Omaha Public Power District

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October 5, 1979

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Reference: Docket No. 50-285

Dear Mr. Denton:

The Omaha Public Power District received your letter of September 17, 1979, requesting that a review be conducted of the Fort Calhoun Station on the subject of a potential unreviewed safety question on interaction between non-safety grade systems and safety grade systems. The potential problem was further addressed in IE Information Notice 79-22, dated September 14, 1979. This letter is in response to your request.

The District has reviewed the specific non-safety grade systems listed in IE Information Notice 79-22, as well as others, for potential interactions that could constitute a substantial safety hazard. In this effort we were assisted by our reactor vendor, Combustion Engineering. No interactions constituting a substantial safety hazard were identified. While in some cases we have identified variations from the Fort Calhoun Stati n FSAR licensing bases and identified control system modifications to enhance the level of safety, the basic conclusion of the FSAR that these events do not constitute an undue risk to the health and safety of the public remains unchanged. Therefore suspension, revocation, or modification of Operating License DPR-40 is not warranted.

As a result of the Three Mile Island accident, there are a significant number of industry, governmental and regulatory investigations underway examining the licensing bases and the operating procedures of nuclear generating facilities. These investigations are already identifying areas w ere studies may result in the consideration of new or revised events as part of the bases for assuring continued safety of nuclear facilities. NUREG-0578 outlines several such events and suggests remedies.

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NUREG-0578 requirements for analyses of potential safety problems envision the kinds of scenarios identified by Westinghouse and made the subject of IE Information Notice 79-22. Section 3.2, page 17, states in part:

"...The NRC requirements for non-safety systems are generally limited to assuring that they do not adversely affect the operation of safety systems..."

Further, on page A-45 of NUREG-0578:

"Consequential failures shall also be considered..."

We therefore believe that the scope of the action required by IE Information Notice 79-22 is fully encompassed by the requirements of NUREG-0578 and should therefore be integrated with the planned response sequence for compliance with the NUREG. As such, the District will continue to evaluate this concern, in conjunction with our efforts to respond to NUREG-0578.

Sincerely,

W. C. Jones
Division Manager
Production Operations

WCJ/KJM/BJH:jmm

Attach.

cc: LeBoeuf, Lamb, Leiby & MacRae
1333 New Hampshire Avenue, N. W.
Washington, D. C. 20036

Sworn and subscribed to before me this

day of , 1979.

Notary Public

GENERAL NOTARY - State of Nebraska
J. T. GLEASON
My Comm. Exp. July 26, 1982

ATTACHMENT A

EVALUATION SUMMARY

The Omaha Public Power District has reviewed Fort Calhoun Station instrument systems as required by the Commission's letter of September 17, 1979, and as addressed in IE Information Notice 79-22. The following systems have been identified as those which require consideration under the subject letter's guidelines.

1. Feedwater Regulation - including feed flow, steam flow, steam generator level, the analog system, and valve positioning.
2. Remote Operated Main Steam Safety Valves - MS-291 and MS-292.
3. Pressurizer Power Operated Relief Valves.
4. Pressurizer Pressure Control.
5. Pressurizer Level Control.
6. Steam Dump and Bypass System - including atmospheric dump.
7. Reactor Regulation/Automatic Rod Control.
8. Steam Generator Blowdown.

Each of the systems was evaluated, taking into consideration the Fort Calhoun Station Unit No. 1 main steam line break; feedwater line break; LOCA analysis; and Combustion Engineering report CEN-114-P, Small Break LOCA Analysis. The FSAR control rod ejection accident was also considered, but is not discussed further in this summary since the accident consequences were found to be bounded by the small break LOCA analysis.

The following assumptions establish the means by which the most limiting control system failure scenario was developed.

1. Unqualified equipment, which is exposed to environmental conditions caused by a high energy line break, fails in the most adverse direction.
2. All qualified equipment operates as required by its inputs.
3. Equipment not exposed to the high energy break operates as required by its inputs.
4. Random failures do not occur in the control systems.

