WOLF CREEK
NUCLEAR OPERATING CORPORATION

Richard A. Muench
Vice President Technical Services

SEP 13 2001

ET 01-0026

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

Reference: 1. Letter WO 00-0036, dated September 15, 2000, from B.T. McKinney, WCNOC, to USNRC

Subject: Docket No. 50-482: Response to Request for Additional Information Regarding the Application to Amend Technical Specification Table 1.1-1

Gentlemen:

Reference 1 submitted an application to amend Technical Specification Table 1.1-1, "MODES," for the Wolf Creek Generating Station (WCGS), Unit No. 1, Facility Operating License No. NPF-42, in accordance with 10 CFR 50.90. In a telephone conference on July 17, 2001, between Wolf Creek Nuclear Operating Corporation (WCNOC) and Mr. Jack Donohew, NRC Project Manager, WCNOC agreed to provide a response to questions provided by Mr. Donohew in electronic mail dated June 8, 2001. The attached information was discussed with the NRC on August 16 and 24, 2001, and WCNOC agreed to provide additional changes to the technical specifications. The information requested supports the proposed changes to the technical specifications to allow operation of WCGS with one reactor pressure vessel head closure bolt not fully tensioned. Attachment 1 to this letter provides the requested information and additional changes to the technical specifications. The proposed changes to Technical Specification Table 1.1-1 and Administrative Controls Section 5.5 provided in Attachment II, supercede the changes proposed in Attachment II of Reference 1.

There are no licensing commitments contained in this submittal. The supplemental information provided in this submittal does not impact the conclusions of the No Significant Hazards Consideration provided in Reference 1.

In accordance with 10 CFR 50.91, a copy of this correspondence is being provided to the appropriate Kansas State Official.

A001

If you have any questions concerning this matter, please contact me at (620) 364-4034, or Mr. Tony Harris at (620) 364-4038.

Very truly yours,

RAMuench

Richard A. Muench

RAM/rlr

Attachments: I - Response to Request for Additional Information
II - Proposed Technical Specification Changes
III - FSAR Table 5.3-6

cc: V. L. Cooper (KDHE), w/a, w/e
J. N. Donohew (NRC), w/a, w/e
W. D. Johnson (NRC), w/a, w/e
E. W. Merschoff (NRC), w/a, w/e
Senior Resident Inspector (NRC), w/a, w/e

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Richard A. Muench, of lawful age, being first duly sworn upon oath says that he is Vice President Technical Services of Wolf Creek Nuclear Operating Corporation; that he has read the foregoing document and knows the contents thereof; that he has executed the same for and on behalf of said Corporation with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By RAMuench

Richard A. Muench
Vice President Technical Services

SUBSCRIBED and sworn to before me this 13th day of Sept. , 2001.

NOTARY PUBLIC STATE OF KANSAS CINDY NOVINGER My Appt. Exp. 7/8/02

Cindy Novinger
Notary Public

Expiration Date 7/8/02

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

In a telephone conference on July 17, 2001, between Wolf Creek Nuclear Operating Corporation (WCNOC) and Mr. Jack Donohew, WCNOC agreed to provide a response to questions provided by Mr. Donohew by electronic mail on June 8, 2001. The information requested supports the proposed changes to the Technical Specifications to allow operation of WCGS with one reactor pressure vessel head closure bolt not fully tensioned. For the purposes of this response, reactor vessel head closure bolts as specified in the Technical Specifications, are equivalent to reactor vessel closure studs or reactor pressure vessel head studs. The information concerning the operating experience of reactor pressure vessel (RPV) head studs in the industry is limited to nuclear power plants in the United States.

QUESTION 1:

Discuss bolt degradation mechanisms. For instance, are any of the bolts overly hard (yield strength over 150 ksi, hardness higher than Rockwell C32) and, therefore, subject to hydrogen embrittlement from moisture in the air?

RESPONSE:

There have been no reported failures of RPV head studs at nuclear power plants. Industry experience with failures of primary coolant pressure boundary bolting has been primarily with fasteners on steam generator manways and reactor coolant pump (RCP) main flange bolts[1]. The degradation mechanisms that have been experienced by those fasteners are boric acid corrosion and stress corrosion cracking. Those fasteners are in locations where they are more likely than RPV head studs to be wetted by leakage during operation or become wet from leakage or other moisture sources during periods of layup. Because of the uniqueness of the RPV head seal design and the way head studs are handled during refueling outages, the failure experience with other primary system pressure boundary fasteners is not considered to be relevant.

