UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 161 TO FACILITY OPERATING LICENSE NO. DPR-40

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT NO. 1

DOCKET NO. 50-285

## 1.0 INTRODUCTION

By letters dated December 19, 1990, and June 1, 1992 and supplemented by letters dated February 1, 1993, and February 25, 1994, Omaha Public Power District (OPPD) submitted a request for changes to the Fort Calhoun Station (FCS), Unit No. 1 Technical Specifications (TS). The requested changes revise the TS by changing the pressure-temperature limits in TS 2.1.2 and would make the limits valid for 20 effective full-power years (EFPY) of operation. The amendment also modifies TS 2.1.1 to change the minimum requirements for starting a non-operating reactor coolant pump (RCP) and modifies TS 2.3(3) to change the requirements for disabling high-pressure safety injection (HPSI) pumps during scheduled heatup and cooldown operations. Lastly, the amendment modifies TS 2.1.6 to change the power-operated relief valve (PORV) limiting conditions of operation (LCO) and surveillance requirements. The amendment request was filed in response to Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)," dated June 25, 1991.

Generic Issue (GI) 70, "Power-Operated Relief Valve and Block Valve Reliability," involves the evaluation of the reliability of PORVs and block valves and their safety significance in PWR plants. The generic letter discussed how PORVs are increasingly being relied on to perform safety-related functions and the corresponding need to improve the reliability of both PORVs and their associated block valves. Proposed staff positions and improvements to the plant's technical specifications were recommended to be implemented by all affected facilities. This issue is applicable to all Westinghouse, Babcock & Wilcox, and Combustion Engineering (CE) designed facilities with PORVs.

Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light-Water Reactors," addresses concerns with the implementation of the requirements set forth in the resolution of Unresolved Safety Issue (USI) A-26, "Reactor Vessel Pressure Transient Protection (Overpressure Protection)." The generic letter discussed the continuing occurrence of overpressure events and the need to further restrict the allowed outage time for a low-temperature overpressure (LTOP) protection channel in Operating Modes 4, 5, and 6. This issue concerns only Westinghouse and CE facilities.

The February 1, 1993, and February 25, 1994, letters provided clarifying information that did not change the initially proposed no significant hazards consideration determination.

## 2.0 EVALUATION

### 2.1 Generic Issue 70

The actions proposed by the NRC staff to improve the reliability of PORVs and block valves represent a substantial increase in overall protection of the public health and safety and a determination has been made that the attendant costs are justified in view of this increased protection. The technical findings and the regulatory analysis related to GI 70 are discussed in NUREG-1316, "Technical Findings and Regulatory Analysis Related to Generic Issue 70 - Evaluation of Power-Operated Relief Valve Reliability in PWR Nuclear Power Plants."

In response to the NRC recommendation, Fort Calhoun Station (FCS) included PORVs and block valves within the scope of an operational quality assurance program that is in compliance with 10 CFR Part 50, Appendix B. Fort Calhoun specified that the PORVs and associated block valves are designated as Critical Quality Equipment (CQE) per the FCS CQE List and Computerized History and Maintenance Planning System (CHAMPS) Equipment Database. This designation ensures that they have been installed, are tested and repaired or replaced under requirements specified in USAS B31.7, ASME Section III and ASME Section XI. The PORVs and their associated block valves are listed on the FCS CQE List and as such are tested and maintained in accordance with the quality assurance requirements as set forth in 10 CFR Part 50, Appendix B.

Fort Calhoun explained that the PORVs and associated block valves have been evaluated using the guidelines as stated by the FCS Preventative Maintenance (PM) Program and the appropriate tasks are scheduled/performed in accordance with this program. The PM Program task basis and content is controlled by the Special Services Engineering Department. The PM Program is implemented by the FCS Maintenance Department by personnel trained and qualified under FCS Standing Orders and Administrative Procedures. The Program evaluation includes the manufacturer's recommendations which have been evaluated by Special Services Engineering personnel for applicability.

