DAIRYLAND POWER COOPERATIVE • P.O. BOX 817 • 2615 EAST AV SOUTH • LA CROSSE, WISCONSIN 54601
(608) 788-4000

JUNE 29, 1981

In reply, please
refer to LAC-7633

DOCKET NO. 50-409

Director of Nuclear Reactor Regulation
ATTN: Mr. Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: DAIRYLAND POWER COOPERATIVE
LA CROSSE BOILING WATER REACTOR (LACBWR)
PROVISIONAL OPERATING LICENSE NO. DPR-45
TMI LESSONS LEARNED - SHORT TERM REQUIREMENTS
NUREG-0578, SECTION 2.1.2

References:
(1) NUREG-0578, Section 2.1.2, dated July 1979.
(2) DPC Letter, Linder to Denton, LAC-6680, dated December 6, 1979.
(3) NUREG-0737, dated October 31, 1980.
(4) Evaluation of Fluid Conditions at the Main Steam Safety Valve Inlets for Expected Transients, DPC/NES Report, dated June, 1981.

Gentlemen:

This letter is to present preliminary information regarding the status of the LACBWR program to satisfy the intent of the recommendation provisions of NUREG-0578, Section 2.1.2, entitled "Performance Testing for BWR and PWR Relief and Safety Valves", (Reference 1). The purpose of the NUREG recommendations is to require qualification of relief and safety valves under off-normal reactor overpressure conditions.

At LACBWR there are three Crosby HCU-Spec. 3"xMx6" spring-loaded safety valves installed on a horizontal run of shutdown condenser steam line. They are designed to meet the reactor primary system overpressurization requirements of ASME Nuclear and Boiler Pressure Vessel Codes. The safety valves are located approximately four feet above the reactor vessel steam line outlet and approximately 13 feet above the normal operating unvoided water level.

A046
S 1/1

The three valves, se at 1390 psig and 1426 psig relief pressure (Note: At least one of three must be set at 1390 psig and at least one of three must be set at 1426 psig), are designed to free discharge directly the containment building and are sized to accommodate full desig steam flow of 611,500 lbm/hr at 100% (165 MWth) reactor power. The valves are frequently verified to be operable by bench testing and they have never been required to actuate in 14 years of reactor operation.

Power operated relief valves are not used to control overpressurization transients at LACBWR. In the event of a postulated overpressurization transient coincident with the loss of several safety related equipment functions, the spring-loaded valves would safely relieve steam to the containment building. A general description of the safety valves is contained in Reference (2).

An evaluation has been completed which addresses the fluid conditions that could exist at the safety valves (Reference 4, attached). It was concluded that liquid or two-phase flow conditions would not be present in the safety-valve steam line during any postulated transient condition in which high-pressure relief would be required.

In order to assure that the HCU safety valves would operate satisfactorily even under the most improbable postulated transient conditions, (as discussed in Reference 4), the valves were tested with saturated steam at full-flow ($\sim$ 300,000 lbm/hr), full steam pressure (1405 psia) at the Wyle Laboratory testing facility in Huntsfille, Alabama, in conjunction with Consumers Power Company, which has generically equivalent safety valves at the Big Rock plant. The testing program of other BWR owners (whose operating reactors are not generically equivalent to LACBWR and BRP) was factored into our review and test plan.

The full-flow test program was designed to monitor and record important variables such as valve body temperature, relief pressure, backpressure, accumulation pressure, reset pressure, and valve disc lift distance. (Live steam tests of the LACBWR valves were also satisfactorily conducted on limited flow test facilities in 1977). The full-flow tests were completed on June 26, 1981. The NRC Project Manager for LACBWR was notified prior to the test.

The tests conducted at Wyle Laboratory were witnessed by technical representatives from Wyle Lab, Dairyland Power Cooperative, Consumers Power Company and Crosby Valve Company. The preliminary results of the full-flow tests indicate the valves operated satisfactorily. Minor adjustments were made to the valve nozzle ring and guide ring settings to obtain optimum performance.

A final summary report of the test data will be submitted to you in the near future.

If you have any questions regarding this letter, please let us know.