Vessel head studs have experienced minor boric acid corrosion (pitting or wastage a few mils deep) from leakage through conoseal joints or vent connections on the reactor vessel head. More significant boric acid corrosion of vessel head studs was experienced at Turkey Point, Unit 4, in 1987 where the top end of several studs and the nuts experienced boric acid corrosion as the result of leakage from a conoseal joint on the reactor head.[2,3] This case of boric acid corrosion was not the result of leakage through the RPV head seals. Significant boric acid wastage was observed on the vessel head fasteners in the area of the leakage. The probability of such an event is not affected by operation of the RPV with one stud untensioned. In 1988, several leaks through the reactor vessel head O-ring seals at Millstone 2, with leak rates up to half the Technical Specification limit, caused minor corrosion of nine studs. Most (7 or 8 of the 9 studs that experienced minor corrosion) of these affected studs were suitable for reuse.[11]

Only one case of cracking of a RPV head stud was found in the technical literature. This was a stud from Dresden, Unit 2, a boiling water reactor (BWR). The cracked stud is discussed in an Electric Power Research Institute (EPRI) report on ultrasonic examination of studs and bolts.[4] A crack found by ultrasonic testing (UT), and confirmed by metallography, was reported as being

0.7 in. deep. A crack in another stud was sized by UT as 2.09 in. deep, but this was not confirmed by metallography. The cracks initiated at pits at the thread roots and were at angles of 60° to 70° to the stud axis. The studs were extensively oxidized and pitted. The studs are reported to have been in service for 19 years. The report contains no discussion of the service conditions that led to the cracking. The report also describes crack-like UT indications found on studs from Cooper Nuclear Station, a BWR. However, metallographic analysis showed that the UT signals were produced by corrosion degradation of the threads and not by cracks. Given the lack of detail available regarding the causes of cracking of the Dresden studs, it is difficult to evaluate whether the Dresden experience is relevant to the Wolf Creek Generating Station (WCGS). It should be noted that there are significant differences between the RPV head closures at BWR and pressurized water reactor (PWR) plants. BWR operating pressures are lower so the studs tend to be smaller in diameter than those used for PWR vessels. It is also common practice to leave most studs for BWR vessels in place during refueling operations. This exposes the studs to the high purity reactor coolant at warm temperatures for the duration of the refueling. The reported angle of the cracks to the stud axis is not what would be expected if the cracks had initiated and grown while the studs were tensioned.

There has been no reported cracking degradation of RPV head studs at PWR plants.

Based on the certified mill test reports for the WCGS RPV head stud materials, none of the studs have a yield strength greater than 150 ksi.[5] The highest reported yield strength (Bar No. 133-1) is 149.2 ksi. The average yield strength is 146.5 ksi. The reported hardnesses of the studs are less than or equal to 341 Brinell. This corresponds to Rockwell C36.6 on standard hardness conversion charts. Although this is greater than Rockwell C32, it is considered that the combination of yield strength and hardness makes the studs, at worst, mildly susceptibleto hydrogen embrittlement at low temperature in a corrosive environment. Given that the studs are in a hot dry environment when the plant is operating and are removed from the vessel flange for storage in a clean dry environment during refueling outages, they are not exposed to a low temperature corrosive environment. Thus, stress corrosion cracking degradation is not a plausible degradation mechanism for the WCGS RPV head studs.

<u>QUESTION 2:</u>

Could any degradation mechanism fail a stud next to or near the untensioned stud and lead to multiple failures of the studs? Discuss how many studs will have to fail before a leakage occurs and how many studs will have to fail to result in a off-design condition that would not meet ASME Code allowable stress criteria?

<u>RESPONSE:</u>

Degradation of RPV head studs adjacent to or near an untensioned stud would only be expected if significant leakage through the head seals were to occur at the location. Leakage is not expected from operation of the RPV with one head stud untensioned. The analysis performed[6] to verify that ASME Boiler and Pressure Vessel Code (ASME Code) stress criteria are satisfied for operation with one stud untensioned also shows that the additional flange separation at the location of the untensioned stud is less than 0.0005 in. (0.5 mil). The minimum elastic spring back of the O-ring is specified in the RPV design stress report[7] to be 0.013 in. and the report notes that the maximum gasket opening is 0.005 in. (This occurs briefly during the heatup transient. At steady-state conditions, all of the 0.013 inch springback is available to accommodate pressure and thermal transients). Thus, the additional flange separation at the location of an untensioned stud is small in comparison to the maximum gasket opening. The maximum calculated flange separation at the O-rings in the vicinity of an untensioned stud is much less than the opening displacement (0.013 in.) that the O-ring is designed to seal. Because there is a small increase in calculated flange separation with one stud untensioned, there may be a very small theoretical increase in the probability of leakage past the O-ring relative to the ideal case with all 54 studs uniformly tensioned. However, because the separation with one stud untensioned meets the maximum separation criterion for the O-ring seal, the probability of seal leakage relative to the design assumptions is unchanged. Industry experience shows that most leakage of RPV O-ring seals is the result of foreign matter under the seal or small defects (e.g., scratches or wrinkles) on the O-ring or flanges. These dominant contributors to the probability of leakage are unaffected by having one stud untensioned. Based on this analysis, it is concluded that the probability of leakage through the vessel flange seals is not increased significantly by operating with one stud untensioned. As discussed in the response to Question 6, small-undetected leakage through the O-rings would not be damaging and procedures require investigation and corrective action for detectable leakage. Thus, there is no increased probability of degradation of studs adjacent to one that is untensioned.

