Consumers Power Company

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October 9, 1979

Director, Nuclear Reactor Regulation
Att: Mr Dennis L Ziemann, Chief
Operating Reactors Branch No 2
US Nuclear Regulatory Commission
Washington, DC 20555

DOCKET 50-155 - LICENSE DPR-6 -
BIG ROCK POINT PLANT - RESPONSE
TO QUESTIONS REGARDING INTERACTION
BETWEEN NON-SAFETY GRADE AND SAFETY
GRADE SYSTEMS

NRC Office of Inspection and Enforcement Information Notice 79-22, issued September 14, 1979, discussed a concern involving potential effects of non-safety grade equipment on safety analyses and performance of safety grade equipment. This concern related to the effect on non-safety grade equipment of an adverse environment which might be produced by failure of a high energy line. Consumers Power Company was requested by NRC letter dated September 17, 1979, to evaluate the concerns discussed in I&E Information Notice 79-22 as they apply to the Big Rock Point Plant. Consumers Power Company was specifically requested to consider whether an unreviewed safety concern could exist and to provide information to enable the NRC Staff to determine whether modification of License DPR-6 was required.

Consumers Power Company has evaluated the effect that adverse environments which might be created by a high energy line break might have on non-safety grade control equipment at Big Rock Point. The results of this evaluation are reported in the attachment to this letter as a matrix of possible effects of non-safety grade equipment failures and explanatory notes. For purposes of preparing the attached matrix of possible adverse effects, each non-safety grade system was arbitrarily assumed to fail with the identified failure mode regardless of design features which might prevent such a failure. The explanatory notes describe how each evaluated failure is bounded within the existing licensing bases for Big Rock Point, regardless of its probability of occurrence, or describe why the failure mode is considered unlikely.

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The attached evaluation concludes that no modification of License DPR-6 is needed as a result of the concerns discussed in I&E Information Notice 79-22.

David A Bixel (Signed)

David A Bixel
Nuclear Licensing Administrator

CC: JGKeppler, USNRC

EVALUATION OF CONCERNS DISCUSSED IN I&E INFORMATION NOTICE 79-22
BIG ROCK POINT

The effects of the failure of non-safety grade systems on safety grade systems, and the resulting impact on the accident analysis are summarized in Table 1. An explanation of each item listed on the Table is given as a series of footnotes, each corresponding to the numbers found in Table 1.

NAE = No Adverse Effect
PAE = Possible Adverse Effect

TABLE 1

Control System	Failure Mode	Recirc. Line Break	Steam Line Break	FW Line Break
Initial Pressure Regulator	a) Signals for Increased Flow (a decreased pressure) i.e. admission valves open	NAE (1)	NAE (1)	NAE (1)
	b) Signals for Decreased Flow (or increased pressure) i.e. admission valves close	NAE (1)	NAE (1)	NAE (1)
Bypass Valve Control	a) Valve Fails Open	NAE (1)	NAE (1)	NAE (1)
	b) Valve Fails Closed	NAE (1)	NAE (1)	NAE (1)
Recirc Pump Control	a) Pumps Continue to Run	NAE (2)	NAE (2)	NAE (2)
	b) Pumps Trip	NAE (2)	NAE (2)	NAE (2)
Recirc Pump Valves	a) Signaled to Open	NAE (3)	NAE (3)	NAE (3)
	b) Signaled to Close	PAE (3)	PAE (3)	PAE (3)
FW Control System	a) Signaled to Decrease	NAE (4)	NAE (4)	NAE (4)
	b) Signaled to Increase	NAE (4)	NAE (4)	NAE (4)
	c) Loss of FW Heaters	NAE (4)	NAE (4)	NAE (4)
Control Rod Drive	a) Sig. to Withdraw	PAE (5)	PAE (5)	PAE (5)
	b) Sig. to Insert	NAE (5)	NAE (5)	NAE (5)
Rx Vessel Head Vent	a) Sig. to Open	NAE (6)	NAE (6)	NAE (6)
	b) Sig. to Close	NAE (6)	NAE (6)	NAE (6)

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NAE = No Adverse Effect
PAE = Possible Adverse Effect

TABLE 1 (Continued)

Control System	Failure Mode	Recirc. Line Break	Steam Line Break	FW Line Break
Emergency Condenser	a) Sig. to Activate	NAE (7)	NAE (7)	NAE (7)
	b) Sig. to Isolate	NAE (7)	NAE (7)	NAE (7)

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1. <u>High Energy Line Break with Failure of IPR or Bypass Valve Control</u>

Only high energy line breaks outside[1] of containment can affect the IPR or bypass valve control system. These breaks range from the main steam line break, (MSLB) in which complete severence of the main steam line is considered, to a feedwater line break, which is in effect a loss of feedwater transient.

Failure of the IPR (which controls the position of the turbine admission valves) or the bypass valve in the open position would result in increased steam flow from the primary system. Steam flow from the system is limited on the high end by the flow area of the main steam line. Since the bypass valve and admission valves are fed by the main steam line, opening of one or both cannot result in a greater steam flow from the primary system than that found in the main steam line break analysis. This event is therefore bounded by the MSLB analysis. Failure of the IPR in the closed position in the event of a break outside containment with coincident failure of the bypass valve results in the same sequence of events as the turbine trip without bypass. Reactor would scram on a high flux signal.

