deferral of the inspection to the end of the interval is not permissible.

Basis for Relief (I, II and III)

The previous BFN history of Control Rod Drive (CRD) leakage, the industry and BFN specific experience with cracking of CRD cap screws, and the reduction in expected radiation exposure associated with deferring additional inspections until after Unit 2 is chemically decontaminated during the Cycle 8 Outage provides the basis for these relief requests.

(I)

Inspection of CRDs that Exhibit Leakage

In accordance with the requirements of Table IWB-2500-1, Examination Category B-P, Item No. B15.10, a leakage test of the reactor pressure vessel pressure retaining boundary is conducted prior to plant startup following each reactor refueling outage. The leakage test is conducted at nominal system pressure (1005 psig at the RPV dome) immediately prior to the startup of the unit. This examination includes the 185 CRD connections located on the bottom of the reactor pressure vessel. During re-pressurization following unit refueling, it is not uncommon to have small amounts of leakage at some of the CRD connections. This leakage is typically on the order of 1 to 30 drops per minute and the Nuclear Steam Supply System (NSSS) supplier, General Electric (GE), has informed Boiling Water Reactor (BWR) owners that leakage from these cap screw connections is a common occurrence. In most instances, this leakage stops within 8 hours of the connection being pressurized to 1000 psig.

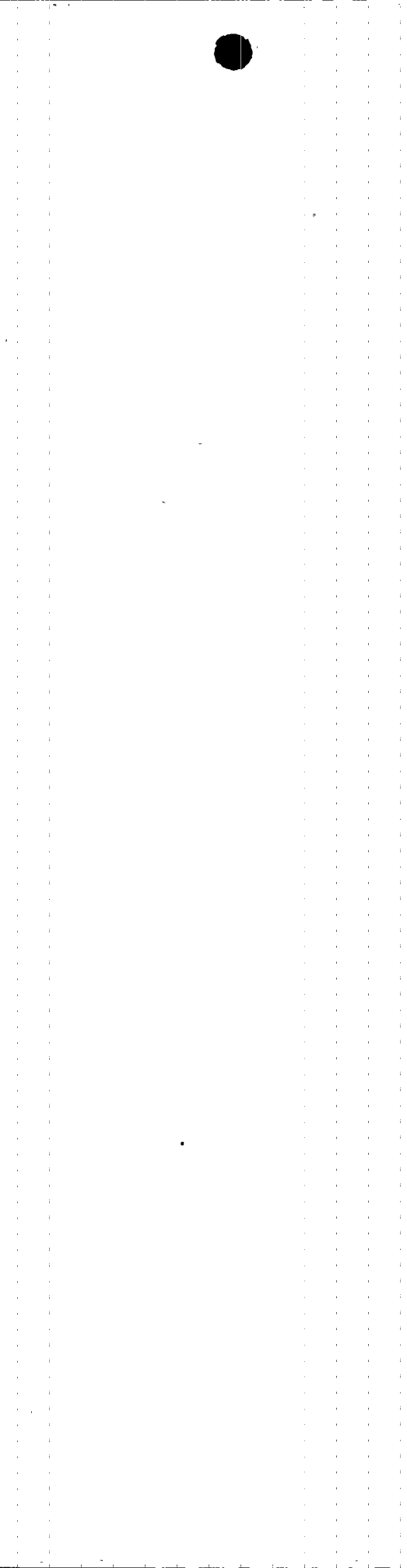


BFN inspection results provide additional evidence to support this conclusion. By TVA letter to NRC, dated April 8, 1993 (Reference 1), TVA stated that leakage from the CRD connections would be documented and evaluated based on the GE recommendations during the Class 1 component leakage test following refueling during the Cycle 6 outage. The Unit 2 Cycle 6 inspection showed 36 CRDs were initially leaking during the RPV System Leakage Test. Maintenance is normally recommended for leaks that are greater than 30 drops per minute (DPM), which do not exhibit a decreasing trend. Two of the three worst case leaking CRDs quickly showed a decrease to less than 30 DPM and the remaining worst case CRD leaker showed a leakage of approximately 40 DPM with a decreasing trend. The leakage rates for the other 33 CRDs were well below 30 DPM. These leakage rates were evaluated by TVA and GE and determined to be acceptable.

