SAFETY EVALUATION REPORT OF

THE LOW TEMPERATURE OVERPRESSURE

PROTECTION SYSTEM DESIGNED FOR

FORT CALHOUN UNIT NO. 1

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 39 TO LICENSE NO. DPR-40

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT NO. 1

I. Introduction

By letter dated August 5, 1977 (Reference 7) Omaha Public Power District (OPPD) submitted an application for amendment to Facility Operating License No. DPR-40 for the Fort Calhoun Station Unit No. 1 (Fort Calhoun). This application requests additions to the Technical Specifications appended to the Fort Calhoun Operating License which would incorporate the proposed low temperature overpressure protection system (OPS) into the Limiting Conditions for Operation and Surveillance Requirements.

The system proposed by OPPD for Fort Calhoun incorporates a defense in depth concept for overpressure protection, utilizing operator training, administrative procedures, Technical Specifications, and hardware improvements to meet the criteria established by the staff. The objective of the OPS is, first, to insure that pressure transients while operating at low reactor coolant system (RCS) temperatures become and remain unlikely events, and second, to mitigate the consequences of a pressure transient should one occur. The proposed mitigating system includes sensors, actuating mechanisms, and valves to prevent a RCS pressure transient from exceeding the pressure-temperature limits included in the Fort Calhoun Technical Specifications as required by Appendix G to Chapter 10, Code of Federal Regulations, Part 50 (10 CFR 50). These Appendix G limits are those established by using procedures defined in Appendix G to Section III of the ASME Code. Appendix G to 10 CFR 50 states that these ASME Code limits can be used for startup and shutdown when the reactor is not critical. For criticality, Appendix G to 10 CFR 50 requires more stringent rules than Appendix G to Section III of the ASME Code.

II. Background

The history of the generic low temperature overpressure protection issue is described in NUREG-0138 (Reference 1). Briefly, a series of over thirty incidents had occurred in pressurized water reactors (PWRs) since 1972 in which the Appendix G pressure-temperature limits had been exceeded at temperatures less than normal operating temperature.

These incidents consisted of two varieties of pressure transients: a mass input type from charging pumps, safety injection pumps, or safety injection accumulators, and an energy input type caused by thermal feedback when a reactor coolant pump (RCP) sweeps cooler primary system water through a steam generator with a hot secondary side. These incidents usually occurred in a water solid system during startup or shutdown operations.

Pressure transients which could occur at normal operating temperature, approximately 570°F, are mitigated in most plants by large code safety valves located on the pressurizer. These are mechanical valves which open against a spring pressure of about 2400 psia. The code safety valves are quite simple, having no electrical components, and as such are considered passive, failure free components. These code safety valves are tested in accordance with ASME Code, Section XI requirements.

Prior to the introduction of an OPS, pressure transients initiated while operating at lower temperatures were not protected against and there were no pressure relief devices in the reactor coolant system to prevent these transients from exceeding the Appendix G pressure-temperature limits. A reactor such as Fort Calhoun, which has a pressure limit in excess of 2500 psia at 570°F, has only a 700 psia limit at 200°F. The code safety valves with settings in the 2400 psia range would not be able to relieve a pressure transient at low RCS temperature without the Appendix G limits being violated by a large amount.

The Appendix G pressure limit drops off rapidly at lower temperatures because the reactor vessel material and welds have significantly less toughness at lower temperatures and are, therefore, more susceptible to flaw induced failure. In addition, factors such as copper content in welds and neutron fluence levels affect the material toughness and contribute to the reduction in safety margin to vessel failure at low temperature conditions.

As a solution to the low temperature overpressurization problem, the licensee identified a set of power operated relief valves (PORVs) located on the pressurizer which are normally available for overpressure protection during normal plant operations. These usually have a single pressure setpoint just below the opening pressure of the mechanical code safety valves and are designed to relieve small pressure transients without requiring the code safety valves to lift. The licensee proposed to provide the PORV's with a low pressure setpoint to which they could be switched as the plant cooled down. If a pressure transient would occur at these lower temperatures and the lower setpoint had been selected, there would then be a pathway to relieve system pressure.

