WOLF CREEK
NUCLEAR OPERATING CORPORATION

Britt T. McKinney
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SEP 1 5 2000

WO 00-0036

U. S. Nuclear Regulatory Commission
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Washington, D. C. 20555

Subject: Docket No. 50-482: Application For Amendment To Technical Specification Table 1.1-1, "MODES"

Gentlemen:

In accordance with the provisions of 10 CFR 50.90, Wolf Creek Nuclear Operating Corporation (WCNOC) is submitting a request for an amendment to WCNOC's license NPF-42 to change the Technical Specifications for the Wolf Creek Generating Station (WCGS).

The proposed amendment would revise the WCGS Technical Specification (TS) Table 1.1-1, "MODES," notes that specify that all reactor vessel head closure bolts are required to be fully tensioned in MODES 1-5. Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-88, proposed a similar change to the Improved Standard Technical Specification. The NRC staff rejected TSTF-88 and indicated that the proposed change would require plant specific review.

Attachment I provides a detailed description, safety analysis of the proposed change, and WCNOC's determination that the proposed change does not involve a significant hazards consideration. Attachment II provides the existing TS page marked-up to show the proposed change. Attachment III provides the revised TS page. Attachment IV provides a summary of the licensing commitments made in this submittal.

WCNOC requests approval of the proposed License amendment by March 30, 2001, to be implemented within 60 days of the issuance of the License amendment. The approval date was administratively selected to allow for NRC review but the plant does not require this amendment to allow continued safe full power operation.

A001

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated Kansas State Official. If you should have any questions regarding this submittal, please contact me at (316) 364-4112, or Mr. Tony Harris at (316) 364-4038.

Very truly yours,

Britt T. McKinney

BTM/rlr

Attachments:
- I - Description and Assessment
- II - Proposed Technical Specification Changes
- III Revised Technical Specification Page
- IV - List of Commitments

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

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Britt T. McKinney, of lawful age, being first duly sworn upon oath says that he is Vice President Plant Operations and Plant Manager of Wolf Creek Nuclear Operating Corporation; that he has read the foregoing document and knows the content thereof; that he has executed that 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 Britt T. McKinney
Britt T. McKinney
Vice President Plant Operations and
Plant Manager

SUBSCRIBED and sworn to before me this 15th day of Sept., 2000.

Julie A. Dale
Notary Public

JULIE A. DALE
Notary Public - State of Kansas
My Appt. Expires 10/20/2002

Expiration Date 10/20/2002

ATTACHMENT I

Description and Assessment

Safety Analysis

1.0 INTRODUCTION

This proposed License Amendment Request (LAR) is a request pursuant to 10 CFR 50.90 to revise Wolf Creek Generating Stations (WCGS) Technical Specification (TS) Table 1.1-1, "MODES."

2.0 DESCRIPTION

The proposed License amendment would revise the TS Table 1.1-1, "MODES," notes that specify that all reactor vessel head closure bolts are required to be fully tensioned in MODES 1-5. Notes (b) and (c) to Table 1.1-1 state that MODE 6 is defined as "One or more reactor vessel closure bolts less than fully tensioned" and MODES 4 and 5 are defined as "All reactor vessel head closure bolts fully tensioned." The purpose of the Notes is to distinguish when the Reactor Coolant System (RCS) can be pressurized and when it cannot. Note (b) is revised from "All ...." to "At least 53 of 54 ...." Note (c) is revised from "One or more ..." to "Two or more ...." For the purposes of this amendment request, reactor vessel head closure bolts as specified in the Technical Specifications, are equivalent to reactor vessel closure studs. The required number of closure studs, which may be less than the total number, has been established by analysis that demonstrates adequate o-ring compression to prevent leakage and ensure that ASME Section III stress limits for affected components are not exceeded.

3.0 BACKGROUND

The basic function of the reactor vessel closure studs is to keep the upper and lower reactor vessel head flanges in metal-to-metal contact so that the two self-energizing hollow metallic o-rings can seal the closure head joint under all anticipated conditions. Two leakage monitors, one between the two concentric o-rings and one outside the outer o-ring, are provided to indicate that the o-rings are performing their sealing function.