TVA's April 8, 1993 letter requested a deferral until the next refueling outage for the VT-3 examination of the eight cap screws at the 36 bolted CRD connections that exhibited leakage. This request was modified on May 5, 1993 (Reference 2) to only address Unit 2. The request was approved for Unit 2 in NRC letter to TVA, dated May 21, 1993 (Reference 3).

Only 4 of the 36 connections that were leaking at the beginning of Unit 2 Cycle 7 were leaking at the end of Unit 2 Cycle 7 operation. One additional leaking CRD was identified that was not leaking at the beginning of the cycle.

The relatively small increase in safety that could be attributed to the performance of a visual examination of the CRD cap screws when CRD leakage is detected during an operational system pressure test is offset by the increase in personnel dose exposure (approximately 0.1 Man/Rem per CRD) and cost that are associated with these examinations. Instead, TVA will evaluate and document the acceptability of the CRD leakage. If the leakage is determined to be acceptable, no further inspections will be performed prior to start-up. During the next refueling outage, the CRDs that exhibited leakage during the operational system pressure testing will have their cap screws inspected in accordance with Code requirements.



(II and III) Industry Experience with Cap Screw Cracking

Industry experience as documented in GE Nuclear Energy Services Information Letter (SIL) No. 483, Revision 2, dated August 5, 1992, has shown that the older CRD cap screws are susceptible to stress corrosion cracking. Due to this susceptibility, GE recommended inspection, and if indications were found, these cap screws should be replaced with the new design that includes higher strength and more corrosion resistant material cap screws and the new design washer to facilitate drainage.

As documented in SIL 483, GE has determined that CRD cap screw cracking does not generically effect structural integrity and plant safety. This evaluation is based in part on the following:

1. three uniformly distributed uncracked cap screws are capable of supporting the CRD loads, and the probability that through-wall cracks will occur in five or more cap screws on a single CRD is extremely small;
2. if such a failure were to occur, leakage at the connection would precede failure, and the leak detection system and/or drywell temperature monitoring system would detect this leakage at very low leakage rates;

[It should be noted that Browns Ferry Technical Specification 3.6.C.1 requires unidentified reactor coolant leakage into primary containment not exceed 5 gallons per minute (gpm). In addition, total reactor coolant system leakage into primary containment is prohibited from exceeding 25 gpm. The average Unit 2 Cycles 6 and 7 unidentified and identified leakage rates were well below these limits.]

3. the CRD support structure under the reactor vessel would allow the CRD to drop a maximum of one inch in the event of total bolted joint failure;
4. the evaluation of the loss of one CRD from any cause has been considered in the plant safety analysis report.

BFN Experience with Cap Screw Cracking

The revised cap screw design is being incorporated at Browns Ferry. The cap screws for 26 CRD units were changed during the Unit 2 Cycle 6 refueling outage. The cap screws for eighteen CRD units were changed to the new bolting design during the Unit 2 Cycle 7 refueling outage.

The cap screws, which are removed, are examined in accordance with Section XI of the ASME Boiler and Pressure vessel code. None of the 208 cap screws examined during the Unit 2 Cycle 6 refueling outage exhibited indications of cracking.

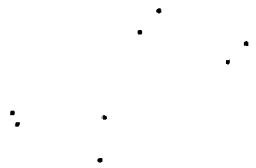
In the relief request, dated September 1, 1994 (Reference 4), TVA provided the Staff with the results of additional reviews of this issue, which concluded that bolt failures have a low probability of occurrence on the CRD cap screws at BFN for the reasons provided below:

1. Approved lubricants are used at BFN and are controlled by procedures. The primary lubricant for this application is Never-Seez, which does not contain molybdenum disulfide. Molybdenum disulfide has been identified as a contributor to cracking. In addition, the atmosphere in the drywell is required by Technical Specifications to be inerted with nitrogen during power operations. This would deprive the CRD cap screws of free oxygen that would aid in chemical and stress corrosion cracking.
2. The CRD cap screws at BFN are torqued under administrative controls to 350 foot pounds, which results in a preload stress of approximately 50 percent of the yield strength.
3. Chemical attack from borated water can lead to bolt failures. Unlike pressurized water reactors, BFN does not use borated water in its primary coolant system for reactivity control during normal operating conditions. The reactor coolant system used demineralized water and is monitored for chemical composition and contaminants.