Very truly yours,

DAIRYLAND POWER COOPERATIVE

Frank Linder

Frank Linder, General Manager

FL:CWA:af
Attachments

cc: NRC Resident Inspectors
J. G. Keppler, Director, NRC-DRO III

EVALUATION OF FLUID CONDITIONS AT THE MAIN STEAM SAFETY VALVE INLETS FOR EXPECTED PLANT TRANSIENTS

I. INTRODUCTION

Section 2.1.2 of NUREG-0578 (TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations) requires qualification testing of safety and relief valves connected to the RCPB's of PWR's and BWR's to verify the capability of these valves to function under all required conditions. As noted in the NUREG, present qualification requirements for these valves include only flow under saturated steam conditions. However, some reactor coolant system transients and accidents as well as alternate core-cooling methods for some reactor designs can result in solid-water on two-phase steam-water flow through these valves. This report assesses the possibility of solid-water and two-phase flow through the safety-valves at the La Crosse Boiling Water Reactor for expected transient and accident conditions.

II. DESCRIPTION OF REACTOR COOLANT SYSTEM OVERPRESSURE PROTECTION DEVICES AT THE LACBWR.

Dry, saturated steam generated in the Reactor Vessel exits the vessel through two 8-inch nozzles which join together at a 10-inch steam-line inside the biological shield. The 10-inch main steam line within the biological shield connects to the shutdown condenser by means of a 10-inch line. The main steam line passes down through the biological shield to below grade level where it is directed out through the containment building shell to the turbine-generator and auxiliary systems. (See Figure 1).

Under most circumstances, rapid surges in primary system pressure will be relieved by steam dump to the main condenser through the Turbine Main Steam Bypass Valve. This automatically controlled valve will be fully open when steam pressure reaches 30 psi over normal system pressure to prevent reactor scram. The turbine bypass valve is capable of bypassing at least 100 percent of rated steam flow.

An overhead, shell-and-tube, "shutdown condenser" is also provided at LACBWR. The function of the shutdown condenser is to provide a backup heat sink/over-pressure relief system for the reactor in the event that the reactor is isolated from the main condenser by closure of either the reactor building (MSIV) or turbine building steam isolation valves.

The shutdown condenser is located on a platform 10 feet above the main floor in the reactor building at Elev. 711'-0". Steam from the 10-inch main steam line passes through a 6-inch line, two parellel inlet steam control valves, back to a 6-inch line and into the tube side where it is condensed by evaporating cooling water on the shell-side. Condensate leaving the tube side passes through two parallel condensate outlet control valves and enters the 8-inch feedwater line in containment.

The steam inlet control valves and condensate outlet control valves are pneumatically operated, requiring air to open. Loss of control air approximately 125 VDC control power to the solenaids cause the valves to open.

The shutdown condenser is automatically placed into operation when any of the following conditions exist:

(1) Reactor Building Steam Isolation Valve Not Fully Open.

(2) Turbine Building Steam Isolation Valve Not Fully Open.

(3) Reactor Pressure Channel 1 above 1325 PSIG.

(4) Reactor Pressure Channel 2 above 1325 PSIG.

(Full scram signals are also generated upon conditions (1), (3) and (4). Condition (2) generates a partial scram signal in addition to a shutdown condenser initiation signal).

Ultimate protection against primary system overpressure is provided by three ASME Code safety valves mounted directly on the 10-inch line leading to the shutdown condenser. At least one is set to open at 1390 PSIG with a flow capacity of approximately 290,000 lb/hr. At least one is set at 1426 PSIG with a flow capacity of approximately 302,000 lb/hr. Discharge of the safeties is directed through a short 10-inch vertical vent pipe directly to the containment atmosphere.

Power Operated relief valves are not utilized at the LACBWR to control over-pressurization transients. There are no alternate shutdown cooling modes which require liquid flow through primary system safety or relief valves at reduced pressure.

III. ASSESSMENT OF LIKLIHOOD OF TWO-PHASE OR LIQUID FLOW THROUGH THE SAFETY-VALVES FOR EXPECTED TRANSIENT SEQUENCES AT LACBWR.

For the safety valves at LACBWR to pass two-phase or liquid flow, the following conditions must be met during a transient event or accident:

(a) Liquid or a two-phase mixture must pass into the main steam lines. Furthermore, the liquid must accumulate under, or two-phase mixture must flow by the safety valves.

(b) Coincidentally over-pressurization of the main steam line up to the valve setpoint must occur.

The following is a review of LACBWR plant transient scenarios which proceed in directions which could conceivably result in the fulfillment of conditions (a) or (b) or both.

The full range of possible plant transients has been extensively analyzed and studied for LACBWR (References 1, 2 and 3). Per Reference (1), anticipated operational transients at the LACBWR plant can be classified into seven categories as follows:

(1) Reactor pressure increase

(2) Moderator temperature decrease

(3) Reactor vessel coolant inventory decrease.