In order to satisfy the total bolting area requirements of the ASME Code for the RPV flange, 50 of the 54 studs must be installed. If the missing studs are all adjacent to each other, at least 52 studs are required. Based on the analyses that have been performed for WCGS[6] and Callaway Plant[9], it is estimated that it would require more than two adjacent studs untensioned to increase the probability of leakage (i.e., to increase the flange separation at the O-rings to more than 0.013 in.). Four untensioned studs (the maximum number that would still meet ASME Code stress requirements) distributed around the flange would have no effect on the probability of leakage. This is based on the fact that the region of increased flange separation is limited to the missing stud and the adjacent studs on either side. By superposition, two untensioned studs with two or more properly tensioned studs between them do not interact in terms of flange separation.

QUESTION 3:

Discuss degradation mechanisms that could lead to plant operation with a bolt not fully tensioned. Because it is stated in the application that operation with one bolt not fully tensioned is "not standard operating practice" and because such operation would be in a degraded condition (i.e., the plant be in an off-design condition), discuss what efforts would be done to prevent operation with one bolt not fully tensioned.

RESPONSE:

The anticipated scenario that could lead to the need to operate with one RPV head stud untensioned is damage to the threads of one of the studs or the mating threads in the vessel flange such that the stud becomes stuck in the flange with the threads not fully engaged so that it cannot be fully tensioned. This scenario has occurred at several plants with RPV's similar to WCGS's, including D. C. Cook, (1 in 1986), Catawba (1 in 1989), Callaway (5 in 1987), Commanche Peak (1 in 1992, 3 in 1994), Sequoyah (1 in 1996), Seabrook (1 in 1997), and Braidwood (1 in 1992). WCGS has not experienced any stuck studs, but has had one stud that was difficult to turn in and out of the flange threaded hole. During Refueling Outage VIII (1996), WCNOC experienced difficulties installing one stud, and during Refueling Outage IX the flange stud hole was cleaned and inspected, and minor repairs made to the vessel stud hole threads. At present, the RPV studs can be inserted to their design thread engagement and removed freely. A stuck stud that cannot be fully tensioned is not an expected occurrence at WCGS, but is considered to be a possibility.

When a stuck stud situation occurs, the stud is found to be stuck either when removal is attempted at the beginning of a refueling outage or when the studs are installed at the end of a refueling outage. If a stud becomes stuck in a partially inserted condition with less than required thread engagement, it cannot be tensioned. Additionally, the vessel flange threaded holes could be damaged such that the stud may not be capable of being inserted into the threaded hole. Typically, when a stud becomes stuck, the threads are galling and additional attempts to move the stud can result in damage to the vessel flange threaded holes. The corrective action for a stuck stud is to cut it off and bore out the portion in the vessel flange. This requires special equipment from a vendor. Mobilization for this activity once a refueling outage is in progress is usually impractical without a significant extension of the outage. WCNOC engineering, outage management, and scheduling practices support leaving the untensioned stud for one operating cycle and performing a Code repair as a planned activity during the next scheduled refueling outage.

QUESTION 4:

Describe the inspections, and their frequency, of the bolts and which are required by regulations. Discuss what would be the maximum non-detectable flaw for the bolts for these inspections?

RESPONSE

RPV head studs are removed from the vessel flange at the beginning of each refueling outage and stored in a dry clean environment. During the refueling outage, each stud is visually inspected (VT1). This inspection, which is beyond what is required by ASME Code Section XI, is to detect evidence of corrosion or mechanical damage. The visual inspections are sufficient to detect any general corrosion or mechanical damage to the studs. In accordance with ASME Code, Section XI, all studs are inspected by volumetric UT and MT surface examination. Typically, one third of the studs are inspected each two, eighteen month operating cycles. The surface and volumetric examinations will detect any cracking degradation. The UT inspections are done using longitudinal UT calibrated with a 0.157 inch-deep notch with a reflective area of 0.059 square inches calibrations standard (in accordance with ASME Code, Section XI, Appendix VIII). It is not expected that new subsurface flaws will occur in service, so the UT primarily serves to detect cracks growing toward the center of the stud from the thread roots. The MT surface examination will identify any surface connected linear indications greater than

1/16 in. long. The technique is sensitive to very shallow (a few mils depth) cracks. Any systematic degradation mechanism that affects most of the studs would be detected by the sampling surface inspection. An unusual surface flaw could be present on some of the uninspected studs. However, no flaws were identified by pre-service or first 10-year inspection interval inspections. If a previously unidentified degradation mechanism begins to affect the studs, it should affect most of the studs because they are all of the same material and all operate in the same environment. The sampling inspections are expected to identify any degradation mechanism, should it occur.

QUESTION 5:

If one stud is stuck in the vessel flange and cannot be fully tensioned, how will that affect the torquing sequence of the remaining bolts? How will you ensure that the torquing sequence will not increase the probability of developing leakage? Discuss the effect this leakage would have on plant operation.