During the review of the September 1, 1994 relief request, the Staff verbally requested additional information, including the performance of an under-vessel inspection of the CRDs immediately after the start of the Unit 2 Cycle 7 refueling outage. Response to the Staff's requests were provided in a revised relief request, dated October 3, 1994 (Reference 5). The results of this inspection support the GE judgment that CRD leakage is reduced or stops after the system achieves normal operating pressures and temperatures.

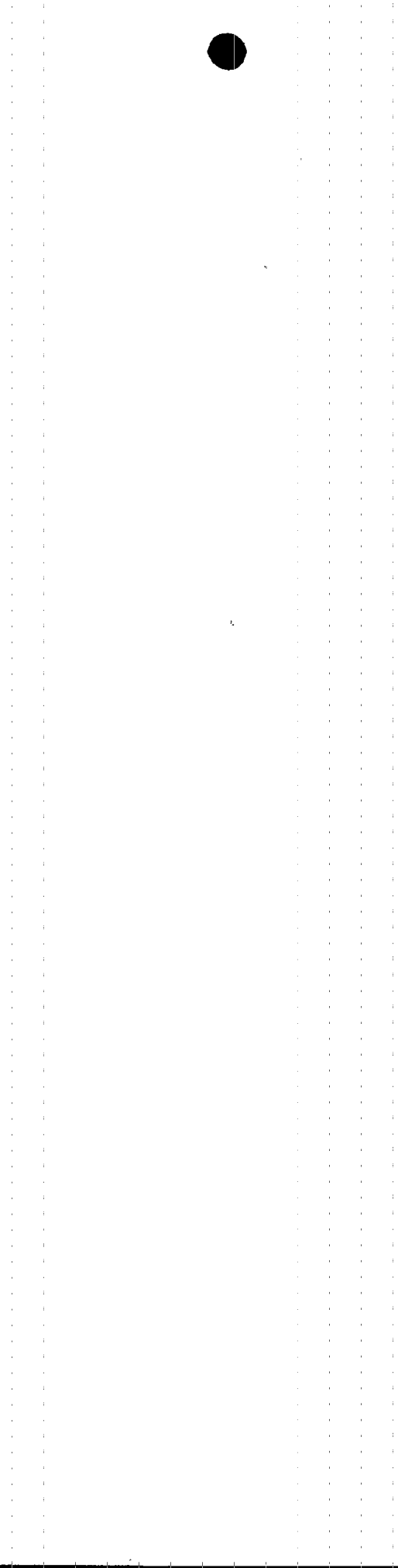
Of the 152 cap screws examined during the Unit 2 Cycle 7 refueling outage, either because the CRDs were being replaced or the cap screws were being inspected due to CRD leakage observed during the under-vessel inspection, only 8 cap screws showed indications of cracking in excess of the inspection criteria provided in Paragraph IWB-3410 of the 1986 Edition of the ASME Code. All 8 cap screws in those CRDs that were being replaced or had cap screws that were being inspected due to CRD leakage, were replaced with the higher strength and more corrosion resistant material cap screw and the new design washer to facilitate drainage. A record of the CRD leakages during Unit 2 Cycles 6 and 7 is provided in Table 1 for those CRDs, whose bolts exhibited indications of stress corrosion cracking.



Two bolts, with visual indications in excess of the ASME allowables, were found in both CRDs 06-27 and 18-39. One bolt, with visual indications in excess of the ASME allowables, was found in CRDs 06-19, 18-35, 22-19, and 34-59. However, when the bolts from CRD 06-19 were bagged, identification of which bolt had visual indication in excess of the ASME allowables, was lost. There were three bolts from CRD 06-19 that had indications, either above or below the ASME allowables. Therefore, a total of ten bolts (three bolts from CRD 06-19 and the remaining seven bolts with definite indications in excess of ASME allowables) were submitted for further laboratory analysis. Therefore, this ten bolt sample included the eight bolts that were originally identified with visual indications in excess of the ASME allowables. Details regarding the indications, hardness, and tensile testing results for these cap screws is provided in Table 2.

