Omaha Public Power District
1623 Harney Omaha, Nebraska 68102-2247
402/536-4000

June 28, 1988
LIC-88-477

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, DC 20555

References:
1. Docket No. 50-285
2. Letter from NRC (A. Bournia) to OPPD (R. L. Andrews) dated February 17, 1988
3. Letter from OPPD (R. L. Andrews) to NRC (Document Control Desk) dated May 27, 1938 (LIC-88-384)

Gentlemen:

SUBJECT: Response to Request for Additional Information concerning NUREG-0737, Item II.D.1

The Omaha Public Power District (OPPD) received Reference 2 which detailed the NRC staff and its consultant's review of NUREG-0737 Item II.D.1, Performance Testing of Relief and Safety Valves for Fort Calhoun Station.

Reference 3 was OPPD's response to the questions listed in Reference 2. The response to Question 12 indicated that additional time would be required to fully respond to the question and that OPPD's response would be submitted by June 30, 1988. Attached please find our response to Question 12. Also attached please find copies of Boeing Computer Services memoranda which were not included in Reference 3.

If you have any questions, do not hesitate to contact us.

Sincerely,

W. Gary Gates
for R. L. Andrews
Division Manager
Nuclear Production

Attachments

RLA/me

8807070532 880628
PDR ADOCK 05000285
P PDC

A046
1/1

c: LeBoeuf, Lamb, Leiby & MacRae
R. D. Martin, NRC Regional Administrator
P. D. Milano, NRC Project Manager
P. H. Harrell, NRC Senior Resident Inspector

ATTACHMENT

NRC Question 12:

NUREG-0737, Item II.D.1 requires that the plant specific PORV Control Circuitry be qualified for design-basis transients and accidents. OPPD's response to this was, "The control circuitry for the PORV is, for the most part, located outside of the containment building, in the switchgear and control rooms. As such, it would not be subjected to a harsh environment. The solenoid valves which open the PORVs are located at the PORVs inside containment. For the Fort Calhoun Station, the transients which might challenge the PORVs, namely loss of load or loss of feedwater flow, do not create a harsh environment in the containment. In the highly unlikely event that both PORVs failed to open when challenged, either of the two safety valves could provide more than enough capacity to handle the amount of steam that would be generated."

The licensee's statement is considered evasive since it does not address the pertinent requirements of NUREG-0737, Item II.D.1, namely, accidents and transients inside the containment that subject the PORV circuitry to harsh environment during which the PORV may operate.

The staff has agreed that meeting the licensing requirements of 10 CFR 50.49 for this circuitry is satisfactory and that specific testing per NUREG-0737 requirement is not required. Therefore verify whether the PORV control circuitry has been reviewed and accepted under the requirements of 10 CFR 50.49.

If the PORV circuitry has not been qualified to the requirements of 10 CFR 50.49, provide information to demonstrate that the control circuitry is qualified per the guidance provided in Reg. Guide 1.89, Revision 1, Appendix E.

As an alternative, the staff has determined that the requirements of NUREG-0737 regarding the qualification of the PORV control circuitry may be satisfied if one or more of the following conditions is met.

NRC POSITION

12a. The PORVs are not required to perform a safety function to mitigate the effects of any design basis event in the harsh environment and failure in the harsh environment will not adversely impact safety functions or mislead the operator (PORVs will not experience any spurious actuations and, if emergency operating procedures do not specifically prohibit use of PORVs in accident mitigation, it must be ascertained that PORVs can be closed under harsh environment conditions).

OPPD RESPONSE

The following discussion provides OPPD's analysis of the expected PORV operations and anticipated plant response, during various plant operating modes.

1.0 NORMAL OPERATION (RCS > 1700 PSIA)

For the purposes of the PORV review "normal operation" is considered to be above 1700 psia where Pressurizer Low-Low Pressure (PPLS) is enabled (unblocked) and Low Temperature Over-Pressure Protection (LTOP) is thus disabled. For the normal operating condition, only the loss of load or the power increase events are applicable. In both cases the Pressurizer Quench Tank (PQT) is sized adequately to contain the primary system volume released, thus no harsh environment is created. Should one or both PORVs stick open the RCS transient which would occur is bounded by the LOCA analysis. A LOCA qualified acoustic monitoring system for PORV flow is available to insure the solenoid limit switches do not mislead the operator as to the PORV's open or closed status. Please note that per OPPD's EEQ Program the PORV controls are not identified as being LOCA qualified while the PORV acoustic flow monitor is identified as being qualified via an orange dot on the control board. This permits the operator to easily identify qualified and thus reliable instrumentation.