RESPONSE:

If the RPV head is to be installed leaving one stud untensioned, the tensioning pattern will be adjusted such that the sealing O-rings are fully compressed before the stud not being tensioned is encountered in the pattern. At WCGS, the RPV studs are tensioned using hydraulic tensioners. Typically, three tensioners are used in a symmetric pattern. The first two tensioning sets tension six studs symmetrically arranged about the vessel. The load applied by these six studs is sufficient to fully compress the sealing O-rings and bring the flanges together in metal-to-metal contact. If one stud is left out of one of the tensioning sets after the first two sets, there is no significant effect on the sealing system because there is no additional compression of the O-rings. Tensioning the additional studs only increases the contact pressure between the vessel and head flanges. The final tensioned condition of the studs is verified by an elongation measurement when all studs have been tensioned. Elongations must be within an acceptance criteria. The acceptance criteria for the final stud elongations was established to ensure that all ASME Code stress limits are met for all specified service loads for the worst case flange and stud bending that results from various tensioning patterns, including one stud untensioned. Therefore, leaving one stud untensioned has no significant impact on the head tensioning procedure.

QUESTION 6:

Discuss any evidence of cracking of any of the studs at Wolf Creek. Discuss the industry experience for such cracking and how it relates to Wolf Creek plant-specific operating history.

RESPONSE:

There has been no experience with cracking of RPV head studs at WCGS or at other PWR plants. Industry experience with failures of primary pressure boundary studs and bolts has been primarily with steam generator primary manways and RCP main flanges. Fasteners used in these locations can be wetted by leakage from the primary system. They may also become moist during periods of layup at ambient temperature because they are not always removed during outages. In most cases where cracking was experienced with fasteners for steam generator manways and RCPs, the fasteners were contaminated with lubricants containing sulfur or were wetted by fluid contaminated with impurities leached from insulation. Such conditions are not experienced by the RPV head studs at WCGS. Studs have always been removed and are expected to be removed from the vessel flange during refueling outages. They are stored in a clean dry environment so they are not susceptible to stress corrosion cracking. The lubricant used on the RPV stud threads is nuclear grade Neolube, which does not contain any compounds that are detrimental to the stud material. During power operation, the studs are in a high temperature dry environment that does not cause stress corrosion cracking. Substantial leakage through both the primary and secondary O-ring seals in the RPV flange would be required to create moist aggressive conditions at the location of the studs. Finite element stress analysis of the vessel head closure shows that operation with one stud untensioned results in a negligible increase in the flange separation at operating pressure compared to the normal condition with all studs tensioned, and the separation remains small compared to the maximum separation that the O-rings are designed to seal. Although leakage due to one stud not fully tensioned is not expected, leakage may occur. Therefore, there is no increased probability of leakage through the vessel head seals. Leakage sufficient to maintain a moist environment at a stud should be detected by flow through the leak-off port between the two O-rings, increased radiation levels in containment, or because operational leakage exceeds the leakage limits of Technical Specification 3.4.13, "RCS Operational Leakage." Undetected O-ring leakage of this magnitude has not been experienced at WCGS or elsewhere in the industry and is not expected to occur. Temperature indication is provided for monitoring reactor vessel flange leakoff. Plant procedures provide the actions to be taken for a high reactor vessel flange leakoff temperature, which includes verifying leakage past the RPV O-rings and that operational leakage is in accordance with Technical Specification 3.4.13. If leakage through the outer O-ring is detected, plant procedures require a plant shut down.

QUESTION 7:

The finite element analyses performed by the Dominion Engineering Inc. assumed the remaining 53 studs are in sound condition with no degradation. Analyses performed for similar conditions typically assume that studs are degraded to the minimum detectable non-destructive examination (NDE) limit which is the minimum detectable flaw using ultrasonic techniques (UT). Experience has shown that a minimum detectable flaw is about 0.1 inches. Assuming an 0.1 inch crack extending 360 degree around the remaining 53 studs, how will that affect the finite element analyses results? Specifically, discuss if the results would still meet the ASME Code allowable stress criteria?

RESPONSE:

Design analyses per the ASME Code do not typically assume degradation of the pressure boundary components. If a degradation mechanism is known to exist, condition assessment evaluations may be performed which assume the presence of a flaw at the detection limit for the NDE technique which is being used to monitor the degradation. As discussed in the response to

Question 1, no mechanism has been identified by which studs are degraded as a result of normal plant operation.

The only concern is that problems with threads may occur that prevent a stud from being properly installed so that it can be fully tensioned (i.e., it becomes stuck in a position where it does not have full thread engagement). WCGS has not experienced any stuck studs to date. However, there have been cases when some studs could not be easily turned into or out of the flange. When this has occurred, a thorough visual and surface examination of the affected stud threads has been performed to identify any non-conformances. Minor damage to the stud threads has been repaired, and there has been no evidence of cracking in the affected studs. The stud holes in the vessel flange where studs could not be easily turned in and out have been inspected and damaged threads have been repaired. Inspections and evaluations of the vessel flange stud holes have been performed to verify that there is adequate thread engagement to meet all ASME Code allowable stress criteria. At present all studs are installed and removed normally without problems. Therefore, there is no ongoing degradation from stud installation and removal.