The reactor vessel and associated head closure studs are designed and fabricated to the requirements of ASME Code, Section III (Reference 1), Class 1, 1971 Edition through Winter 1972. This ASME Code section specifies limits on closure stud membrane stress, stud membrane + bending stress, and fatigue.

During tensioning or detensioning of the reactor vessel head closure studs, there is some potential for getting closure studs stuck in a position where there is not enough thread engagement length to tension the closure stud without seriously overloading the vessel flange threads. Conditions also arise where there is some question regarding the integrity of a stud or threads after completion of the tensioning, i.e., where a closure stud is stuck in the vessel flange and cannot be removed for cleaning and/or inspection.

The ability of a reactor vessel to function with a single untensioned closure stud was demonstrated in 1987 at the Callaway Plant (Reference 2). In this case, a closure stud was stuck about half way into the vessel flange and it was not possible to tension the closure stud or remove it before the end of the outage. The Callaway Plant reactor vessel was operated for 18 months with this closure stud untensioned.

Industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-88, proposed a similar change to the Improved Standard Technical Specification. The NRC staff rejected TSTF-88 and indicated that the proposed change would require plant specific review. WCNOC had incorporated TSTF-88 into the license application for conversion to the improved Standard Technical Specifications and subsequently withdrew the change in response to Request for Additional Information Question Q 1.1-9 (WCNOC letter ET 98-0071 dated August 28, 1998).

4.0 TECHNICAL ANALYSIS

An evaluation (Reference 3) was performed to determine whether the reactor vessel would continue to meet ASME Code allowable stress criteria with a single untensioned closure stud, or with a single closure stud which fails in service. These conditions are most likely to occur when a closure stud becomes stuck in the vessel flange and it is not possible to remove the closure stud for inspection, or when a closure stud 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.

While this evaluation has determined that it is technically acceptable to operate with an untensioned or failed closure stud, it is not standard operating practice to operate in this condition. The main concern with an untensioned or failed closure stud is that there is some increased potential of o-ring seal leakage which is hard to quantify. Leakage is not expected to occur since adequate compression remains. However, if leakage were to result from having less than the total number of closure studs fully tensioned it would be detected by an increase in the temperature on the leak-off line from the annular space between the inner and outer vessel head o-rings. That temperature increase would be detected by installed temperature indicators and alarmed in the control room. Any leakage would be detected as an increase in RCS identified LEAKAGE and evaluated in accordance with plant procedures. Accordingly, the following precautions will be observed before operating with an untensioned or failed closure stud:

1. the particular circumstances will be reviewed to determine whether 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 heat-up rate will be held to 50 °F per hour (half of the typical 100 °F per hour design heat-up rate) until the closure stud is returned to service.

The approach used to evaluate an untensioned or failed closure stud (off-design conditions) is to: (1) determine the stresses in the closure stud and vessel for the intact case and the two off-design conditions under preload plus design pressure conditions, (2) determine the increase in stresses when going from the intact to off-design conditions, (3) add the predicted increase in

stress to the stress given in the design report, which includes the effect of thermal transients, and (4) determine whether the predicted off-design condition stresses still meet ASME Code requirements. The evaluation assumes that all closure studs are initially uniformly tensioned.

Primary Membrane Stresses in Closure Studs

The first step in the evaluation was to show that the primary membrane stresses in the closure studs do not exceed the ASME Code allowables under design temperature and pressure conditions for the case of one closure stud untensioned or failed in service. A calculation was performed to determine the primary closure stud stresses. The assumptions made in the calculation are:

1. The reactor vessel and head are rigid. This assumption is consistent with the usual treatment of primary loads which only considers net forces and moments and not localization of stress from geometry and compliance effects. A consequence of this assumption is that the distribution of stud forces varies linearly with the distance from the neutral axis.

2. The reactor vessel and head exert no contact forces on each other (i.e., the compression forces on the mating surfaces are neglected). These secondary forces serve to mitigate the redistribution of loads when studs fail, so this assumption conservatively maximizes the calculated maximum stud stress value.

3. The design pressure acts out to the radius of the inner o-ring of the vessel.

4. Each stud is individually treated as a point force.

The calculation determined the average stud force and stud force distribution with the following results:

PARAMETER	VALUE
Design Pressure (psig)	2,485
Inner o-ring radius (inches)	88.938
Intact Condition	
- Stud Load (kips)	1,143
- Stud Stress (ksi)	32.2
One Untensioned or Failed Stud	
- Average Stud Stress (ksi)	32.8
- Maximum Stud Load (kips)	1,211
- Maximum Stud Stress (ksi)	34.0

The calculation determined that the ASME Code primary membrane stress allowables are met with one stud untensioned or failed in service.