The ten CRD cap screws exhibited corrosion pitting and numerous linear crack-like indications in and near the transition region between the head and shank. Although the fluorescent magnetic particle examination indicated eight of the cap screws to possess circumferential indications greater than $\frac{1}{4}$ inch in length, observation under a stereo microscope revealed the cracks to be short, discrete and typically non-connected. The close proximity of the numerous cracks in a given region provided the appearance of a continuous indication.

Each of the eight cap screws, which did not meet the flaw acceptance criteria, were sectioned perpendicularly through the largest observable surface indication in order to expose the nature and depth of the underlying flaw. Each cross-section was micrographed. Shallow, multi-branching blunted cracks were typically found. In all cases, the cracks contained observable oxides and had been arrested. In many cases, the underlying cavity had the appearance of an elongated corrosion pit rather than a stress corrosion crack. These flaws appeared to be previous shallow stress corrosion cracks, perhaps initiated in the bottom of a corrosion pit, that had subsequently arrested while pitting corrosion continued.



Each of the ten cap screws were evaluated for hardness. Rockwell hardness values ranged from 29 to 31.5 HRC. These values indicate adequate tensile strength with good ductility for this alloy. Two cap screws (from CRDs 06-19 and 18-35), which possessed a large region of circumferential defects, were subjected to a full size axial tensile test. Both met the minimum load requirements of 125 ksi imposed by the fastener specification. Chemical analysis showed all required elements to be within the specification for the specified steel, with the exception of a single carbon analysis that showed 0.01 percent below the specification requirements. This deviation is not significant to the structural integrity of the cap screw or a factor for the observed cracking.

In accordance with Paragraph IWB-3142 of the 1986 Edition of the ASME Code, TVA and GE have evaluated the cap screws with indications of stress corrosion cracking and have determined that this specific condition did not effect structural integrity and plant safety for the following reasons:

1. The observed indications were reviewed by GE. The indications were consistent with the overall BWR experience in terms of crack morphology and depth and it was concluded that the indications did not represent a new phenomena.
2. The results of the hardness, tensile and metallurgical testing and analysis show the BFN cap screws are within nominal tolerances for the specific steel specified for this application.
3. The maximum depth of the observed cracking was 0.055 inches. GE SIL No. 483, Revision 2, states that the deepest crack measure to date was 0.077 inches. A crack that deep (0.077 inches) would not prevent an individual bolt from performing its structural function.



4. Conservatively assuming all the cracks would grow at a rate of 0.05 inches per 18 month cycle and the cracks extended 360 degrees around the bolt, there would be sufficient bolt area to meet the ASME Code stress limits at the end of the next cycle.
5. The low frequency of cap screw cracking observed at BFN.
6. Full size tensile testing performed on two cap screws, that were selected based on the degree of their surface indications, demonstrated that the existing cracks did not influence the failure mode or reduce the fastener load carrying capacity. Fracture of the screws occurred in the shank, approximately midway between the threaded bolt shank and the location of the cracking in the head-to-shank transition region. The failure in both screws was ductile, with significant necking of the section.

There are a total of 185 CRDs for Unit 2. Of these, 44 CRDs have had their cap screws replaced with the new design. In order to minimize future stress corrosion cracking in the CRD cap screws, TVA will replace the stress corrosion cracking susceptible cap screws in the remaining 141 CRDs during the Unit 2 Cycle 8 refueling outage with the new design that includes a higher strength material cap screw and the new design washer to facilitate drainage.

The relatively small increase in safety that could be attributed to expanding the examination of additional cap screws is offset by the increase in personnel dose exposure (approximately 0.1 Man/Rem per CRD or approximately 14 Man/Rem if the entire remaining population of cap screws requires examination) and costs that are associated with these examinations. Reduction in expected radiation exposure are associated with deferring additional inspections since TVA intends to chemically decontaminate during the Cycle 8 Outage. In lieu of this expanded sample, TVA will replace the remaining 1,128 cap screws that are susceptible to cracking during the Unit 2 Cycle 8 refueling outage.