Because there is no degradation mechanism identified that can affect the remaining studs during an operating period when one stud is left untensioned, there is no need to assume the presence of a flaw in the remaining studs in the stress analysis for this case. Furthermore, the studs are inspected by MT surface examination as well as UT. The MT examination will identify very small surface connected flaws (greater than 1/16 in. long by a few mils depth). Flaws at the MT detection limit have no structural significance. No flaws have been identified in the WCGS studs by pre-service or first 10-year inspection interval inspections. Inspection of all studs by MT during each 10-year inspection interval provides assurance that any structurally significant flaws in the studs will be identified. The finite element analysis performed by Dominion Engineering, Inc.[6], shows that all ASME Code requirements are satisfied with 53 of 54 studs tensioned. The Code requires that there be sufficient total bolting area to withstand the operating pressure with an average primary membrane stress less than S_m. At 650°F, S_m = 34.8 ksi for the studs. With 53 studs installed, the calculated average primary membrane stress is 32.8 ksi. Other Code limits on local membrane and membrane plus bending stress for all specified loadings that the closure may experience are also satisfied. Code allowable stresses are based on a factor of safety of 3.0 for primary membrane stress.

QUESTION 8:

Discuss the minimum number of failed bolts, and in what pattern, that could cause the RPV head to fail, and the uncertainty in the number. Discuss the expected frequency of that number of bolt failures, and the basis for the estimate. What is the effect of the proposed amendment on this number of failed bolts and the frequency of that number failing.

RESPONSE:

As discussed in the response to Question 2, 50 of 54 studs are required to meet ASME Code average primary membrane stress requirements and if the missing studs are all adjacent to each other, at least 52 studs are required. However, this is only a Code requirement and does not represent a condition where failure of the vessel head closure could occur. As discussed in the response to Question 7, there are large margins implicit in the ASME Code stress limits. Bolted closures, because of their high degree of redundancy, rarely experience catastrophic failure. The usual failure mode is leakage, which would only be expected if several adjacent studs were severely degraded.

There is no expectation of failures of RPV head studs during an operating period once they have been properly tensioned. There is no known service induced cracking degradation mechanism for studs. When a stud has been installed in the vessel flange such that it has design thread engagement, the stud has been tensioned, and the tensioning has been verified by an elongation measurement, the stud will be structurally reliable until it is detensioned at the end of the operating cycle. Because the stresses in the studs result primarily from preloading the flange joint using measured stud elongation criteria, the stresses in the remaining studs are affected only slightly by leaving one stud untensioned. The stresses in the remaining studs meet all ASME Code criteria. Code limits are established to provide a high degree of structural reliability. No degradation of load carrying capacity during an operating cycle for a tensioned stud is expected, and none has been observed at WCGS or elsewhere in the industry.

The requested Technical Specification change will allow operation with one stud untensioned or missing. This condition has been analyzed and shown to be acceptable in accordance with the ASME Code. Operation with more than one stud untensioned is not anticipated and would require additional analysis, and NRC approval. Furthermore, the proposed change would not permit the plant to exit MODE 6 and enter MODE 5 without 53 studs fully tensioned.

QUESTION 9:

In considering the risk of operating indefinitely with one bolt not fully tensioned, as proposed in your application, discuss how you meet the guidance in Regulatory Guide 1.174 including the following: (a) compare the changes in core damage frequency and LERF (including the impact on ATWS sequences) to the Regulatory Guide 1.174 guidance, and (b) discuss your proposed monitoring program. Also discuss why the reliability, redundancy, diversity, and defense-in-depth of the RPPV head leak detection system is adequate.

RESPONSE:

As discussed in the response to Question 7, operation of the RPV with one head stud untensioned has been analyzed using finite element stress analysis. The condition with one stud untensioned meets all requirements of the ASME Code for stresses. There is a negligible increase in flange separation in the vicinity of the untensioned stud at operating pressure and calculated flange separation at the seal remains much smaller than the minimum O-ring spring back. From these results, it is concluded that there is no increased probability of either stud failures or flange leakage compared to the design base case with all studs tensioned. Therefore, there is no increased risk.

QUESTION 10:

Discuss the effect of a failure of a bolt during power operation. How could this be detected? What is the history of such failures? Address defense-in-depth and changes to safety margins in the proposed operation of the plant with a failed bolt.

RESPONSE:

Degradation of a single stud during an operating period would probably have no effect on the plant operation and would not be detectable, as analysis and industry experience both demonstrate no increased probability of O-ring leakage with one stud untensioned. As was discussed in the response to Question 1, there has been no history in the nuclear industry of failure of RPV head studs during operation. Operation with one stud untensioned has been analyzed and this condition meets the requirements of the ASME Code, which is the RPV design basis. Thus, there are no changes to safety margins or defense in depth by operating with one stud untensioned.