The PORV's are significantly more complicated than the code safety valves in that they require electrical circuitry to sense pressure, transmit a signal to the valve, and actuate the solenoid to open the valve. Thus, it is desirable to insure redundancy and separability in the circuitry to preclude a single failure from disabling the entire OPS system.

In a series of meetings and through correspondence with PWR vendors and licensees, the NRC staff developed a set of criteria, which if adhered to, would produce an acceptable OPS. These criteria are:

1. Operator Action:

   The licensee could not take credit for operator action for 10 minutes after the operator became aware of an ongoing transient.

2. Single Failure:

   The system had to be designed to relieve overpressure transients assuming the worst case single failure in addition to the event which caused the transient.

3. Testability:

   The system had to be testable on a periodic basis consistent with the system's employment.

4. Seismic and IEEE 279 Design:

   Ideally the system should meet seismic Class I and IEEE 279 design requirements. The basic objective is that the system should not be vulnerable to a common failure mode which both initiated a pressure transient and caused a failure of equipment needed to terminate the transient.

In addition to the four formally stated criteria mentioned above, additional criteria were established during NRC staff's review of generic submittals from the various vendors and in the exchange of information between the NRC staff and the licensees.

Foremost among these was the requirement that the licensees show protection for the limiting mass addition transient regardless of the administrative procedures proposed to eliminate that potential scenario. Each licensee, therefore, was required to analyze the effects of the single pump start which would produce the most limiting mass addition transient and most severely challenge the Appendix G limits.

For the worst case energy addition transient the licensees were allowed to limit the severity of the transient in their analyses by assuming a maximum ΔT across the steam generator. By maximum ΔT we mean the maximum difference in the temperature between the primary loop coolant and the secondary loop water in the steam generator. For this case and for other scenarios the licensees were required to develop Technical Specifications which delineated the actions required to limit the severity of these scenarios and also provide justification for their action.

Another criterion for the design of the OPS required the electrical instrumentation and control system to provide a variety of alarms that (1) alert the operator to properly enable the low temperature OPS at the proper temperature during cooldown, and (2) indicate if a pressure transient was occurring. Additionally the electrical system had to provide positive assurance that the isolation valve upstream of each PORV was open when the system was enabled by wiring its position into the enable alarm. The enable alarm would not be permitted to clear until the OPS mode selector switch for each PORV system was placed in the low pressure setpoint position and the isolation valve was opened.

OPPD submitted a generic overpressurization protection report prepared by Combustion Engineering (CE) in Reference 2. This report was prepared for the CE Owner's Group comprising five utilities. The generic report provided information on RCS response to postulated pressure transients that could occur at low temperatures during heatup and cooldown and provided a general description of design modifications which could be used to prevent overpressurization of CE designed Nuclear Steam Supply

Systems (NSSS). The NRC staff, in conjunction with its review of the CE generic report, requested that the OPPD commit to a schedule for implementing a permanent or interim version of the OPS by December 31, 1977, and requested additional information related to the application of the generic aspects of the OPS as pertinent to the Fort Calhoun plant (Reference 3). In References 4 and 5 the licensee submitted additional information to the NRC staff on equipment and procedural improvements as well as a schedule for implementation of the proposed system. The Fort Calhoun plant specific report was submitted in Reference 6 and additional plant specific data was supplied by the licensee in Reference 7.

III. System Description and Evaluation

A. Procedures and Technical Specifications

One cornerstone of the Fort Calhoun OPS is the use of procedures and Technical Specifications to limit the probability of initiating pressure transients at low temperatures (<300°F) and to insure the enabling, disabling, and proper functioning of the OPS. Procedures and Technical Specifications described and submitted by the licensee proposed to accomplish the following:

1. Reactor Coolant System Operable Components

A non-operating reactor coolant pump shall not be started unless at least one of the following conditions is met: (1) a pressurizer steam space of 60% by volume or greater exists; or (2) the steam generator secondary side temperature is less than 50°F above that of the reactor coolant system cold leg. These two procedures are proposed to prevent a reactor coolant pump induced overpressurization and are acceptable.

2. Pressurizer and Steam System Safety Valves

Whenever there is fuel in the reactor, and the reactor vessel head is installed, a minimum of one operable safety valve shall be installed on the pressurizer. However, when in at least the cold shutdown condition, safety valve nozzles may be open to containment atmosphere during performance of safety valve tests or maintenance to satisfy this specification.

