WOLF CREEK
NUCLEAR OPERATING CORPORATION

John A. Bailey
Vice President
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March 22, 1988

ET 88-0043

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Subj: Docket No. 50-482: Request for Relief from Certain Requirements Delineated in ASME Section XI

Gentlemen:

The purpose of this letter is to transmit two requests for relief from certain requirements delineated in Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. The first request concerns a case in which a code required N-2 Manufacturers Data Form is missing for a Reactor Coolant Pump Seal Water Injection Throttle Valve. The second request concerns material which does not fully meet ASME code requirements which was used to manufacture the packing box for a pressurizer spray valve which does not meet code requirements. These requests would allow continued operation until the next outage of sufficient duration at proper plant conditions to replace these components. The details of the relief requests have been discussed with Mr. Paul O'Connor and other members of the Nuclear Regulatory Commission Staff.

Enclosed is a check (No. 1018) for $150.00 application fee required by 10 CFR 170.21. If you have any questions concerning this matter, please contact me or Mr. O. L. Maynard of my staff.

Very Truly Yours,

John A. Bailey

John A. Bailey
Vice President
Engineering and Technical Services

JAB/jad

Attachment

Enclosure

cc: P. W. O'Connor (NRC), 2 w/a
R. D. Martin (NRC), w/a
B. L. Bartlett (NRC), w/a

A047
1/1
w/check $150
1018

Attachment 1 to WM 88-0043

Component

Valve BG V200 (ASME Code Class 2) - Reactor Coolant Pump 'C' Seal Water Injection Throttle Valve

Description

The reactor coolant pump seal water injection is part of the Chemical and Volume Control system which continuously supplies filtered water to each of the reactor coolant pump seals. Seal water flow is throttled to approximately 8-12 gpm per pump by valves BG-V198, V199, V200, and V201. In addition, these valves serve a flow-limiting function to prevent run-out of the safety injection/centrifugal charging pumps and to limit flow to an affected loop following a LOCA. During Refuel II, valve BG-V200 had the stem/disc assembly replaced (these items are pressure retaining per definition of the ASME Code). After the resumption of power operation, it was discovered that the N-2 Manufacturers Partial Data Report does not exist for these parts as required by the ASME code.

Requested Relief

Article IWA-7210 of ASME Section XI states, in part, "Replacements shall meet the requirements of the edition of the Construction Code to which the original component or part was constructed...". This valve was originally constructed to ASME Section III which in Article NA 3130 (f) requires, "the Manufacturer or the Installer shall be responsible for Code compliance of the completed item or installation including code symbol stamping and providing the Data Report Form is properly executed and countersigned by the Inspector".

10 CFR 50.55a (g)(5)(iii) states that, "If the licensee has determined that conformance with certain code requirements is impractical for its facility, the licensee shall notify the Commission and submit, as specified in paragraph 50.4, information to support the determinations". Wolf Creek Nuclear Operating Corporation (WCNOC) has determined that the code requirement to have the Data Report Form is impractical since this form does not exist for this valve. The Commission was notified on March 18, 1988, via telephone, of this situation and information to support the determination is provided below.

Basis for Relief

Upon discovery of the code noncompliance, a detailed review of the documentation provided with the stem/disc assembly was performed. This review revealed that the design requirements for the stem/disc assembly are correct and the material met the code requirements. In addition, the documentation showed that the welds were NDE examined and accepted.

This failure to meet the code requirements is considered to be a documentation problem and WCNOC does not have any technical concerns with this issue. As such, continued operation of the plant is considered acceptable with relief from this code requirement. However, to restore this component to compliance with the ASME code requirements, the stem/disc assembly shall be replaced at the next outage of sufficient duration at proper plant conditions. In any case, this assembly will be replaced no later than startup following the next refueling outage.