There is no known degradation mechanism for WCGS RPV head studs that could result in a failure of a stud during an operating period. Studs are made from a ductile material with good fracture toughness. The ASME Code stress limits provide large factors of safety to ensure that catastrophic failures will not occur. If some unanticipated degradation of a stud occurred such that its cross sectional area was significantly reduced during an operating cycle, it would deform plastically so as to reduce the load it carries. The vessel flanges are very stiff (the stud spacing is much smaller than the 29 in. height of the vessel flange). Therefore, the flange cannot deflect significantly in the region of a degraded stud such as to maintain the load in the stud if it elongates plastically. Plastic deformation of a degraded stud would result in a small increase in the stress levels in adjacent studs as indicated by the finite element analysis for one stud untensioned. If a number of studs lost significant cross sectional area, relaxation of the stud loads by plastic deformation could result in leakage through the flange seals.

The redundancy of bolted flange closures is high such that they are well protected by leak before break. Analyses of other PWR primary system pressure boundary bolted closures[8] (e.g., steam generator manways, RCP closures, and check valves) show that very large leak rates will occur before catastrophic failure of a bolted joint. Although the RPV head closure is larger than these other components, the head flange has a higher degree of redundancy. Therefore, the conclusions from the analyses of other components should be applicable to, if not conservative for, the RPV.

Based on a detailed stress analysis of the RPV head closure, leakage or failure of the joint is not anticipated either with all studs tensioned or with one stud left untensioned.

QUESTION 11:

Discuss the alternative approach to the proposed changes to TS Table 1.1-1 of having a program on the bolts in Section 5.5 of the Administrative Controls. The program would provide the controls to allow power operation of the plant with one bolt not fully tensioned, including the following: stating whether plant operation would be limited to only one operating cycle without NRC staff approval, listing the inspections and frequency of the bolts, discussing criteria by which a failed bolt would not be repaired during a current/upcoming refueling outage and plant operation would be started/continued with one bolt not fully tensioned, listing the analyses upon which the plant operation would be based, and listing the commitments provided in the application.

RESPONSE:

As discussed with the NRC on August 16 and 24, 2001, WCNOC agreed to revise the proposed Technical Specifications provided in Reference 10 and include an Administrative Controls program for reactor vessel head closure bolt integrity for operation with one reactor vessel head closure bolt less than fully tensioned. Attachment II provides the proposed changes to the Technical Specifications.

The following aspects of a proposed Technical Specification Administrative Control program were discussed in a telephone conference on August 24, 2001.

- The development of written procedures that are implemented and maintained.

  Technical Specification 5.4.1a. requires written procedures shall be established, implemented, and maintained covering the following activities: a. The applicable procedures recommended in Regulatory Guide 1.3.3, Revision 2, Appendix A, February 1978. Appendix A, Item 2.k requires procedures for refueling and refueling equipment operation and Item 2.l requires procedures for refueling and core alterations. Procedures FHP 02-01, "Refueling Operations," FHP 02-009, "Reactor Vessel Stud Removal, Installation and Cleaning," and QCP 20-519, "Section XI Visual Examinations," are the primary procedures associated with vessel stud activities and visual examinations.

  As such, WCNOC believes that it is unnecessary for this Technical Specification program description to require procedures.

- Periodic surveillance of the bolts conducted that is consistent with the regulations and the ASME Code, and includes visual examinations to detect general corrosion and mechanical damage to the studs.

  10 CFR 50.55a(g) provides the regulations associated with inservice inspection requirements with the exception that visual examinations (VT-1) are not required for the vessel head studs. In accordance with the ASME Code, Section XI, all bolts are inspected by volumetric UT and MT surface examination during each 10-year inservice inspection interval. The proposed Technical Specification 5.5.17b. (Attachment II) indicates that visual examinations are performed on the closure bolts.

As such, WCNOC believes that the existing regulation and proposed wording for Technical Specification 5.5.17 is sufficient.

- The reactor vessel will not be subjected to hydrostatic test conditions before the closure bolt is returned to service; and the heatup will be maintained ≤ 50 °F in any 1-hour period until the closure bolt is returned to service.

Reference 10 committed to the following precautions before operating with an untensioned or failed closure stud:

1. the particular circumstances will be reviewed to determine that the analysis is still applicable,

2. the reactor vessel will not be subjected to hydrostatic test conditions before the closure stud is returned to service, and

3. the heatup rate will be held to 50 °F per hour (half of the typical 100 °F per hour design heatup rate) until the closure stud is returned to service.