2.0 HEATUP AND COOLDOWN LTOP OPERATION (RCS <1700 PSIA)

During heatup and cooldown (RCS ≤1700 psia) the PPLS circuit is blocked and the LTOP circuit is enabled. In the event of a pressure excursion, per the LTOP circuit setpoints, the PORVs serve to reduce the RCS pressure. Under this condition the volume of RCS discharge is not specifically known; however, OPPD believes it to be less than the volume of discharge as a result of an overpressure condition under "normal operation". The PQT system is designed to contain the entire RCS discharge resultant from a full power loss of load trip. Therefore, the PQT system would remain intact for an overpressure condition under LTOP operation and no harsh environments would be created. This is considered conservative because of the significant difference in the energy within the RCS under the two conditions. Under "normal operation" the RCS pressure would be decreasing from 2100 psia, the cold leg temperature would be 535°F or greater, and decay heat would be exponentially decreasing from 100% reactor power. The conditions for which the LTOP system is enabled would be RCS fluid pressure of 1700 psia or less at approximately 450°F with decay heat below 1% reactor power. The reactor coolant pump heat would be the only other major source of heat input to the system for both scenarios. Therefore, any LTOP transient would not be as severe [illegible] an RPS high pressurizer pressure - PORV transient.

The heat removal characteristic of the steam generators is not always obvious in the comparison of RPS PORV function versus the LTOP PORV function. The main steam safety valves and steam dump and bypass valves are assumed to function properly for decay heat removal in the case of high pressurizer pressure. In the case of a heatup or cooldown overpressure, the RCS pressure, temperature, and decay heat removal are controlled by manual control of steam generator steaming. The recovery from either situation would require the use of the steam generator(s) for heat removal.

The existing LTOP configuration is adequate and the PQT would be expected to remain intact. Should the PORVs open and the PQT rupture disc burst, operator action can be tak[illegible]ntrol RCS pressure and close the PORV block valves. Should the [illegible] to close the event is still bounded by the LOCA analysis wit[illegible]w indication available for the PORVs.

3.0 NORMAL OPERATION CONFIGURATION - LOCA RESPONSE

The LOCA response of the LTOP circuit due to temperature input failure would generate a PORV open signal. This would be prevented during power operation by PPLS being unblocked which disables the control capability of the LTOP circuit.

The area of concern here is the post-LOCA action where engineered safeguards are reset. PPLS must be blocked as the first step to reset safeguards, which could automatically open the PORVs. OPPD will add a step to AOP-23, Safeguards Reset Procedure, to clearly require the PORV control switches to be placed in "close" position prior to blocking PPLS. The control circuitry required to prevent energization of the PORVs would not be subjected to a potentially harsh environment.

4.0 LTOP OPERATION CONFIGURATION - LOCA RESPONSE

In the event of a LOCA occurring when the LTOP is enabled, spurious actuation could occur. It is judged that the existing configuration is adequate for the following reasons:

a. Operating time with the RCS below 1700 psia and above 300 psia is limited to approximately one to two heatup and cooldown cycles per year. Each heatup or cooldown process has a duration of approximately 48 hours which greatly reduces the probability of a concurrent LOCA.

b. One of the first operator actions following a LOCA is to unblock PPLS to initiate safeguards. This deenergizes the PORV solenoids; unless mechanical binding occurs, the solenoids should reposition. The solenoids are large masses of copper and iron which generate heat when energized and thus would not be greatly influenced by initial stages of LOCA induced transients. Exposure to steam heating in the first moments of a LOCA is not expected to cause binding.

c. Any PORV failure would be bounded by the LOCA analysis and qualified position indication is provided to prevent operator confusion.

5.0 PORV SOLENOID LIMIT SWITCH & SOLENOID

5.1 The PORV solenoid limit switch failure is bounded by the LOCA qualified acoustic flow position indication position indication discussion in Section 1.