Attachment 2 to WM 88-0043

COMPONENT

Valve BB PCV 455B American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code Class 1) - Pressurizer Spray Valve

DESCRIPTION

Component Description and Operating History

Valve BB PCV 455B is one of two 4 inch pressurizer spray valves. It is an ASME Code Section III Class 1 air-operated ball valve built by Fisher Contrcls and supplied by Westinghouse. The packing box assembly consists of the packing box and attached leakoff pipe nipple. The packing box assembly is the structural interface between the valve body and the actuator assembly and forms part of the valve pressure boundary. It carries the weight and other operational loadings imposed by the actuator assembly. The packing box contains the packing seal between the shaft and packing box.

This valve has performed satisfactorily since initial installation except as noted below. During plant operations it has been subjected to routine operational function tests, leakage tests and normal preventative maintenance.

On January 9, 1987, a body-to-packing box leak was discovered. The packing box-to-body bolting was torqued to vendor specifications, 220 ft-lbs, without stopping the leakage. The decision was made to temporarily seal the valve by leak-seal compound injection. This was successfully completed on January 19, 1987. A replacement packing box assembly was ordered and installed during the Refuel II Outage on October 9, 1987. Post maintenance testing was completed on December 28, 1987, prior to plant start-up.

Procurement and Installation of Replacement Part

Upon determination that replacement of the subject valve's packing box assembly was required, WCNOC Purchase Order No. 521144 was issued on August 3, 1987. This safety-related P.O. directed Westinghouse to procure two each of the packing box assemblies in accordance with their "Renewal Parts Procurement and Supply System," OPR 405-5. Westinghouse is qualified by WCNOC to procure replacement parts using this system and when invoked, it is intended to provide parts of the same quality as the originals.

In the case of the packing box assemblies, Westinghouse inadvertently ordered the items from the manufacturer (Fisher Control) at a lower quality level than the original parts. As a result, the manufacturer fabricated the packing box assemblies at a non-nuclear facility using available stock, Type 316 stainless steel bar material.

The two packing box assemblies and manufacturer's Certification of Conformance were received at Wolf Creek on August 19, 1987. Using the Westinghouse Purchase Order (No. MN76817D) to Fisher Control as criteria, Quality Control inspectors verified that the proper documentation had been received. However, because the Westinghouse P.O. was in error, the inspectors

verified only the receipt of the Certificate of Conformance, rather than the more comprehensive ASME Code data package which would have included an N-2 code data form, material test reports and non-destructive examination reports.

One of the two packing box assemblies was installed on the subject valve on October 9, 1987. This installation was controlled in accordance with the WCNOC ASME Section XI Repair and Replacement program. This included in process QC inspections and a VT-2 visual examination for leakage at nominal system pressure. These inspections were completed on December 28, 1987, with no adverse results.

During a completion review it was discovered that the required vendor documentation was not in the package for the new packing box. Upon investigation by station personnel, it was discovered on March 17, 1988, that the ASME Code required documentation could not be provided and that only the Fisher Controls "Manufacturers Certification" was available. The ANII determined that this package was deficient because it did not contain the ASME Code required documentation, specifically the Code Data Report, Material Test Reports and NDE reports. This nonconformance has been documented on Corrective Work Request No. 1285-88

REQUESTED RELIEF

Article IWA-7210 of ASME Section XI states, in part, "Replacements shall meet the requirements of the edition of the Construction Code to which the original component or part was constructed ... ". Valve BB PCV 455B is an ASME Section III Class 1 valve manufactured to the 1974 Edition through Summer 1975 Addenda (Subsection NB). In accordance with Fisher Controls drawing Number 54A0202 Rev. G, the packing box is required to be ASME Section III SA-182-F316 material. This is a forged product form with chemical and physical requirements as specified in SA-182-F316 and additional examination requirements as specified in Section III paragraphs NB-2510(b) and NB-2541. Certified Material Test Reports, in accordance with paragraph NB-2130, and a Code Data Report (Form N-2) for welding performed on the part, in accordance with NA-3130(f), are required to be supplied.