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5. Protection system operation is that assumed in the FSAR.

6. Operator action is the same as assumed in the FSAR and as presently indicated by operating procedures.

The following summary provides results and recommendations of the District's evaluations.

1. Feedwater Regulation System

   a. System Description

      The feedwater regulation system consists of two (one dedicated to each steam generator) 3-element (feed flow, steam flow, level) control systems for automatic operation in the range of approximately 30% to 100% power. The system provides the capability for remote-manual positioning of the main feedwater regulating valves. Remote-manual operation of feedwater bypass (around the main feedwater regulating valves) valves is also provided. This system contains no environmentally qualified control equipment. Therefore, the feedwater control system is postulated to fail as a result of a high energy line break in Room 81 (outside of containment) or inside containment.

   b. Impact of Feedwater Regulating System Malfunction During LOCA

      In reviewing the LOCA analysis in report CEN-114-P, it was noted that, for line breaks above .02 sq. ft., feedwater has no effect on the analysis; therefore, feedwater regulation system failure would not affect the capability to mitigate the consequences of this accident. For breaks smaller than .02 sq. ft., it has been shown in report CEN-114-P that establishing auxiliary feedwater flow within 30 minutes results in acceptable consequences. This 30 minute requirement is presently addressed in the station emergency procedures. Therefore, the loss of feedwater has no adverse consequences for the LOCA transient and current operating procedures assure initiation of auxiliary feedwater.

      The failure of the feedwater control system in the "full feed" mode has the potential for overfilling the steam generators. In this case, numerous alarms are available to the control room operator and the operator would either use manual main feedwater control or he can trip the main feedwater pumps and establish auxiliary feedwater using normal procedures. Under normal circumstances, no operator action is required to prevent overfilling of steam generators because when ECCS is initiated main feedwater flow is automatically isolated.

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c. Impact of Feedwater Regulating System Malfunction During Main Steam Line Break Accident

The main steam line break analysis (for both inside and outside containment) shows that feedwater is assumed to ramp to 5% in 60 seconds and the SGLS and ECCS would actuate to mitigate the accident. A failure maintaining full feedwater flow would cause a slightly accelerated shrinkage of reactor coolant system inventory until a turbine/reactor trip is initiated and ultimately ECCS actuation occurs. At this time, the closing of the feedwater isolation valves would terminate the transient. If the feedwater flow continued as a result of a single failure of an isolation valve, manual action would be taken to trip the main feedwater pumps and terminate the transient. Emergency procedures currently provide for this action.

An immediate feed flow failure during a main steam line break would result in only a slightly smaller steam generator inventory. This scenario would result in a less severe transient, due to the reduction in available secondary side inventory, and would cause a less severe primary system temperature transient than the analyzed steam line break transient.

2. Remote Operated Main Steam Safety Valves

The Fort Calhoun Station remote operated main steam safety valves (MS-291 and MS-292) consist of two safeties, one on each main steam line, which are equipped with air operators attached to the manual actuator handles of these code safeties. These were provided as a method to remove heat from the primary system with the main steam isolation valves closed. These valves are located in Room 81 (outside of containment).

As a result of this evaluation, the solenoid operators on these valves have been replaced with LOCA qualified solenoids. This eliminated the possibility of a cooldown of both steam generators if a main steam line or feedwater line broke in Room 81, which resulted in the failure of the solenoid valve on the intact steam line safety valve and a heat extraction with no means of isolating the safety valve. However, during the time the unqualified solenoid were installed, it was felt that this type of failure would not be instantaneous and, due to the blowout domes and relative low pressure in Room 81 (1.5 psig vs. 60 psig LOCA), the failure may never have occurred. Appendix M of the FSAR analyzed these effects on Room 81.