Replacement parts, including spares as well as complete components, are procured in accordance with the original construction codes and standards as required. Parts/components are repaired/replaced and retested using ASME Section XI repair and replacement criteria or Equipment Qualification guidelines as applicable. CQE parts and components are inspected by FCS Quality Control personnel upon receipt and are controlled using CQE receipt and storage requirements.

Fort Calhoun stated that they include the PORVs and block valves within the scope of a program covered by subsection IWV, "Inservice Testing of Valves in

Nuclear Power Plants," of Section XI of the ASME Boiler and Pressure Vessel Code. The block valves are stroke tested quarterly in the closed direction under the Inservice Test Program. In addition to the Inservice Testing of the block valves in accordance with ASME Section XI requirements, the block valves are also included in the expanded MOV Test Program (GL 89-10).

The PORVs are stroke tested in both the open and closed direction prior to entering Cold Shutdown in accordance with the guidelines set forth in the FCS Inservice Inspection (ISI) Program Plan, 3rd Ten Year Interval. The PORVs are tested at a pressurizer pressure between 350-450 psia and a Reactor Coolant System (RCS) temperature of between 300-350°F. Testing the PORVs at this temperature and pressure with a steam bubble in the pressurizer ensures that steam is released to the Quench Tank when the PORVs are stroked open. This allows for a more controllable test pressure and ensures that the position indicators are reading accurately. This test pressure is higher than that listed in the ISI Program Plan, Revision 5 and was increased to eliminate a concern raised by FCS Operations about the possibility of RCS subcooling which could cause a loss of net positive suction head to the Reactor Coolant Pumps. OPPD's official copy of the ISI Program Plan has been revised to reflect this change.

The PORVs are 2 1/2" Dresser valves (1 3/32" internal orifice) with a soft seat and are designed to open on a differential pressure. The low pressure testing of the PORVs conservatively tests the operability of the valves. To test the PORVs at a higher pressure could result in damage to the soft seat, should a leak develop. In addition, the conditions under which a low-temperature overpressure transient is most likely to occur is when the reactor coolant temperature is less than or equal to 200°F. Low-temperature overpressure protection (LTOP) transients that have challenged the overpressure protection systems at other nuclear facilities have occurred with the RCS temperatures in the range of 80°F to 190°F. Under the current FCS ISI Program Plan, the PORVs are tested at 300-350°F, significantly above this temperature.

OPPD is confident that periodic testing of the PORVs at this pressure/temperature, along with the administrative controls (e.g., Operating Instructions, Operating Procedures), is adequate to ensure the operability of the PORVs when needed to prevent damage to the RCS due to potential overpressurization at low temperatures.

The action statements in TS 2.1.6(5)a. through d. were modified or added to ensure that the operability requirements of GL 90-06 were incorporated. The LCO statement was clarifying by replacing "all" with "both". The requirement to maintain power to closed block valve(s) was included because removal of power would render the block valve(s) inoperable, and the requirements of action statement c. would apply. Power is maintained to the block valve(s) so that it is operable and may be subsequently opened to allow the PORV to be used to control RCS pressure. Closure of the block valve(s) establishes the RCP boundary integrity for a PORV that has excessive seat leakage. The integrity of the RCP boundary takes priority over the capability of the PORV to mitigate an overpressure event.

Action statements b. and c. include the removal of power from a closed block valve as additional assurance to preclude any inadvertent opening of the block valve at a time in which the PORV may not be closed due to maintenance to restore it to operable status.

Action statement d. has been modified to establish remedial measures that are consistent with the function of the block valves. The primary function is the capability to close the block valve to isolate a stuck-open PORV. Therefore, if the block valve(s) cannot be restored to operable status within one hour, the remedial action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck-open PORV at a time that the block valve is inoperable. The time allowed to restore the block valve(s) to operable status is based upon the remedial action time limits for inoperable PORVs from action statement b. and c. since the PORVs are not capable of mitigating an overpressure event when placed in manual control. These actions are also consistent with the use of the PORVs to control RCS pressure if the block valves are inoperable at a time when they have been closed to isolate PORVs with excessive seat leakage.