Contrary to these requirements, WCNOC was provided two packing box assemblies as listed below:

a. Packing box material has been determined to be ASTM A-276 type 316 SS bar instead of ASME SA-182-F316 forging.

b. There was no NDE performed on the completed packing box assembly either volumetric or surface examination. The equipment specification for the Class 1 component required both types of inspection; the ASME Code allows an MT or PT in lieu of volumetric for forged valves of up to four inches.

   There was no surface non-destructive examination performed on the pipe nipple to packing box weld.

c. Pipe nipple material utilized for the leakoff connection is not known to be in compliance with the material requirements (i.e., SA 312 TP 316).

d. ASME Code N-2 Data Report for the packing box assembly was not supplied.

It should be noted that the code compliance issue is restricted only to the replacement packing box assembly. All other aspects of the valve are in compliance with applicable ASME Code requirements.

Because the packing box on this valve does not meet the ASME Section XI Code requirements as mentioned above, WCNOC is hereby requesting relief from the code pursuant to 10 CFR 50.55a(g)(6)(i) which states, "The Commission will evaluate determinations under paragraph (g)(5) of this section that code requirements are impractical.". Paragraph (g)(5)(iii) of 10 CFR 50.55a states, "If the licensee has determined that conformance with certain code requirements is impractical for its facility, the licensee shall notify the Commission and submit ... information to support the determinations.". On March 18, 1988, WCNOC notified the Commission, via telephone, of this situation due to the inability to replace the affected part during power operation. Information to support this determination is provided below.

BASIS FOR RELIEF

Quality Analysis

Because packing box assemblies were intended to be ASME Code pressure boundary parts, they should have been classified and procured by Westinghouse as "QA Code A." Instead, Westinghouse incorrectly designated the procurement as "QA Code D." This ultimately resulted in the incorrect parts being provided to WCNOC.

WCNOC has dispatched two Supplier Quality Representatives to Westinghouse to investigate this programmatic issue and ensure that adequate corrective actions are taken. This will include determination of extent of the problem, root cause analysis and appropriate remedial and preventative actions.

On March 19, 1988, a WCNOC Supplier Quality Representative was dispatched to the Fisher Valve, Mansfield, Ohio facility. This is the non-nuclear facility that fabricated the replacement packing box assemblies. On-site investigation there has determined that both packing box assemblies received at Wolf Creek were fabricated from the same heat of Type 316 SS annealed material manufactured by Carpenter Technology Corporation. Discussions with Carpenter Technology on March 21, 1988, provided three additional pieces of information: 1) although not stated on the Carpenter CMTR, the material meets ASTM A-276, 2) the 8 1/2" diameter round bar was made from a billet which was rotary forged and then rough turned and, 3) non-nuclear work is processed through the Carpenter facility in the same way as nuclear work, except for the retention of documentation. Further investigation at Fisher's Mansfield facility revealed the following:

a. The replacement packing box assemblies were machined using the same drawing used by Fisher's Marshalltown, Iowa, nuclear facility (original manufacturer of the nuclear valve), thereby providing interchangeability of the parts.

b. Welding of the pipe nipple to the packing box was performed using welding procedures qualified by Fisher Marshalltown in accordance with ASME Section IX.

c. Welding was performed by a welder who was originally qualified to ASME Section IX, but for whom required documentation has not been kept to show current compliance with Section IX. Subsequent review of the welder's qualification test report also shows he was not specifically qualified for the diameter of pipe weld required for this work since he qualified on a larger diameter pipe.

On March 18, 1988, the second packing box assembly, which was in the WCNOC warehouse, was sent to METLAB Testing Laboratories in Tulsa, Oklahoma. METLAB is qualified by WCNOC to perform testing and analysis services. This packing box assembly was destructively tested by METLAB to determine conformance with ASME SA-182-F316. Chemical analysis, mechanical testing, and metalurgical analysis were performed on a section removed from the packing box assembly. Since the material was thought to be bar stock instead of a forging, mechanical tests were performed parallel and transverse to the longitudinal axis of the original bar stock. Results of this testing and analysis are presented in the engineering evaluation section.