5.2 The PORV solenoid failure in a harsh environment is considered bounded by the LOCA analysis in the case of an open PORV. A failure to open analysis is not required, as PORV opening in a harsh environment is not required.

6.0 POST LOCA LONG TERM CORE COOLING USING THE PORVs

6.1 The LOCA analysis states that the PORVs can be used for long term core cooling in the event both steam generators are not available. Both steam generators would be unavailable only in the event all feedwater (main and auxiliary) was lost. The AFW system is considered to meet single failure criteria for events requiring decay heat removal and is of adequate reliability, thus the total loss of feedwater is not considered credible. In addition, OPPD has committed to the addition of a third AFW pump to further improve system reliability.

7.0 ONCE THROUGH COOLING USING THE PORVs

7.1 Once through cooling of the core is required only in the event of a loss of all feedwater (main and auxiliary). In this mode the PORVs would be opened and HPSI pumps would be used to provide make-up to the RCS. This cooling mode is discussed only in EOP-20 Success Path HR-4 and is not part of the USAR 14.10 Malfunctions of the Feedwater System Analysis.

Once through cooling is not considered as part of the Fort Calhoun design basis. The auxiliary feedwater system is considered of adequate reliability and meets Fort Calhoun Station design basis single failure criteria for events requiring decay heat removal.

NRC POSITION

12b. The PORVs are required to perform a safety function to mitigate the effects of a specific event, but are not subjected to a harsh environment as a result of that event.

OPPD RESPONSE

The PORVs function is discussed in the response to 12a, see Sections 1.0, 2.0, 3.0 and 4.0.

NRC POSITION

12c. The PORVs perform their function before being exposed to the harsh environment, and the adequacy of the time margin provided is justified; subsequent failure of the PORVs as a result of the harsh environment will not degrade other safety functions or mislead the operator (PORVs will not experience any spurious actuations and, if emergency operating procedures do not specifically prohibit use of PORVs in accident mitigation, it must be ascertained that PORVs can be closed under harsh environment conditions).

OPPD RESPONSE

The PORVs function is discussed in response to 12a, see Sections 1.0, 2.0, 3.0 and 4.0.

NRC POSITION

12d. The safety function can be accomplished by some other designated equipment that has been adequately qualified and satisfies the single-failure criterion.

OPPD RESPONSE

See response to 12a, Sections 5.0 and 6.0.

CONCLUSION

OPPD believes that the previously discussed configuration is adequate to insure safe operation in all plant operating modes requiring PORV operation and that adequate control and indication has been provided to mitigate potential accident failure modes of the PORVs.

ATTACHMENT 1

BCS BOEING COMPUTER SERVICES COM

ATTACHMENT FOR QUESTION 10.

September 14, 1982
G-7610-190

To:	C. R. Harvey	CV-45
	D. Johnson	7A-44
cc:	R. W. Blohm	7A-21
	R. D. Broad	6K-39
	E. J. Corrie	7A-20
	D. P. Konichek	9A-02
	R. C. Lundquist	7A-36
	J. F. Presti	7A-21
	M. J. Synge	7A-21
	J. L. Tocher	9C-02
	J. C. Turley	7A-23
	R. Vontoble	6K-39

Subject: Certification: Force V2

The FORCE program is certified to perform as described in attachment 1.

Technical Requirements for Class B, Regulated, described in document G-40356.01, have been med as evidenced in attachment 2.

Conditional Certification, Class B and Category Regulated is granted. Unconditional Certification will be granted upon completion of an audit of EECCL, Nuclibe Vendor, for development and maintenance practice.

FORCE will be installed on the EKS Mainstream and VSP Services.