The operating mode for meeting LCO commitments (HOT STANDBY) in the section 2.1.6 action statements a. to d., described above, and the time to achieve cold shutdown conditions (24 hours) were changed to be consistent with the Fort Calhoun TSs for safe shutdown of the unit. The safe design for Fort Calhoun is HOT SHUTDOWN. Technical Specification 2.0.1 allows 36 hours to achieve cold shutdown from a hot shutdown condition. The recommendations in GL 90-06 were reviewed and the LCO action statements were modified to meet the intent of the GL, yet remain consistent with the other Fort Calhoun TS action statements. This remains consistent with the design and operating license requirements for Fort Calhoun.

Surveillance requirement 22 in Table 3-3 of the TSs is proposed for modification to allow an exception for testing the block valves when they are closed for isolation of an inoperable PORV. If the block valve is closed to isolate a PORV with excessive seat leakage, the operability of the block valve is of importance, because opening of the block valve is necessary to permit the PORV to be used for manual control of reactor pressure. If the block valve is closed to isolate an otherwise inoperable PORV, the maximum allowable outage time is 72 hours, which is well within the allowable limits (25%) to extend the block valve surveillance interval (92 days). Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to an operable status. The position of the PORV block valves will be verified on a daily basis in response to the requirements of GL 90-06.

The times to complete the action statements have been conservatively reduced to ensure prompt compliance with the requirements for safe operation.

A number of items were not incorporated into the Fort Calhoun TSs as requested by GL 90-06 and are discussed below. The verification of the PORV block valve position is completed on the control room log (Form FC-75) once a day which is more frequent than the 72 hour requirement Specification 4.4.9.3.c contained in the GL. The reporting requirements in Specification 3.4.9.3.e are redundant to reporting requirements of 10 CFR 50.73; thus, the reporting requirement will not be added to the FCS TSs since the requirement to produce an account of the event is redundant to the Code of Federal Regulations.

The staff has reviewed the licensee's proposed modifications to the FCS technical specifications. Since the proposed modifications are either consistent with the staff's position previously stated in the generic letter or justified in the above evaluation, the staff finds the proposed modifications acceptable.

2.2 Generic Issue 94

The actions proposed by the NRC staff to improve the availability of the LTOP system represents a substantial increase in the overall protection of the public health and safety and a determination has been made that the attendant costs are justified in view of this increased protection. The technical findings and the regulatory analysis related to GI 94 are discussed in NUREG-1326, "Regulatory Analysis for the Resolution of Generic Issue 94, Additional Low-Temperature Overpressure Protection for Light-Water Reactors."

Operating Modes 4 and 5 were added to TS 2.1.6.(4) to specify requirements in addition to normal heatup and cooldown operations. The applicability of the PORV operating requirements for Modes 4 and 5 was also added. One PORV is allowed to be inoperable for up to 7 days if the RCS is not water solid. This allows for the expansion of the coolant during a heatup event. One PORV may only be inoperable for 24 hours if the RCS is in a water solid condition. The time periods meet the intent of the GI 94 resolution contained in the GL 90-06. A 72 hour time period was specified to reach cold shutdown and complete depressurization and venting if both PORVs are inoperable. The time period for depressurization and venting is longer than that contained in the GL due to the safe shutdown mode for FCS. Since FCS was designed as a hot shutdown plant, it requires a longer time period to reach a cold, depressurized condition without compromising plant or personnel safety. The definition of "venting" was also added to the basis to indicate an area greater than 0.94 $in.^2$, which is equivalent to the cross sectional area of a PORV.

The LTOP analysis was performed with the objective of evaluating and modifying if necessary, the existing provisions for low temperature overpressure protection at FCS to ensure that reactor coolant pressure boundary integrity will continue to be maintained in low temperature modes of operation.

The primary objective of an LTOP system is to automatically prevent pressure excursions above the applicable P-T limits during pressurization events that could result from operator error or equipment malfunction.

Technical Specifications which are affected by the LTOP system requirements determined for FCS by the analysis are those which concern the requirements for starting the first idle RCP and the limits on HPSI pump availability during heatup and cooldown.