METLAB also performed chemical analysis of the pipe nipple welded to the packing box. The chemical analysis was in conformance with the ASME SA-312-TP316 required by the valve design drawing. Mechanical tests could not be performed due to the small diameter and thickness of the pipe.

On March 21, 1988, the installed valve was closed and allowed to cool down. An ASME Section XI VT-1 visual examination was completed on March 22, 1988, and found acceptable and is being submitted as an alternative to the Section III required surface examination of the packing box and pipe nipple to packing box weld. In addition, an ultrasonic examination of the flange connecting the packing box to the valve body will be performed and will provide an additional alternative to the required surface examination.

Engineering Evaluation

The packing box contains the packing seal between the shaft and packing box. The connecting weld between the leakoff nipple and the packing box is required to be of Class 1 construction because of its connection to a Class 1 component. However, it is not a part of the pressure boundary of the valve. Therefore, the lack of Code documentation and specific welder qualification is not a technical issue.

As noted in the Quality Analysis section, testing of the second packing box was performed at METLAB. Chemical analysis and mechanical tests performed parallel and transverse to the longitudinal axis of the original bar stock indicate that the actual material properties of the two replacement packing boxes, including the one currently in service, exceed the minimum specified mechanical properties and meets the specified chemical properties of

SA-182 F316, with the exception of the transverse elongation (44 percent actual versus 45 percent minimum specified) which is deemed to be inconsequential. In addition, Charpy V-notch tests indicate a fracture toughness similar to that of the SA-182 material.

The metalurgical analysis of photomicrographs of the material indicated that it was difficult to determine if the material was a forging or a bar. Although there was some indication of directionality, distinct grain orientation expected from a bar stock was not observed. (Distinct grain orientation parallel to the longitudinal axis of the bar may result in reduced strength of the material in a direction transverse to the longitudinal axis of the bar.) The metalurgical analysis therefore provided consistent results with the mechanical tests which were performed in the transverse direction.

As a result of the testing performed on this second packing box assembly, which was from the same heat and same piece of material as the packing box assembly installed in the plant, it is therefore expected that the installed packing box would be capable of functioning in a similar fashion to a part fabricated from a forged SA-182 F316 material.

The requirements for the installed part are similar to the requirements for the ASME Coded part. The parts are interchangeable with respect to fit and function as presented in the Quality Analysis.

The packing box was designed to ANSI B16.34 for the pressure/temperature conditions applicable to this component. These are the same pressure/temperature requirements for Class 1 valves of four inches and below, as NB-3513 references B16.34 for pressure/temperature ratings.

Since the packing box was originally designed using the rules of ANSI B16.34, there was no specific stress analysis for the packing box. A conservative analysis to determine the stress levels present in the packing box has been performed in accordance with ASME III Subsection NB. The analytical techniques employed classical formulas of stress analysis. The stresses due to internal pressure were determined using formulas for a thickwalled cylinder subjected to internal pressure. Beam bending and torsion theory were used to calculate the stresses due to seismic and deadweight loads. The stresses resulting from the bolt preload were determined using formulas for bending of flat plates. The maximum primary plus secondary stress intensity at the hub to flange juncture due to these loads was determined to be acceptable to the allowable values for SA 182 F316 material.

In addition to the stress analyses, a fracture mechanics evaluation has been performed, using a conservatively selected fracture criterion. Appropriate loading conditions, such as normal operating and seismic, have been considered. Two aspects of the valve packing box were evaluated.

1. For an estimated surface bending stress of 49,000 psi, an assumed flaw 0.3 inches deep at the flange to packing box body juncture and completely around the flange was shown to be stable.

2. Circumferential through wall flaws were assumed to exist in the cylindrical portion of the valve packing box. Normal operating loads

were not sufficiently large to produce a calculable leak rate. Also, thin wall flaws up to 4 inches long were shown to be stable under combined pressure and seismic loadings.