The precautions given in items (2) and (3) from the analysis[6] are intended only to be recommended good practices that provide added assurance that O-ring leakage does not occur. The analyses described in Appendix F[6] demonstrate that the ASME Code stress limits are satisfied for all design conditions, including design heatup rate. The calculated maximum flange separation during heatup at the design rate is 0.0055 inches, which is much less than the minimum O-ring springback of 0.013 inches. Thus, the conclusions reached in Appendix F do not rely upon either of these two precautions in (2) and (3) above being followed. Nonetheless, application of hydrostatic test pressure and heatup at rates approaching the analyzed 100 °F/hour are two practices that most challenge the integrity of the O-ring seal. The purpose of the recommendations was to propose methods of operation that provide additional assurance that the inner O-ring will seal when operating with an untensioned stud. These specific recommendations were determined by discussions with Callaway personnel in 1987 to be readily achievable without significantly impacting normal operation, and were proposed in the interest of treating the vessel as gently as practical when operating with one stud untensioned. While no formal calculations were prepared to support the 50 °F/hour recommendation, the amount of flange separation at the inner O-ring during the heatup transient may be assumed to be roughly proportional to the heatup ramp rate. Therefore, by limiting heatup to 50 °F/hour, the O-ring flange opening at the inner O-ring would be expected to be reduced by about 2 to 3 mils, which more than compensates for the calculated effect of the untensioned stud on flange opening.

As such, WCNOC believes that these 2 items do not meet the criterion of 10 CFR 50.36(c)(2)(ii) for inclusion into the Technical Specifications, but would implement the precautions/recommendation as commitments.

- With operation with one bolt not fully tensioned, a plan would be developed to ensure that the one bolt would be returned to service in the next scheduled refueling outage.

  10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," states, in part: "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected." As such, if a vessel head stud could not be fully tensioned and the decision was to operate for one cycle in this condition, the failure of the stud would be addressed by the WCNOC Corrective Action Program (AP 28A-001, "Performance Improvement Request" and AP 16C-007, "Work Order").

  As such, WCNOC believes that a requirement to develop a plan is not required based on the existing regulation.

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[1] IE Information Notice No. 80-27, “Degradation of Reactor Coolant Pump Studs,” June 11, 1980. IE Information Notice No. 82-06, “Failure of Steam Generator Primary Manway Closure Studs,” March 12, 1982. IE Bulletin No. 82-02, “Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants,” June 2, 1982. Information Notice No. 90-68, “Stress Corrosion Cracking of Reactor Coolant Pump Bolts,” October 30, 1990. Generic Letter 91-17, “Generic Safety Issue 29, Bolting Degradation or Failure in Nuclear Power Plants,” October 17, 1991.

[2] C. J. Czajkowski, “Survey of Boric Acid Corrosion of Carbon Steel Components in Nuclear Plants,” NUREG/CR-5576, Brookhaven National Laboratory, Upton, NY 11973, June 1990.

[3] A. S. O'Neill and J. F. Hall, “Survey of Boric Acid Corrosion of Carbon and Low-Alloy Steel Pressure Boundary Components in PWR Power Plants,” RP2006-18, Electric Power Research Institute, Palo Alto, CA 94303, April 1988.

[4] Steven Kenefick, “Technical Basis for Ultrasonic Examination of Studs and Bolts,” EPRI TR-104997, Electric Power Research Institute, Palo Alto, CA 94303, November 1995.

[5] FSAR Table 5.3-6 (Attachment III)

[6] D. J. Gross, E. S. Hunt, and J. B. Broussard, III, “Reactor Vessel Bolting Evaluations– Wolf Creek Generating Station,” R-4328-00-1, Rev. 0, Dominion Engineering, Inc., May 1997.

[7] “Analytical Report for Kansas Gas & Electric Company Wolf Creek Nuclear Power Plant, “ CENC-1313 (1978), with Addendum 1(CENC-1377, 1979), and Addendum 2 (CENC-1532, 1982), Combustion Engineering, Inc.

[8] “Degradation and Failure of Bolting in Nuclear Power Plants,” EPRI 5769, Vol. 1, Electric Power Research Institute, Palo Alto, CA 94303.

[9] E. S. Hunt, “Stuck Stud Evaluation – Callaway Unit 1 Reactor Vessel,” DEI-234, Rev. 0, Dominion Engineering, Inc., October 1987.

[10] Letter WO 00-0036, “Application For Amendment To Technical Specification Table 1.1-1, “MODES,”” from B. T. McKinney, WCNOC, to USNRC, September 15, 2000.

[11] “Boric Acid Corrosion Guidebook,” EPRI-TR-102748, Electric Power Research Institute, Palo Alto, CA 94303, April 1995.

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 350
4	Hot Shutdown[b]	< 0.99	NA	$350 > T_{avg} > 200$
5	Cold Shutdown[b]	< 0.99	NA	≤ 200
6	Refueling[c]	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

, except as specified in Specification 5.5.17, "Reactor Vessel Head Closure Bolt Integrity."

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program, dated September 1995."