One of the bases of the LTOP analysis by ABB CE is a steam generator temperature that is less than 30°F above that of the reactor coolant system (RCS) cold leg. This assumption reduces the severity of the pressure transient associated with the start of the first RCP when the steam generator temperature exceeds that of the RCS cold leg, and, therefore, allows for a greater operating window.

It was also determined as part of the LTOP analysis that a pressurizer steam space of greater than or equal to 53% would ensure that the start of a RCP, when no other reactor coolant pumps are in operation, would not result in an overpressurization of the RCS if the secondary temperature exceeds that of the RCS cold leg by 30°F or more. The basis for "RCS venting" was also defined more specifically as an area equal to or greater than 47 in.$^2$, removing the reference to pressurizer manway which was to reduce the potential for misinterpretation of the actual vent area requirement.

These assumptions modify the current TS 2.1.1(11) which requires either a 60% pressurizer steam space or less than a 50°F secondary-to-primary temperature differential for the start of a non-operating RCP. The revisions to the TSs clarify that these requirements apply only to the case where no RCPs are currently in operation.

Another assumption that was made in the LTOP analysis that must be translated to a TS is the limitation on allowed HPSI pump operability during heatup and cooldown. TS 2.3(3) is modified as follows:

> Whenever the RCS cold leg temperature is below 385°F, at least one (1) HPSI pump shall be disabled.
>
> Whenever the RCS cold leg temperature is below 320°F, at least two (2) HPSI pumps shall be disabled.
>
> Whenever the RCS cold leg temperature is below 270°F, all three (3) HPSI pumps shall be disabled.
>
> In the event that no charging pumps are operable, a single HPSI pump may be made operable with mass input restricted to that no greater than the three charging pump flow rate,and utilized for boric acid injection to the core.

These limitations on HPSI pump operability either reduce or eliminate the potential for pressurization events. This allows for an expanded operating window and improved heatup and cooldown rates.

The staff has reviewed the licensee's proposed modifications to the FCS technical specifications. Since the proposed modifications are either consistent with the staff's position previously stated in the generic letter or justified in the above evaluation, the staff finds the proposed modifications acceptable.

## 2.3 Pressure - Temperature (P-T) Limits

The Fort Calhoun TS 2.1.1, 2.1.2, 2.3(3), and Figures 2-1A, 2-1B and 2-3 are amended to update the current pressure-temperature (P-T) limits, as well as specifications related to the LTOP system, for continued operation beyond 14 EFPY. The Fort Calhoun TS amendment requested continued operation through 20 EFPY.

To evaluate the P-T limits, the staff uses the following NRC regulations and guidance: 10 CFR 50.36(c)(2); Appendices G and H of 10 CFR Part 50; Generic Letter 88-11; Regulatory Guide (RG) 1.99, Rev. 2; and Standard Review Plan (SRP) Section 5.3.2.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide Technical Specifications for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the Technical Specifications. The P-T limits are among the limiting conditions of operation in the Technical Specifications for all commercial nuclear plants in the U.S.

Appendix G to 10 CFR Part 50 requires that "When the core is not critical pressure-temperature limits for the reactor vessel must be at least as conservative as those obtained by following the methods of analysis and the required margins of safety of Appendix G of the ASME Code..." Appendix G also imposes requirements on the minimum temperature for criticality, the closure head flange, and hydrostatic pressure tests or leak tests.

Appendix H of 10 CFR Part 50 requires licensees to establish a surveillance program to monitor embrittlement of reactor vessel materials. The program includes capsules that contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and be removed from the reactor vessel periodically for testing. The test results may be used in calculating P-T limits.

Generic Letter 88-11 suggested that licensees use the methods in RG 1.99, Rev. 2, to predict the embrittlement effect of neutron irradiation on reactor vessel materials. The embrittlement effect is defined in terms of adjusted reference temperatures (ART), which is the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the calculation.

SRP 5.3.2 describes a calculation of the P-T limit curves based on the principles of linear elastic fracture mechanics. SRP 5.3.2 calculation follows the methodology specified in Appendix G to the ASME Code, Section III.