The valve packing box has thus been shown to be stable under severely assumed flawed conditions. Significant margins against conservative criteria have been demonstrated. The packing box should satisfy its service requirements without incident.

In addition to the above, an analyses on the packing box based on the procedures of ASME Section III Subsection NB and Appendix XI was performed. The resulting stresses show that the Code stress requirements have been met. In addition, Fisher Controls has also done a finite element analysis of the packing box which confirms that the stresses are acceptable.

Operations Evaluation

Continued operation of WCGS with the incorrect packing box assembly for Valve BB PCV 455B poses no inordinate hazards to plant operation. Valve BB PCV 455C is still available for normal pressurizer spray flow as needed and to aid in the event of a steam generator tube rupture.

Were the part to fail in such a manner as to induce a Reactor Coolant System leak, the symptoms would be that of a vapor space break, i.e., the RCS pressure would lower, the pressurizer level would increase, and the levels on the particulate, gaseous, and iodine monitors in the containment atmosphere would increase. In an earlier pressurizer spray bypass valve packing leak, the containment monitors were a very good indicator of the problem. A lowering RCS pressure would ultimately trip the plant at pressurizer pressure equal to 1875 psig with a further safety injection (SI) and reactor trip at 1830 psig. The operator would proceed to Plant Emergency Procedure E-02 for either cases and would follow the procedure to "reactor trip response" if pressure stabilized, or would go on to the proper Loss of Coolant procedure E-1 as a result of an SI. If the pressure did not stabilize, the SI could not be terminated, as one of the criterion for reset is RCS pressure stable or increasing. Along with the criteria is the need for subcooling to be maintained. Either one or both of these would override the increasing pressurizer level because all criteria must be met to terminate an SI, not just inventory.

If the valve itself failed open, the reactor would trip on low pressure or at the safety injection setpoint. In any case, the reactor would be tripped so the affected RCP (PBB01A) could be turned off. At powers less than 48 percent, the affected Reactor Coolant Pump would be tripped with orderly shutdown to HOT STANDBY as required by Technical Specification 3.4.1.

A preliminary safety assessment of the effect of a catastrophic failure of the pressurizer spray valve packing box was performed. The complete failure of the packing box represents a small break LOCA with fluid discharge reaching the break from the pressurizer and the cold leg. Although this scenario is not analyzed in the WCNOC USAR, analyses using the NOTRUMP small break LOCA ECCS Evaluation Model indicates that this event is bounded by a small break LOCA in the cold leg. Westinghouse and the Westinghouse Owner's Groups examined a scenario similar to this using the current Westinghouse small break LOCA ECCS Evaluation Model incorporating the NOTRUMP analysis methodology in

WCAP-10054. Analyses of a plant similar in design to Wolf Creek and using the NOTRUMP small break LOCA ECCS Evaluation Model show that the reactor core is not expected to be uncovered in the unlikely event that there is a catastrophic failure of the pressurizer spray flow control valve packing box.

For non-LOCA events, normal reactor control system and engineered safety systems are not required to function in the ultimate mitigation of the consequences from accidents. However, in the analysis of the Chapter 15 events, control system action is considered if that action results in more severe consequences. For some events, the analysis is performed both with and without control system operation to determine the worst case.

The Turbine Trip event is analyzed with and without pressurizer spray actuation. The consequences are acceptable without pressurizer spray. All other events are not impacted by not having pressurizer spray available.

CONCLUSION

Based upon the above information, WCNOC considers the use of the packing box currently installed on the Pressurizer Spray Valve BB PCV 455B to be acceptable and continued operation is justified. In order to be assured that the valve does not degrade, a periodic visual inspection will be performed until the packing box assembly is replaced. The replacement of the packing box will be performed during the next outage of sufficient duration and appropriate plant conditions. In any case, this assembly will be replaced no later than the next refueling outage.