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 48 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.20% of containment air weight per day.

d. Leakage rate acceptance criteria are:

   1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests;

   2. Air lock testing acceptance criteria are:

      a) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

      b) For each door, leakage rate is $\leq 0.005\ L_a$ when pressurized to $\geq$ 10 psig.

e. The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

f. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.5.17 INSERT A

INSERT A

5.5.17 Reactor Vessel Head Closure Bolt Integrity

This program provides the requirements to support normal plant operation with one reactor vessel head closure bolt less than fully tensioned for one operating cycle. The provisions of this program shall be implemented when a head closure bolt becomes stuck in a partially inserted position such that the amount of thread engagement is not sufficient to take the tensioning loads without damage to the vessel threads or a bolt is not capable of being inserted into the bolt hole.

Prior to operation with one reactor vessel head closure bolt less than fully tensioned, the following conditions shall apply:

a. The circumstances associated with the less than fully tensioned closure bolt will be verified to be bounded by the analysis that was referenced in the letter dated September 15, 2000 (WO 00-0036).

b. A review of the results of the visual examinations performed on the closure bolts shall be performed to ensure that there is no indication of sufficient degradation of closure bolts that could affect the conclusions of Specification 5.5.17a. above.

Within 30 days following startup of the plant, a report shall be submitted to the Commission identifying the circumstances for operation with one reactor vessel head closure bolt less than fully tensioned.

Operation with the same reactor vessel head closure bolt less than fully tensioned shall be limited to one operating cycle (i.e., until the next refueling outage).

TABLE 5.3-6

WOLF CREEK UNIT NO. 1 REACTOR VESSEL CLOSURE HEAD BOLTING MATERIAL PROPERTIES

Closure Head Studs

Heat No.	Material Spec. No.	Bar No.	0.2% Yield Strength (Ksi)	Ultimate Tensile Strength (Ksi)	Elongation (%)	Reduction in Area (%)	Energy at 10 F (FT LB)	Lateral Expansion (MILS)	BHN
84730	SA540, B24	505	139.0	155.0	17.0	52.2	52, 52, 51	30, 31, 28	331
84730	SA540, B24	505-1	142.2	157.0	16.0	51.4	50, 51, 50	30, 28, 26	331
84730	SA540, B24	510	144.7	158.0	16.0	51.9	49, 50, 49	30, 31, 27	311
84730	SA540, B24	510-1	141.0	156.0	16.0	52.6	54, 51, 52	34, 30, 28	321
84730	SA540, B24	512	144.5	160.0	16.0	51.4	48, 49, 48	29, 28, 30	341
84730	SA540, B24	512-1	141.0	155.5	15.5	51.5	53, 51, 51	32, 31, 32	331
84730	SA540, B24	515	141.5	157.5	16.0	51.7	51, 51, 55	31, 29, 38	321
84730	SA540, B24	515-1	140.5	155.5	16.5	53.8	54, 54, 55	36, 34, 33	321
84730	SA540, B24	521	135.5	153.0	17.0	56.0	53, 55, 56	31, 35, 35	331
84730	SA540, B24	521-1	143.7	160.0	17.5	53.8	49, 49, 50	26, 27, 27	331
84730	SA540, B24	528	143.0	159.0	17.5	55.7	51, 54, 55	33, 33, 33	331
84730	SA540, B24	528-1	143.0	158.0	17.5	53.8	56, 55, 55	35, 33, 34	321

Closure Head Nuts & Washers

63182	SA540, B24	132	148.0	162.0	17.5	57.3	51, 52, 51	31, 32, 30	331
63182	SA540, B24	132-1	148.7	162.0	17.0	54.7	49, 48, 49	29, 26, 29	331
63182	SA540, B24	133	147.2	161.0	17.0	55.2	52, 50, 51	31, 30, 30	321
63182	SA540, B24	133-1	149.2	162.5	17.5	54.7	51, 51, 49	29, 31, 27	331
63182	SA540, B24	135	147.5	161.0	17.0	53.0	49, 49, 51	28, 29, 30	321
63182	SA540, B24	135-1	143.2	157.0	17.5	55.2	55, 54, 52	33, 32, 31	321
63182	SA540, B24	137	145.0	159.0	16.5	54.8	54, 54, 53	33, 33, 29	331
63182	SA540, B24	137-1	147.0	160.0	17.0	55.7	54, 55, 54	34, 36, 33	321
63182	SA540, B24	143	145.0	159.0	18.0	58.1	55, 54, 54	33, 32, 32	331
63182	SA540, B24	143-1	147.0	160.0	17.0	57.3	54, 50, 52	33, 29, 30	321
63182	SA540, B24	145	145.0	159.0	17.0	56.0	54, 54, 55	34, 35, 34	321
63182	SA540, B24	145-1	146.2	159.7	17.0	57.0	56, 55, 54	36, 35, 36	331
63182	SA540, B24	148	144.0	157.5	17.5	56.5	56, 55, 55	33, 34, 34	331
63182	SA540, B24	148-1	148.5	162.0	17.0	55.6	52, 51, 52	33, 28, 30	321
63182	SA540, B24	150	144.7	158.0	17.5	55.7	55, 55, 54	33, 30, 31	331
63182	SA540, B24	150-1	145.7	160.0	17.0	56.5	53, 50, 52	33, 30, 31	331