Finite Element Analyses

Finite element analyses of the reactor vessel in the intact condition and the two off-design conditions were performed using Revision 5 of the ANSYS finite element program (Reference 4) from ANSYS, Inc.

The case of an untensioned closure stud is simulated by deleting the beam element representing that stud and adjusting the initial strains on the studs adjacent to the untensioned stud to produce the specified preload. For the case of a stud that fails in service, the initial strain on all remaining studs is equal to that used in the intact case. If the spar elements simulating the vessel to head contact have axial tensile stresses in the pressurized condition, these elements are deleted and the coupled circumferential and radial constraints at these nodes are removed to simulate the fact that there is no longer a frictional restraining force at this location.

Six cases were evaluated as follows:

- Case A1 represents the preload condition with all studs intact
- Case A2 represents the operating condition with the reactor vessel at design pressure and with intact studs
- Case B-1 represents the case of one stud untensioned and with all other studs preloaded to the specified initial stress
- Case B-2 same as Case B1 except the reactor vessel is at design pressure
- Case C-1 represents the case of all studs preloaded to the specified initial strain with one stud failing in service
- Case C-2 same as Case C-1 except the reactor vessel is at design pressure

Although the ASME Code specifies that the service conditions should be used for analysis of the maximum ($3S_m$) and maximum average ($2S_m$) stresses in the bolting, the reactor vessel design pressure was used rather than the operating pressure. This is a conservative assumption. The analyses also assume that all parts are isothermal.

The finite element analyses determined that there was negligible effect of the off-design conditions on stress distributions in the reactor vessel and head. This is due to the fact that the flanges are very stiff relative to the closure studs such that loss of any single closure stud is not significant.

Comparison of Calculated Stresses to ASME Code Allowable Values

The following paragraphs summarize how the ASME Code allowable stress criteria are met for the off-design conditions. The general approach in each case is to determine the increase in the subject stress component which occurs during the off-design conditions relative to that for the intact design condition. This increase is then added to the maximum stress given in the vessel design report which includes the effects of all transient conditions. This new off-design maximum is then compared to the reported ASME Code allowable stress. Although some stress limits are not required to be evaluated at design temperature, all code stress limits in Table 1 were conservatively taken at the design temperature of 650 °F. The analysis results are summarized in Table 1.

1. Shell Stresses - Section III of the ASME Code contains the following requirements for vessel stress:

   a. The general primary membrane stress intensity shall not exceed S_m at design temperature.

   b. The local primary membrane stress intensity alone, or combined with a. above, shall not exceed 1.5 S_m.

   c. The local and general primary membrane plus primary bending stress intensity across the thickness of any solid rectangular section must be limited to 1.5 S_m. For the loadings considered in the vessel, satisfaction of this requirement automatically satisfies the local primary membrane requirement. Although Section III of the current ASME Code categorizes the local bending due to pressure (or boltup plus pressure) at the head/flange transition as a secondary stress, some design reports conservatively treat this stress as primary.

   d. The range of primary plus secondary stress intensity resulting from mechanical loads and general thermal effects shall not exceed 3 S_m.

Table 1 shows that there is no significant increase in any of these three stresses as a result of operating with an untensioned or failed stud. Therefore, operating at the off-design conditions will not increase the stresses above the allowables.

2. Maximum Stud Primary Membrane Stress - The results of the calculation discussed above and Table 1 show that the studs meet the ASME Code primary membrane stress requirements in the intact condition and in the condition with an untensioned or failed stud.

3. Stud Maximum Average Stress - The maximum stud service stress averaged across the cross-section must not exceed 2 S_m. The results in Table 1 show that this requirement is met.

4. Stud Maximum Stress - The maximum stud stress averaged across the cross-section must not exceed 3 S_m. The results in Table 1 show that this requirement is met.