3. Pressurizer Power Operated Relief Valves

The Fort Calhoun Station pressurizer power operated relief valves consist of two electrically (solenoid plunger) actuated valves, which operate in the high pressurizer pressure mode and in the low temperature overpressure protection mode. These valves are

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also equipped with manually actuated motor operated block valves. In analyzing the operating modes of these two control systems, it was found that neither system is capable of inadvertently actuating the PORV's during any evaluated accident. The high pressurizer pressure 2-out-of-4 logic to open system from the RPS uses LOCA qualified equipment to generate the signal. Therefore, no further consideration is required. Although the low temperature protection equipment is not qualified, it is interlocked with the station pressurizer pressure low block signal, such that when the ECCS actuation (PPLS) is armed the low temperature system is not armed. Thus, no matter what the failure mode of the unqualified equipment, it does not actuate the valve. In addition, the solenoid itself is a passive device and will not actuate by itself under high energy break conditions.

4. <u>Pressurizer Pressure Control System</u>

The pressurizer pressure control system consists of two independent control systems (pressurizer spray valves and pressurizer proportional heaters), only one of which is selected to control pressurizer pressure by the operators. The pressure transmitter inputs and the spray valve controls to this system could be affected in the high energy line break situation. In either failure mode (maximum spray with all or no heaters or minimum spray with no or maximum heaters) there is no effect on the LOCA or the main steam line break analysis, since this heat input from the proportional heaters or heat removal from pressurizer spray is not significant.

5. <u>Pressurizer Level Control System</u>

The Fort Calhoun Station pressurizer level control system is a two-channel system (either may be selected to control), which maintains the pressurizer water inventory as a function of average reactor coolant temperature (T average). Included in this control scheme are charging pump starts and stops, backup pressurizer heater starts and stops, regenerative heat exchanger controls, and letdown valve control. The various equipment is actuated to correct level, depending on the deviation from the T average calculated level.

In reviewing the LOCA and main steam line break accident, no safety problem was identified. Report CEN-114-P for small breaks does not identify any charging or letdown concerns. The initiation of ECCS will isolate letdown and trip all backup heaters.

For the large break LOCA, pressurizer level control has no limiting cases which would affect the analysis.

The main steam line break also uses ECCS to mitigate the consequences. This essentially overrides all level control equipment. However, assuming a small break in the main feed line occurs inside of containment, the potential exists for an increased pressurizer inventory. This is because the pressurizer level transmitters may fail causing the pressurizer to fill for a short period of time before a reactor/turbine trip is initiated on low steam generator level.

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LOCA qualified level transmitters will be installed on the pressurizer level control system in order to provide proper post-accident indication to operating personnel and to mitigate the possibility of overfilling the pressurizer. These transmitters will be replaced during the next refueling outage.

6. <u>Steam Dump and Bypass System</u>

The Fort Calhoun Station dump and bypass system consists of two control systems and the associated valves, which are designed to limit primary and secondary transients as a result of a turbine trip. The steam dump is designed to actuate on turbine trip and stay open until the average coolant temperature is 532°F. The steam bypass system also opens on turbine trip and controls the secondary system pressure at 900 psia and 532°F saturation temperature.

For the main steam line break accident, a steam dump failure would either contribute to the initial blowdown or have no effect (if closed). In either case, the closure of the main steam isolation valves will mitigate the steam dump failure.

For the small break LOCA, no credit was taken for the steam dump system.

A large break LOCA would initiate CPHS, which closes the main steam isolation valves, mitigating the effect of the steam dump and bypass.

Also considered under this category was the steam dump system, which includes atmospheric dump, downstream of the main steam isolation valves in Room 81. This evaluation provides the same results as for the steam dump to the condenser discussed above.

7. <u>Reactor Regulation System</u>

The reactor regulation system (automatic rod control system) was disabled electrically on August 5, 1973. No control system failure can result in automatic rod movement.

8. <u>Blowdown System</u>

The failure of the blowdown system was evaluated and no adverse unanalyzed consequences were identified.

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