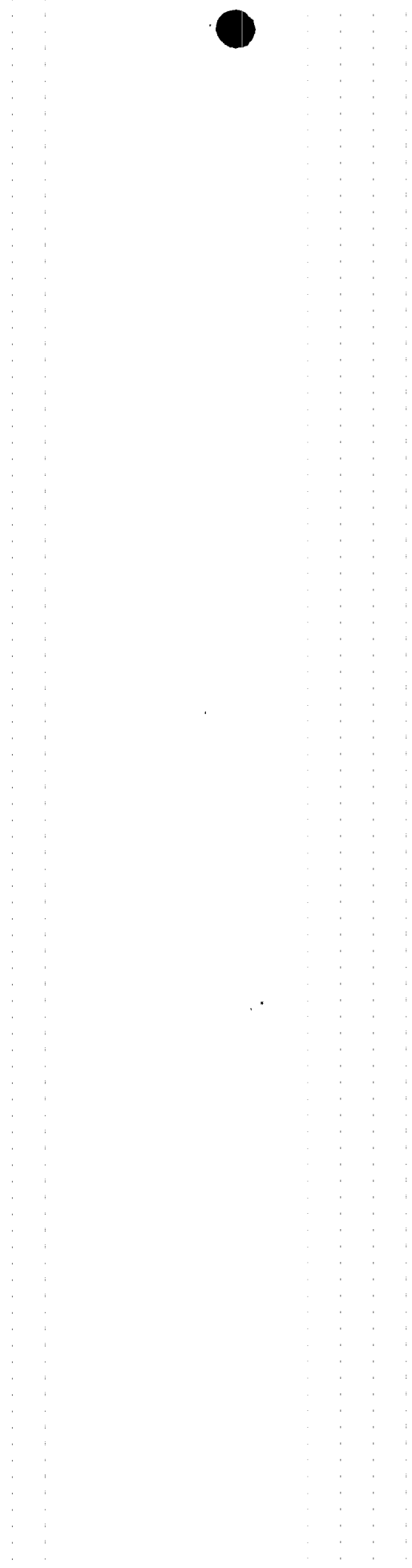
Since TVA will be replacing the remaining 1,128 cap screws that are susceptible to stress corrosion cracking during the Unit 2 Cycle 8 refueling outage, performing an inservice inspection of the cap screws being replaced is of little practical benefit and is offset by the increase in personnel dose exposure and costs that are associated with these examinations. None of the 1,128 cap screws that are susceptible to stress corrosion cracking will be considered for re-use.

Alternate
Testing:
(I)

During the Unit 2 Cycle 7 and subsequent refueling outages, if CRD leakage is observed during an operational system pressure test, TVA will evaluate and document the acceptability of the CRD leakage. If the leakage is determined to be acceptable, no further inspections will be performed prior to start-up. During the next refueling outage, the CRDs that exhibited leakage during the operational system pressure testing will have their cap screws inspected in accordance with Code requirements. VT-1 inspections will also be performed on the cap screws from the CRD connections that are disassembled for routine maintenance. A fluorescent magnetic particle surface examination will be performed in accordance with ASME Section XI if determined necessary as a result of the VT-1 examination.

(II)

During the Unit 2 Cycle 7 refueling outage, 8 of 152 CRD cap screws showed indications of cracking in excess of the ASME code allowables. In lieu of performing an expanded sample of CRD cap screws during this outage, TVA will replace the remaining (1,128) stress corrosion cracking susceptible cap screws during the next refueling outage (Unit 2 Cycle 8) with the new design that includes a higher strength material cap screw and the new design washer to facilitate drainage.



(III)

Since TVA will be replacing the 1,128 stress corrosion cracking susceptible cap screws, performing an inservice inspection of the cap screws being replaced is of little practical benefit and is offset by the increase in personnel dose exposure and costs that are associated with these examinations. None of the 1,128 cap screws, which are susceptible to stress corrosion cracking, will be considered for re-use.