(4) Reactor pressure decrease.

(5) Core coolant flow decrease (recirculation flow).

(6) Core coolant flow increase (recirculation flow).

(7) Positive reactivity insertion.

The following categories can be eliminated from consideration because they do not meet the criteria (a) and (b) above.

(3) Reactor vessel coolant inventory decrease.

(5) Core coolant flow decrease.

(6) Core coolant flow increase.

(7) Positive reactivity insertion.

The remaining categories will now be examined on a case-by-case basis.

Reactor Pressure Increase

Incidents in this category include generator loss of load, turbine trip, IPR failure (increasing pressure) and MSIV closure. For all transients, reactor pressure increase is arrested well below the lowest safety valve setpoint either by automatic Turbine Main Steam Bypass Valve actuation or shutdown condenser initiation. (Refs. 1 and 2). Even if multiple failures were to occur, thus forcing safety valve lifting, the safeties would not be subject to two phase or liquid fluid conditions for this type of transient. The reason is that the sudden increase in reactor pressure associated with these incidents would collapse voids in the reactor vessel, lowering the two-phase RV level. Liquid or low-quality steam carryover into the main steam lines is not a liklihood under these conditions. Neither condition (a) nor (b) above is fulfilled for these transients.

Moderate Temperature Decrease

This category comprises the following incidents: uncontrolled, increase in feedwater flow, loss of all feedwater heaters, liquid poison injection or ECCS operation.
Only the feedwater flow increase transient and ECCS initiation transient are of concern since they result in inadvertant, and initially uncontrolled filling of the reactor vessel.

Under the assumption that a failure in the feedwater control system occurs such that feedwater flow increases at the maximum rate to the maximum available for both feedwater pumps (Refs. 1 and 2), a reactor scram on overpower will occur after a time delay of approximately 9 seconds. In addition, high reactor water level alarms will be annunciated in the main control room, alerting the operators to the probable cause of scram (verified by observation of steam/feedwater flow mismatch). Plant operators then take manual control of the feedwater control system, tripping the feedwater pumps if necessary, to return level to its normal range.

If feedwater flow is not brought under control within 1½ to 2 minutes, some water can be expected to enter the steam lines. Even if this were to occur, safety-valve operation is not required, since the shutdown condenser has ample decay heat removal capacity to preclude system pressurization to the safety-valve setpoints. (Refs. 1 and 2). Furthermore, any moisture reaching RV main steam nozzles would follow the normal main steam flow path and would drain toward the turbine. Again, neither of the adverse conditions (a) or (b) are satisfied.

Reactor Pressure Decrease

Transients in this category include accidental opening of the Turbine Main Steam Bypass Valve, IPR failure (decreasing pressure). This type of transient can be potentially troublesome for some Boiling Water Reactor designs from the standpoint of fluid conditions at the safety valve inlets. This is so because a reduction in main steam pressure causes an increased void content in the reactor vessel, causing an increase in the two-phase reactor inventory level. The resultant two-phase level may increase to the point where two-phase flow enters the main steam lines. Subsequent repressurization due to safety system actuation (e.g., MSIV closure) could lead to safety-valve opening with two-phase flow conditions. At LACBWR, however, transient analyses (Refs. 1 and 2) have shown that the safety-valve setpoints are not reached for these incidents. Furthermore, main steam line geometry at La Crosse is such that liquid would not accumulate at the SV inlet (see Figure 1). Therefore, two-phase or liquid flow through the safeties at the LACBWR is of extremely low probability for this class of transient.