5. Flange Separation at Inner O-Ring - The vessel design report notes that the maximum gasket opening is 0.005 inches and that the o-rings have a minimum springback of 0.013 inches. Therefore, since the additional separation at the inner o-ring location reported in Table 1 is less than 0.5 mils, there should be no leakage of the inner o-ring for the case of operation with one stud untensioned or failed in service.

6. Fatigue Life - As reported in Table 1, the maximum stud membrane + bending stress increases by 0.52 ksi (1%), and the maximum stress in the vessel increases by less than 0.4 ksi. The vessel design report shows a maximum cumulative fatigue usage factor anywhere in the head, head flange, and vessel flange as 0.02. Therefore, the small predicted increases in vessel stress levels will have no significant effect on fatigue usage.

The maximum reported fatigue usage factor on studs, including the effects of using the higher elongation acceptance limits, is 0.610. A calculated 1% increase in maximum stud service stress determined that the fatigue usage in the studs is acceptable. A check of the fatigue cycles not explicitly considered in the vessel report transient groupings determined that none of the less severe cycles account for any fatigue usage.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Determination

WCNOC has evaluated whether or not a significant hazards consideration is involved with the proposed changes by focusing on the three standards set forth in 10 CFR 50.92(c) as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   Overall protection system performance will remain within the bounds of the accident analyses, since no hardware changes are proposed. Since the stresses remain within Code allowables, the proposed change will not affect the probability of any event initiators nor will the proposed change affect the ability of any safety related equipment to perform its intended function. There will be no degradation in the performance of nor an increase in the number of challenges imposed on safety related equipment assumed to function during an accident situation.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   There are no hardware changes nor are there any changes in the method by which any safety related plant system performs it safety function. The method of plant operation is unaffected. Leakage would be precluded since, as noted in the evaluation, adequate compression remains. However, if leakage were to result from having less than the total number of closure studs fully tensioned it would be detected by an increase in the temperature on the leak-off line from the annular space between the inner and outer vessel head o-rings. That temperature increase would be detected by installed temperature indicators and alarmed in the control room. Any leakage would be detected as an increase in RCS identified LEAKAGE. Since stresses remain within Code allowables, no new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change.

Therefore, the proposed change will not create a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No

As indicated in Section 4.0 above, ASME Section III stress limits for effected components are not exceeded. The evaluations indicate that the reactor vessel will continue to meet ASME Code allowable stress criteria with a single untensioned reactor vessel closure stud, or with a single closure stud which fails in service. The proposed change does not alter nor exceed the acceptance criteria for any analyzed event. There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions.

Therefore, the proposed change to the Technical Specifications do not involve a significant reduction in the margin of safety.

Based on the above evaluations, WCNOC concludes that the activities associated with the above described change present no significant hazards under the standards set forth in 10 CFR 50.92 and that there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change. Moreover, because this change does not involve a significant hazards consideration, it will also not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

5.2 Regulatory Safety Analysis

Applicable Regulatory Requirements/Criteria

10 CFR Part 50, Appendix A, General Design Criterion (GDC) 14, "Reactor Coolant Pressure Boundary," requires that the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

10 CFR Part 50, Appendix A, General Design Criterion (GDC) 30, "Quality of Reactor Coolant Pressure Boundary," requires that components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant design.

U. S. NRC Regulatory Guide 1.65, "Materials and Inspections for Reactor Vessel Closure Studs," is NRC guidance which defines materials and testing procedures acceptable for implementing these criteria for reactor vessel closure stud bolting. (Reference 5)

Conclusion

The evaluations performed by WCNOC demonstrate adequate o-ring compression to prevent leakage and ensure that ASME Section III stress limits for affected components are not exceeded. The evaluations indicate that the reactor vessel will continue to meet ASME Code allowable stress criteria with a single untensioned reactor vessel closure stud, or with a single closure stud which fails in service. As such, Wolf Creek Generating Station continues to be in compliance with the above regulatory requirements.

6.0 ENVIRONMENTAL EVALUATION

WCNOC has determined that the proposed amendment would change requirements with respect to the installation or use of a facility component located within the restricted area, as defined in 10 CFR 20. WCNOC has evaluated the proposed change and has determined that the change does not involve (i) a significant hazards consideration, (ii) a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. As discussed above, the proposed amendment does not involve a significant hazards consideration and the reactor vessel will continue to meet ASME Code allowable stress criteria with a single untensioned reactor vessel closure stud, or with a single closure stud which fails in service. Accordingly, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

7.0 REFERENCES

1. USAR Section 5.3.1.7.

2. Letter dated May 26, 1988, from T. W. Alexion, NRC, to D. F. Schnell, Union Electric Co.

3. Report R-4328-00-1, Revision 0, "Reactor Vessel Bolting Evaluations, Wolf Creek Generating Station," Dominion Engineering, May 1997.