The licensee determined that, at 20 EFPY, lower shell longitudinal weld 3-410A is the limiting material. The chemistry for weld 3-410A used in the licensee's calculation was 0.21% copper (Cu) and 1.00% nickel (Ni). The licensee used a margin of 66°F and an initial $RT_{ndt}$ of -56°F in calculating ARTs for weld 3-410A. The licensee calculated the limiting ARTs of 298°F at the 1/4T location (T = reactor vessel thickness at the beltline region) and 241°F at the 3/4T location at 20 EFPY based on Position C.1 of RG 1.99.

The staff has identified the same material, weld 3-410A, as limiting. Based on the staff review of the pressurized thermal shock review (Ref. 1) and ongoing GL 92-01 review, the staff identified the same copper content (0.21%) and nickel content (1.00%) as the licensee had identified for weld 3-410A. The staff verified that the initial $RT_{ndt}$ and margin used in the licensee's calculation are acceptable. Based on the above data and a licensee reported neutron fluence of 1.5E19 n/cm$^2$ on the inside surface of the reactor at 20 EFPY, the staff calculated an ART of 238.5°F at the 1/4T location and 187.5°F at the 3/4T location (Reference 2).

The licensee's calculated ARTs are more conservative than the staff's calculated ARTs because the licensee's ARTs were calculated with a higher neutron fluence. The licensee's ARTs were calculated two years ago in an anticipation of the P-T limits amendment. Since then, the licensee has implemented a flux reduction program, resulting in lower neutron fluences on the inside surface of the reactor. The licensee could have revised its ARTs for the current P-T limits submittal; however, the licensee chose to be conservative. The staff finds that the licensee's ARTs are acceptable.

Based on SRP 5.3.2, the staff verified that the proposed P-T limits for heatup, cooldown, criticality, and inservice hydrostatic test meet the requirements in Paragraphs IV.A.2 & IV.A.3 of Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes a minimum temperature at the closure head flange based on the reference temperature for the flange material. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange reference temperature of 10°F, the staff has determined that the proposed P-T limits have satisfied the requirement on the bolt preload.

The licensee has removed surveillance capsules W-225 and W-265 from Fort Calhoun Unit 1 and has performed required tests. The staff has determined that the surveillance program has satisfied Appendix H to 10 CFR 50.

Pressure instrument loop uncertainties are not included in the P-T limits since these limits have been included in the LTOP PORV trip setpoints. The pressure loop uncertainties have been included in the portion of the P-T limits which is above the LTOP enable temperature, i.e., 385°F.

The staff has performed an independent analysis of the P-T limits to verify the licensee's proposed limits. The staff concludes that the proposed P-T limits for heatup, cooldown, inservice hydrostatic test, and criticality are valid through 20 EFPY because the limits conform to the requirements of Appendix G of 10 CFR Part 50 and Generic Letter 88-11. Hence, the proposed P-T limits may be incorporated in the Fort Calhoun unit 1 Technical Specifications.

2.4 Administrative Changes

The amended P-T Figures 2-1A and 2-1B were renamed "RCS Pressure-Temperature Limits for Heatup" and "RCS Pressure-Temperature Limits for Cooldown" vice "RCS Press-Temp Limits Heatup" and "RCS Press-Temp Limits Cooldown." The amended TS changes the Table of Contents page viii, to account for this renaming of figures.

This change is administrative in nature; and therefore, the staff finds it acceptable.

## 3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Nebraska State official was notified of the proposed issuance of the amendment. The State official had no comments.

## 4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (57 FR 30255). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

## 5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

## 6.0 REFERENCES

1. Letter dated December 3, 1993, from S. Bloom of USNRC to T. Patterson of Omaha Public Power District, subject: Fort Calhoun Station, Unit No. 1- Amendment No. 158 to Facility Operating License No. DPR-40 (TAC No. M82834).

2. Letter from W. G. Gates, Omaha Public Power District to USNRC "Updated Information on Reactor Vessel Structured Integrity and Construction Period Recovery for Fort Calhoun Station (TAC Nos. M82834 and M83465)", October 15, 1993.

Principal Contributors: Steven Bloom
John Tsao

Date: March 23, 1994