During scheduled cooldown of the reactor coolant system the cold leg temperature shall not be brought below 300°F until the pressurizer power-operated relief valves (PORV) have been set to the low setpoint. The PORVs shall remain at the low setpoint whenever the cold leg temperature is below 300°F and the reactor vessel head is installed.

These proposed changes improve maintenance capability by allowing substitution of an open safety valve nozzle for an open safety valve and provide additional capability to prevent pressure transients from resulting in an overpressurization event. Thus, they are acceptable.

3. Emergency Core Cooling System

Whenever the reactor coolant system cold leg temperature is below 210°F and the reactor vessel head is installed, at least two (2) HPSI pump control switches shall be placed in pull-stop. Whenever the reactor coolant system cold leg temperature is below 110°F and the reactor vessel head is installed, all three (3) HPSI pump control switches shall be placed in pull-stop. In the event that no charging pumps are operable, a single HPSI pump may be taken from pull-stop and utilized for boric acid injection to the core.

Inadvertent safety injection constitutes one of the most severe potential reactor coolant system pressure transients. These changes are intended to sharply reduce the probability of such a transient resulting in a RCS overpressurization incident and are acceptable.

4. Minimum Frequencies for Checks, Calibration, and Testing of Instrumentation and Controls

The system is designed such that the actuating circuitry can be tested, up to the PORV solenoids, without opening either PORV or rendering the overpressure protection system inoperable.

The proposed revision to the Technical Specification requires testing of the actuator circuitry prior to bringing cold leg temperature below 300°F. This procedure will assure operability of the actuating circuitry prior to reaching the pressure/temperature regime in which overpressure protection is required. This test will be repeated on a monthly basis for periods in which the cold

leg temperature remains below 300°F. Calibration of the pressure and temperature channels during each refueling is consistent with the frequency utilized for other critical RCS instrumentation. Testing of the PORVs will be in accordance with the OPPD inservice inspection program. We find the above proposed changes to be acceptable.

5. Technical Specifications

The proposed Technical Specifications provide the operability and testing requirements for the OPS components. Revisions to some of the Technical Specification pages, contained in the August 5, 1977 OPPD submittal, were necessary to incorporate changes that have been made to those pages which were approved and issued by license amendments subsequent to the submittal of the OPS proposal.

B. Hardware

1. OPS Functioning

Acceptable performance of the OPS depends on the proper functioning and adequate relief capacity of the two PORVs located on the pressurizer. The NSSS vendor and the licensee demonstrated that with two PORVs functioning, all postulated mass and energy addition transients could be mitigated. If one PORV is assumed to fail, administrative procedures must be relied upon to limit the severity of the limiting transients in both the energy and mass addition cases to insure that Appendix G limits are not violated.

2. Energy Addition Transients

Administrative procedures, backed by Technical Specifications, require that the maximum ΔT across the steam generator be less than 50°F to lessen the consequences of the RCP start energy addition transient. This assures that a single PORV can function to relieve this transient and maintain RCS pressure below Appendix G limits. A ΔT of 100°F would cause Appendix G limits to be exceeded. Numerous assumptions were employed in the modeling of PORV relief to insure conservatism in the analysis of this design base energy addition transient.

a. The RCS was assumed to be water solid.

b. The RCS was also assumed to be rigid during the transient (no expansion).

c. A single PORV was assumed to fail.

d. Letdown flow was assumed isolated.

e. Heat absorption by the RCS metal mass was not considered.

f. Conservatively high heat transfer coefficients were utilized across the steam generator.

g. RCP start was assumed to be instantaneous.

With a PORV low pressure setpoint of 465 psia the licensee showed that one PORV will provide sufficient relief capacity to give a maximum pressure of ~475 psia for an energy addition transient. This analysis assumed a ΔT of no more than 50°F across the steam generator. The licensee will monitor the ΔT across each steam generator and provide annunciation if the ΔT exceeds 50°F.

We conclude that the licensee and vendor have demonstrated that the OPS can protect the RCS from exceeding Appendix G limits for an energy addition transient even with the additional single failure of a PORV. This conclusion assumes, as does the analysis, that the steam generator ΔT remains less than 50°F.