J. C. Jervert
ETA Quality Assurance

F. A. Hanna
Engineering and Scientific Services

Attachments

Attachment I

1. MAINSTREAM-EKS, FORCE, Reference Manual and Access Guide, 10208-2032, July 1982

2. Test Report, FORCE, G-7623-046, August 27, 1982

BCS BOEING COMPUTER SERVICES COMPANY

A Division of The Boeing Company

November 1, 1982 - Revision
G-7623-046R

To: B. Block
F. Hanna
B. Mukherji
S. Pruitt
C. Wolfe

Subject: FORCE Version 2 Quality Assurance, Product Test Report, BCS QA Certification (QA Section 2.3)

Reference: Memo G-7623-028, Test Plan for RELAP 5/1 and FORCE, dated June 28, 1962 (QA Section 2.1.2)

Test Procedure Execution Results (QA Section 2.3.2)

The test cases set forth in the referenced Test Plan were run on the BCS operating system as planned. All files used to create and test this version have been stored on tape (Attachment A). The test case runs, including input and output, are bound in "FORCE Quality Assurance Standard Test Case Set and Hand Calculation Test" and are a part of this certification.

Test Analysis (QA Section 2.3.3)

The results of the three test cases cited in the standard Test Case Set (Section 2.1.2), QA-7, QA-9, and QA-10 are presented as follows:

1. The QA-7 (Combustion Engineering Test 1411 Safety Release Valve, BCS Model) case was run to provide test output from customer initated problems and to demonstrate ability to handle safety release valve tests. Cursory examination of output indicates that the code handles this case successfully.
2. Case QA-9 (Combustion Engineering Steam Test 1411 Safety Release Valve) results matched the Combustion engineering and EPRI results shown in the attached plots, the valves being the same as the "BCS" curve. Attachment B shows an earlier comparison done for verification and Attachment C replicates these results for FORCE certification.
3. Case QA-10 is a simplified model of a pipe and it was run to provide a manageable hand calculational case intended to reinforce that FORCE performs its calculations properly. FORCE calculates forces in a hydraulic pipe based on fluid and gas conditions. The calculations are made from density, velocity, pressure, and time parameters and the geometry of the pipe segments under consideration. A run of this case on the RELAP 5/1 code was made and the output used as input to FORCE. The calculations and comparison with the run are shown in Attachment D

The actual BCS computer runs of cases QA-7 through QA-10, as above, showing both input and output, are bound as part of the certification package.

Test Deficiencies (QA Section 2.3.4)

No deficiencies were found in the testing of FORCE.

Any deficiencies discovered in future will be given in the On-Line News/Error file (see QA Section 3.8).

D. P. Konichek
D. P. Konichek
11/10/82

Attachments:

ATTACHMENT FOR QUESTION 11

October 8, 1984
:4-DAJ-103

To: J. C. Jervert

Subject: T-PIPE, BCS Version 2.1--National Certification, Class A and Category Regulated

Reference: Memo G-7430-MAG-024, Dated November 30, 1979, M. A. Groce to C. S. Bartholomew, et al., Subject T-PIPE Release Certification

T-PIPE, BCS Version 2.1, was previously certified with no qualifications per the reference memo. Subsequently, the category of "Regulated" was used to define a set of certified products that were to be used by the nuclear industry customer set and had completed the certification process as defined by Quality Assurance Procedures 40356.01 series. T-PIPE, BCS Version 2.1, has been certified by this process and has met all requirements for certification to the category of "Regulated."

T-PIPE is a program which performs stress analysis of piping systems. It provides static analysis, dynamic analysis, NRC Regulatory Guide 1.92 mode combination methods, ASME Class I thermal transient analysis, and stress classification according to ASME Boiler and Pressure Vessel Code, Section III, Class 1, 2, and 3, and ANSI B 31.1.

This letter is issued for the purpose of upgrading the Quality Assurance records regarding the category of this product.

Dave

D. A. Johnson, Manager
Headquarters Quality Assurance
7C-36, 763-5122

DAJ:svv

103

November 30, 1979
G-7430-MAG-024

See File

To: ~~C. S. Bartholomew~~
R. J. Flynn
K. D. Johansen
S. L. S. Jacoby
O. M. Langdahl

cc: E. J. Corrie
M. J. Synge
J. L. Tocher
F. L. Wise

Subject: T-PIPE Release Certification

The status of each of the items identified in the T-PIPE Certification Data Package is attached. Federal Systems Group's Energy Technology Applications Division is responsible for T-PIPE Product Support, including configuration management.

It is the recommendation of the T-PIPE Certification Committee that T-PIPE be certified as a BCS Class A product with full certification.

M. A. Groce

M. A. Groce
T-PIPE Certification Committee
Chairman

Attachment