References:

1. TVA letter to NRC, dated April 8, 1993, American Society of Mechanical Engineers (ASME) Section XI Inservice System Pressure Test Program
2. TVA letter to NRC, dated May 5, 1993, Units 1 and 3 Withdrawal of Control Rod Drive (CRD) Request for Relief from the American Society of Mechanical Engineers (ASME) Section XI Inservice System Pressure Test Program
3. NRC letter to TVA, dated May 21, 1993, Safety Evaluation of Requests for Relief from American Society of Mechanical Engineers Boiler and Pressure Vessel Code Requirements
4. TVA letter to NRC, dated September 1, 1994, American Society of Mechanical Engineers (ASME) Section XI Inservice System Pressure Test (SPT) Program - Revised Relief Request SPT-4
5. TVA letter to NRC, dated October 3, 1994, American Society of Mechanical Engineers (ASME) Section XI Inservice System Pressure Test (SPT) Program - Revised Relief Request SPT-4

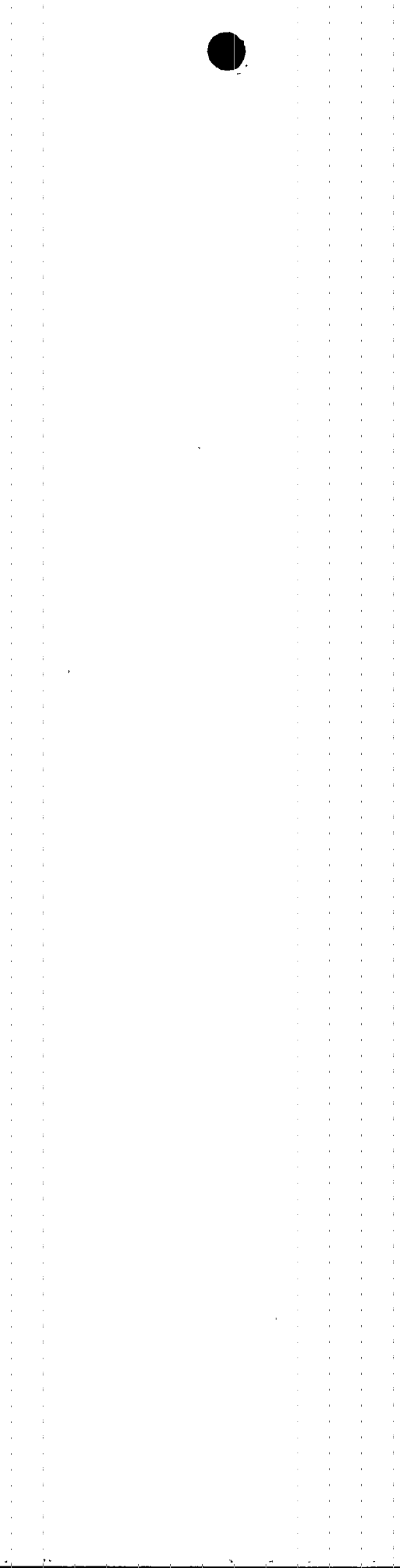


TABLE 1
HISTORY OF CONTROL ROD DRIVE (CRD) LEAKAGES
DURING UNIT 2 CYCLES 6 AND 7 FOR CAP SCREWS
WITH INDICATIONS OF STRESS CORROSION CRACKING

CRD	CYCLE 5 HYDROSTATIC LEAK TEST RESULTS	CYCLE 6 SYSTEM LEAK TEST RESULTS	CYCLE 7 UNDER-VESSEL INSPECTION RESULTS
06-19	No leakage identified.	No leakage identified.	No leakage identified.
06-27	Initially 6 to 12 drops per minute (DPM).	Initially less than 1 DPM.	Two DPM.
18-35	No leakage identified.	No leakage identified.	No leakage identified.
18-39	Initially 6 to 12 DPM.	Initially 4 DPM.	One DPM.
22-19	No leakage identified.	No leakage identified.	No leakage identified.
34-59	No leakage identified.	Initially less than 1 DPM.	Less than 1 DPM.