3. Mass Addition Transients

Protection from the effects of the limiting mass addition transient was afforded by the licensee by assuring that components of the ECCS system would be disabled by procedure and Technical Specification during cooldown. This is accomplished at 210°F by placing two HPSI pump control switches in a pull-stop position and caution tagging them and by placing the third HPSI pump switch in the same configuration at 110°F. This provides assurance that Appendix G limits will not be violated should a single PORV fail prior to or during a mass addition transient. Conservatisms employed in the limiting mass addition transient model the inadvertent single HPSI pump start, were as follows:

a. The RCS was assumed to be water solid.

b. Letdown flow from the RCS was assumed isolated.

c. The RCS was considered to be rigid during the transient (no expansion).

d. Mass addition was assumed to occur with the highest fluid density that could occur. PORV relief was assumed to occur with the lowest fluid density that could occur.

e. A single PORV was assumed to fail.

f. A conservative Bernoulli equation was utilized to model PORV relief.

The NRC staff guidance to the licensee for analyzing the mass addition transient was to show that Appendix G limits were not violated assuming that the safety injection pump which could produce the worst case transient inadvertently started regardless of administrative procedures calling for disabling the pumps at various stages. For the Fort Calhoun plant the worst pump start would be a HPSI pump. The licensee demonstrated that a single HPSI pump plus three charging pump mass input transient would produce a peak pressure of 465 psia, the PORV valve low pressure setpoint. The quick opening time of the valve (~3 milliseconds) results in the transient immediately being relieved, resulting in the equilibrium pressure of 465 psia. This corresponds to an Appendix G limit that would exist at temperatures well below the refueling temperature (~130°F).

We conclude that the licensee has demonstrated that the OPS will prevent overpressurization of the RCS due to mass addition transients assuming the single failure of a PORV.

4. System Testing

The staff position with regard to testability requires the PORV control circuitry and pressure detection sensors to be tested prior to any reliance on the overpressure protection system. The licensee has agreed to test the enabling and actuation circuitry up to the PORV solenoids, without opening either PORV or rendering the overpressure protection system inoperable, prior to each scheduled

cold leg cooldown below 300°F and monthly, whenever the temperature is below 300°F and the reactor vessel head is installed. A test switch, in each channel, will supply an "open" signal to the actuation circuitry. Supervisory and indication lights will provide verification of proper circuit operation. The temperature and pressure channels will be calibrated each refueling and the PORVs will be operated each refueling. We find that the Fort Calhoun OPS properly adheres to our testing criterion and is, therefore, acceptable.

C. System Electrical Description

1. Discussion

The control circuitry for the low temperature overpressure protection system has been designed to comply fully with IEEE Std 279-1971, and seismic Category I criteria. Complete separation of both channels is maintained through elimination of the common variable setpoint and reset switches. As such the system is completely redundant with either channel being capable of actuating both PORVs in the event that an overpressure transient is detected.

The low setpoint PORV actuation circuitry is automatically enabled only when the pressurizer low pressure signal is manually bypassed and the cold leg temperature is below 316± 6°F. A key-operated reset switch is necessary to return the system to the high setpoint. Once the low setpoint is enabled, it seals in. A key-operated switch is also provided for each PORV to allow manual opening or closing. While these switches would bypass the automatic opening circuitry, they would not bypass the low setpoint enabling circuitry. Manual closing of a PORV is annunciated in the control room. Enabling or actuation of a single channel low setpoint circuitry for one PORV automatically enables or actuates the actuation circuitry for the second redundant PORV. Once the low setpoint circuitry has been enabled, an increase in the reactor coolant pressure above the low setpoint automatically opens the PORV. Annunciation is provided if either PORV selector switch is placed in the test position. We find these design features to be acceptable.

2. Monitoring ΔT Across Steam Generator

The licensee has designed a monitoring system which will measure the feedwater temperature in each steam generator and compare it with the reactor coolant system cold leg temperature. This system will function by draining steam generator feedwater through the blowdown system and past temperature elements; one per steam generator. The design is such that the measured temperature accurately reflects steam generator temperature with annunciation being provided for ΔT greater than 50°F. Prior to installation of the proposed permanent system, an interim design using thermocouples will be installed on the steam generator shell or secondary-side handholes. This temporary system will be sufficient and provide a reasonably accurate determination of steam generator temperature. Operation of the reactor coolant pumps relative to the 50°F ΔT limitation will be in accordance with the proposed Technical Specification and corresponding operating instructions. The staff does not require that this alarm circuitry be protection grade. We find this acceptable.