4. "ANSYS Engineering Analysis System," Revision 5.3, ANSYS, Inc.

5. USAR Appendix 3A, Regulatory Guide 1.65.

8.0 PRECEDENTS

Amendment No. 133 dated May 28, 1999, to Facility Operating License No. NPF-30 for the Callaway Plant, Unit 1, approved changes to Table 1.1-1, Notes b to require one less than the total number of reactor vessel head bolts be fully tensioned in MODES 4 and 5. The MODE 6 definition was likewise modified to denote a condition where less than the required number of bolts are fully tensioned.

Table 1
Stress Increases Due to Off-Design Conditions

Load Condition	Studs Missing	Load Case	Shell Stresses (ksi)(1) Gen. Membrane Head	Gen. Membrane Vessel	Local Memb.+Bend. Head	Local Memb.+Bend. Vessel	Maximum SI Head	Maximum SI Vessel	Stud Stresses (ksi) Primary Membrane	Max Membrane	Memb+ Bending	Flange Separation (10^ -3 in)
Preload Only	None	A1	0.52	0.32	19.18	12.66	19.37	13.09	0.00	39.56	59.68	1.89
Preload Only	1 Missing	B1	0.52	0.32	19.20	12.68	19.39	13.12	0.00	39.57	59.85	1.89
Preload Only	1 Failed	C1	0.52	0.32	19.19	12.67	19.38	13.12	0.00	42.30	60.36	1.89
Preload+Pressure	None	A2	15.51	20.61	30.26	22.38	30.50	22.46	32.2	42.23	64.66	2.67
Preload+Pressure	1 Missing	B2	15.51	20.61	30.24	22.39	30.48	22.46	34.0	42.38	64.83	3.07
Preload+Pressure	1 Failed	C2	15.51	20.61	30.24	22.38	30.48	22.50	34.0	45.12	65.18	2.67
Max. Increase from A2 (Cases B2 & C2)			0.00	0.00	0.00	0.01	0.00	0.04	1.8	2.89	0.52	0.40
Vessel Report Value			23.80	24.80	38.90	30.10	50.82	44.63	32.2	45.31	88.55	5.00
New Maximum Value			23.80	24.80	38.90	30.11	50.82	44.67	34.0	48.20	89.07	5.40
Code Stress Limit			Sm= 26.7	Sm= 26.7	1.5 Sm= 40.1	1.5 Sm= 40.1	3.0 Sm= 80.1	3.0 Sm= 80.1	Sm= 34.8	2 Sm= 69.6	3 Sm= 104.4	Max= 13.0 (2)

(1) Local Membrane + Bending and Maximum SI taken at cut lines 1, 2, and 3. General Membrane SI taken at cut lines 4 and 5.
(2) The flange separation limit of 13 mils is not actually a Code Stress Limit, but rather a vendor-specified minimum o-ring springback.

ATTACHMENT II

PROPOSED TECHNICAL SPECIFICATION CHANGES

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

At least 53 of 54

(a) Excluding decay heat.

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

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

ATTACHMENT III

REVISED TECHNICAL SPECIFICATION PAGE

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) At least 53 of 54 reactor vessel head closure bolts fully tensioned.

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

LIST OF COMMITMENTS

The following table identifies those actions committed to by Wolf Creek Nuclear Operating Corporation (WCNOC) in this document. Any other statements in this submittal are provided for information purposes and are not considered to be commitments. Please direct questions regarding these commitments to Mr. Tony Harris, Manager Regulatory Affairs at Wolf Creek Generating Station, (316) 364-4038.

COMMITMENT	Due Date/Event
The amendment will be implemented within 60 days of NRC approval.	Within 60 days of NRC approval
The following precautions will be observed 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 heat-up rate will be held to 50 °F per hour (half of the typical 100 °F per hour design heat-up rate) until the closure stud is returned to service.	Within 60 days of NRC approval