TABLE 2
DETAILS OF CONTROL ROD DRIVE (CRD) CAP SCREWS
WHICH SHOWED UNACCEPTABLE INDICATIONS OF STRESS CORROSION CRACKING

CRD	LOCATION OF INDICATION	MAXIMUM CRACK DEPTH DETERMINED BY METALLOGRAPHY	MAXIMUM INDICATED LENGTH OF A SINGLE CRACK DETERMINED BY MAGNETIC PARTICLE TESTING	COMPOSITION CIRCUMFERENTIAL LENGTH OF INDICATION OBSERVED BY VISUAL TEST	HARDNESS ⁽¹⁾	RESULTS OF TENSILE TESTING
06-19 Bolt 1	Shank to head transition.	0.0043 in.	1/8 in. ⁽²⁾	1/4 and 1/8 in.	31.4 HRC	125.7 KSI
06-19 Bolt 2	Root of 10th thread from end of bolt.	0.0020 in.	7/16 in.	7/16 in.	29.0 HRC	Not Tested
06-19 Bolt 3	Shank to head transition.	0.0106 in.	9/16 in.	2 in.	29.9 HRC	Not Tested
06-27 Bolt 1	Shank to head transition, mid-shank region.	0.0434 in.	7/32 in. ⁽²⁾	1/8 in.	29.7 HRC	Not Tested
06-27 Bolt 2	Shank to head transition.	0.0220 in.	5/16 in.	1 1/2 in.	30.0 HRC	Not Tested
18-35	Shank to head transition.	0.0287 in.	1/2 in.	2 1/4 in.	31.4 HRC	128.4 KSI



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TABLE 2
 DETAILS OF CONTROL ROD DRIVE (CRD) CAP SCREWS
 WHICH SHOWED UNACCEPTABLE INDICATIONS OF STRESS CORROSION CRACKING

CRD	LOCATION OF INDICATION	MAXIMUM CRACK DEPTH DETERMINED BY METALLOGRAPHY	MAXIMUM INDICATED LENGTH OF A SINGLE CRACK DETERMINED BY MAGNETIC PARTICLE TESTING	COMPOSITION CIRCUMFERENTIAL LENGTH OF INDICATION OBSERVED BY VISUAL TEST	HARDNESS ⁽¹⁾	RESULTS OF TENSILE TESTING
18-39 Bolt 1	Shank to head transition.	0.0252 in.	$\frac{1}{8}$ in.	$\frac{1}{8}$ in.	28.8 HRC	Not Tested
18-39 Bolt 2	Shank to head transition.	0.0550 in.	$\frac{5}{16}$ in.	$\frac{1}{8}$ in.	30.5 HRC	Not Tested
22-19	Shank to head transition.	0.0197 in.	$\frac{1}{4}$ in.	$\frac{7}{8}$ and $\frac{9}{16}$ in.	31.5 HRC	Not Tested
34-59	Shank to head transition. Also root of 5th, 6th, 8th, and 9th thread from end of bolt.	0.0110 in.	$\frac{9}{32}$ in. (in 5th root)	$\frac{9}{16}$ in. Random root location.	30.8 HRC	Not Tested

⁽¹⁾ - The value reported is an average of at least three readings.

⁽²⁾ - Acceptable indication length ($< \frac{1}{4}$ in.).



ENCLOSURE 2
TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNIT 2

SUMMARY OF COMMITMENTS

1. During the Unit 2 Cycle 7 and subsequent refueling outages, if Control Rod Drive (CRD) leakage is detected during operational system pressure testing, TVA will evaluate and document the acceptability of the CRD leakage. If the leakage is determined to be acceptable, no further inspections will be performed prior to start-up. During the next refueling outage, the CRDs that exhibited leakage during the operational system pressure testing will have their cap screws inspected in accordance with Code requirements.

2. In order to minimize future stress corrosion cracking in the CRD cap screws, TVA will replace the stress corrosion cracking susceptible cap screws in the remaining 141 CRDs during the Unit 2 Cycle 8 refueling outage with the new design that includes a higher strength material cap screw and the new design washer to facilitate drainage.



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