3. Isolation Valve Alarm

In accordance with the staff position requiring an isolation valve alarm, the licensee has agreed to provide annunciation on control room panel CB-1/2/3 to indicate closure of either PORV isolation valve. This alarm will not clear until both isolation valves have been opened thus assuring a complete pathway from the pressurizer to the quench tank. The staff does not require that this alarm circuitry be protection grade. We find this design to be acceptable.

4. PORV Train Separability

The staff position on PORV train separability of circuits requires that the PORVs should each have a variable setpoint switch and a hi-low reset switch, i.e., two separate key-locked circuits, one for each PORV. The proposed low setpoint PORV actuation circuitry will be automatically enabled only when the cold leg temperature is below 300°F and the pressurizer pressure low signal has been manually bypassed. The common variable setpoint and reset switches in the existing design have been eliminated and complete separation of both channels is maintained. The design of this circuitry satisfies the requirement of IEEE Std 279-1971. We find this to be acceptable.

5. PORV Cycling

The design of the low temperature overpressure protection system is such that following low setpoint actuation of a PORV, reclosure of the PORV would not take place until the reactor coolant system pressure is reduced by 50-100 psig below the setpoint. This design feature is acceptable because it reduces rapid PORV cycling. We conclude that cycling of a PORV may be treated as a series of individual control actions and that the protection system requirement for completion of a protective action (para 4.16 of IEEE Std 279) is satisfied.

IV. Summary

The system presented by Omaha Public Power District to provide protection for the Fort Calhoun plant from low temperature overpressure transients provides assurance that these transients will be unlikely events and that, should they occur, the plant will be protected. Analyses of the limiting mass and energy addition transients were accomplished to demonstrate plant protection from scenarios which could overpressurize the RCS.

The overall low temperature overpressure protection system design in the area of electrical, instrumentation and control (EI&C) is in accordance with those design criteria as originally stated by the staff and later expanded in subsequent discussions with the licensee.

We find the EI&C aspects of the proposed design acceptable on the basis that: (1) the proposed control circuitry complies fully with IEEE Std 279-1971 criteria and is designed as a seismic Class I system; (2) the system is redundant and meets the single failure criterion; (3) the design requires no operator action for ten minutes after the operator receives an overpressure action alarm; (4) the system is testable on a periodic basis; and (5) the proposed changes to the Technical Specification reduce the probability of overpressurization events to acceptable levels.

We conclude, therefore, that the Fort Calhoun OPS meets the criteria established by the NRC staff for overpressure protection and is acceptable as a low temperature overpressure protection system. We further conclude that the Technical Specifications submitted with the proposed OPS design are in consonance with the proposed OPS and are, therefore, acceptable.

V. Conclusions

Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: March 7, 1978

VII. REFERENCES

1. U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from Director, NRR to NRR Staff," NUREG-0138, November 1976.

2. Letter, Short to Lear, Docket No. 50-285, December 2, 1976 forwarding "Generic Report Overpressure Protection for Operating CE NSSS," December 3, 1976.

3. Letter, Lear to Short, Docket No. 50-285, January 12, 1977.

4. Letter, Short to Lear, Docket No. 50-285, March 2, 1977.

5. Letter, Short to Lear, Docket No. 50-285, March 24, 1977.

6. Letter, Short to Lear, Docket No. 50-285, May 10, 1977, forwarding "Specific Plant Report, Low Temperature Overpressure Protection for Fort Calhoun Unit 1", May 1977.

7. Letter, LeBoeuf, Lamb, Leiby, and MacRae to Case, Docket No. 50-285, August 5, 1977, forwarding Application for Amendment of Operating License, Docket No. 50-285; forwarding Letter, Short to Lear, Docket No. 50-285, August 1, 1977; forwarding "Fort Calhoun Unit No. 1, Low Temperature Overpressure Protection System, Request for Information."