A feedwater line break outside of containment has the same effect on the primary system as a loss of feedwater event. Primary coolant is not discharged from the break because the feedwater check valves are located inside containment. The steam line break analyses all assume that loss of feedwater occurs coincident with the break. The failure of the IPR and/or bypass valve in the open position as a result of a feedwater line break outside of containment is bounded by the steam line break analyses. The size of the "break" in this case would be the flow area of the admission valve and the bypass valve. Failure of the IPR and bypass valves in the closed position is effectively a loss of feedwater event without bypass. In this case, the emergency condenser would limit primary system pressure, and begin system cooldown.

2. <u>High Energy Line Break with Failure of Recirculation Pump Control</u>

Only high energy line breaks inside of containment can affect the recirculation pump control system. Since the Big Rock Point recirculation pumps are constant speed pumps (variable speed control system does not exist), the pumps can only fail in the running condition, or in the tripped condition. All safety analyses assume recirculation pump trip coincident with a break, as this is considered the worst case. The continuation of one or both recirculation pumps to run during an accident would result in higher mass flow through the core and therefore a greater thermal margin than is calculated in the accident analysis. No adverse effect on a high energy line break is produced by failure of the recirculation pump control system.

3. High Energy Line Break with Failure of Recirculation Pump Valves

Only high energy line breaks inside of containment can affect the recirculation pump valves. The pump discharge and suction valves are assumed to remain open in all of the safety analyses. No adverse effect is produced if the valves fail in this position.

Failure of the valves in the closed position such that the recirculation loops were isolated could have an adverse effect on accident mitigation. However, it is considered highly improbable that this situation can occur due to the configuration of the Philadelphia Limitorque Motor Operator. This operator provides a heavy duty industrial grade drip proof boundary. To close the valves, a failure would have to involve a simultaneous, low resistance short on all three phases of the 480 V motor starter contacts or a hot wire-to-wire 120 V short sufficient to energize the coil of the "close" contactor. The possibility of control circuit contact shorts causing failure will be precluded by placing the control room hand switch for each of these valves in the pull-to-stop position. Consumers Power Company intends to further evaluate the environmental qualification of the motor starters associated with these valve operators.

4. High Energy Line Break with Failure of the Feedwater Control System

High energy line breaks inside and outside of containment can affect the feedwater control system. The failure can take the form of decreasing feedwater flow, increasing feedwater flow, or decreasing feedwater heating. In all of the accident analyses feedwater is assumed to be lost coincident with the break and therefore no adverse effect exists for this case. Emergency cooling water would normally be supplied from an outside water source (the fire system) at Big Rock Point. Therefore, failure of the feedwater system such that it continued to supply water to the reactor for large breaks has the same effect as the ECCS, with the exception that recirculation core spray systems may have to be actuated sooner. No adverse effect on accident mitigation is produced. Failure of the FW system in this manner for small breaks such that the FW system would override break flow is an event which could also occur during non-accident conditions. Operator action would be required in both cases to prevent the steamline from flooding, but no adverse effect would be produced on mitigation of the small break consequences.

Loss of FW heating as a result of a high energy line break would increase the rate of depressurizing and cooling down the system. This would be a negligible effect for the large break, and would revert to a loss of FW heaters event for a small break. In the case of the small break, a scram may occur on high flux rather than low drum level. In either case, no adverse effect on accident mitigation is produced.

5. High Energy Line Break with Failure of Control Rod Drive System

Failures of the control rod drive system which would cause rods to insert or withdraw would require failure of the control solenoids. Early insertion of control rods as a result of a system failure would have a desirable effect for all breaks. A failure which caused one or more rods to withdraw could have an adverse effect, but such a failure is considered incredible. The control rod drive system is designed such that withdrawal of a control rod requires a number of control solenoid operations, each of which must occur in the proper sequence. The possibility of failures occurring which exactly duplicate this sequence of operations is considered incredible.

6. High Energy Line Break with Failure of Reactor Vessel Head Vent Valve

Only high energy line breaks inside containment can affect the reactor vessel head vent valve. The vent valve is a 1 1/2" motor-operated valve which vents to the steam drum. The valve is normally closed but has a 3/4" bypass around it which is always open.

Failure of the valve in the open position as a result of a high energy line break would produce no adverse effect because of the valve's small size and because the valve would vent water to the primary system and no loss of primary system inventory would occur.

7. High Energy Line Break with Failure of the Emergency Condenser

Only high energy line breaks inside containment can cause failure of the emergency condenser. The emergency condenser is not required to operate for any breaks considered in the LOCA analyses and therefore failure in the isolated mode does not produce a more adverse effect than analyzed. Operation of the emergency condenser at the time of a break inside containment would increase the rate of depressurization and cooldown of the system. For the case of a large break, this would be an insignificant contribution to depressurization. In the case of a small break, depressurization would occur sooner than current LOCA analysis predicts, permitting earlier core spray initiation. Therefore, no adverse effect occurs because of emergency condenser failure.

Conclusion

Based on the considerations discussed above, it is concluded that the concerns identified in Inspection and Enforcement Information Notice 79-22 are addressed within the existing Big Rock Point accident analyses. No modifications to License DPR-6 are required.

CONSUMERS POWER COMPANY

By R B DeWitt (Signed)
R B DeWitt, Vice President
Nuclear Operations

Sworn and subscribed to me on this 9th day of October, 1979.

Dorothy H Bartkus (Signed)
Dorothy H Bartkus
Jackson County

My commission expires March 26, 1983.

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(1) Locations in the plant where an adverse environment could exist were identified through review of a report on pipe breaks outside of containment. (Effect of Compartment Pressurization Due to Pipe System Break Outside Containment - Bechtel, April, 1973). Equipment not located in the areas specified in the report were not considered to be affected by the break.

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