Conclusion

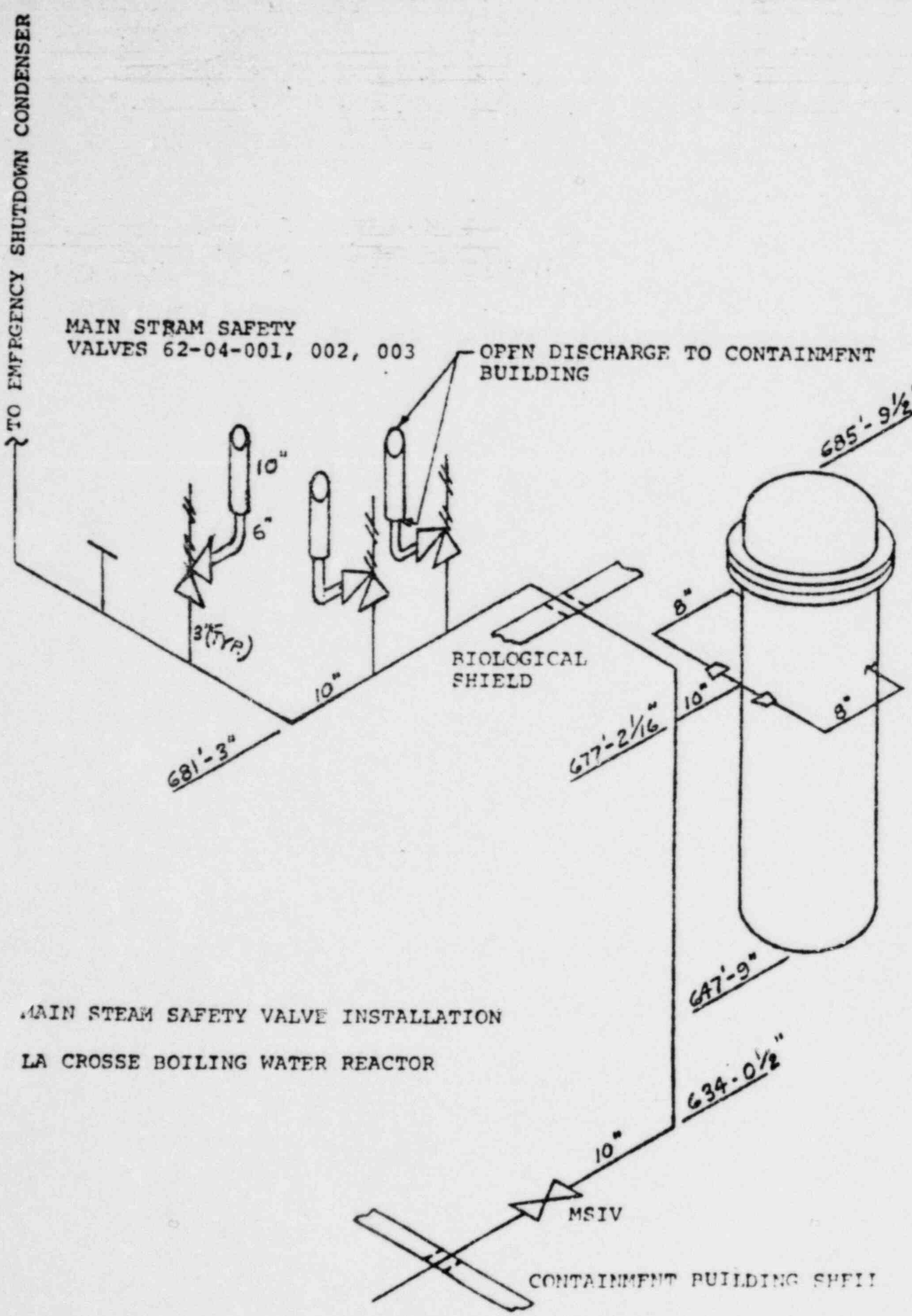
There are no expected transient or accident sequences at the LACBWR that can be expected to result in operation of the safety valves in a two-phase or liquid, flow mode. This is primarily due to the presence of a highly reliable shutdown condenser system, with fail-safe, redundant inlet and outlet control valves, which is more than adequate to dissipate the reactor decay heat and maintain reactor pressure below the safety valve setpoints for all transient conditions. Therefore, there is no justification or basis for prototypical testing of a safety valve installation to qualify it for liquid or two-phase flow. However, since the LACBWR valves were not previously tested on a high steam flow capacity system, saturated steam operability tests were scheduled and were conducted at the Wyle Laboratory high flow facility. The completion date of the test was June 26, 1961.

REFERENCES

(1) NES Report No. 81A0037 Rev.0,
"LA CROSSE BOILING WATER REACTOR - Review of Plant Transients"
July, 1980

(2) NES Report No. 81A0025,
"RESPONSE TO QUESTION 4 - TRANSIENT ANALYSIS FOR LACBWR RELOAD FUEL"
February 18, 1977

(3) Gulf Nuclear Fuels, Co., Report SS-478
"ANTICIPATED TRANSIENTS WITHOUT SCRAM AT THE LA CROSSE BOILING WATER REACTOR"
February 28, 1974



MAIN STEAM SAFETY VALVE INSTALLATION

LA CROSSE BOILING WATER REACTOR

FIGURE 1