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September 23, 1982

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

Subject: Zion Station Unit 1
Proposed Change to Facility
Operating License No. DPR-39
NRC Docket No. 50-295

Dear Mr. Denton:

Commonwealth Edison hereby requests a temporary change to Facility Operating License No. DPR-39. A change is requested to Section 3.8.1.c of the Technical Specifications to add the following note on page 165:

> "For Unit 1 during the period of 1440 hours, September 26 through 1440 hours, September 29, 1982, reactor operation is allowed with 1B charging pump inoperable, provided that the 1A charging pump system is operating, and both safety injection pump systems and both residual heat removal systems are operable."

This change is requested on an emergency basis, because the 7 day outage period permitted by the Technical Specifications will expire at 1440 hours on September 26.

Charging Pump Failure Event

At 1440 hours on September 19, 1982 with 1B charging pump running, the pump's auxiliary lube oil pump auto-started and the operator noticed the pump's running current was high. The flow from the pump went to zero and the pump was shutdown and declared inoperable.

Subsequent internal inspection of the 1B charging pump has revealed that the pump shaft broke between the 10th and 11th stage at the split ring area. The entire pump barrel needs to be replaced.

To ensure that repair of the charging pump will be accomplished in an expeditious manner, around-the-clock maintenance coverage (two 10-hour shifts) will be provided. Post-repair testing will ensure pump operability.

Allowing an additional 3 days outage time to accomplish the repairs using safe and proper procedures will avoid an unnecessary unit shutdown and the resultant thermal cycling involved in taking the unit from power operation to hot shutdown, and then from hot shutdown to cold shutdown.

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Basis for OOS Time

The presently allowable repair period of seven days as stated in the Technical Specification bases for Section 3.8 for a charging pump are based on the following three considerations:

1) Assuring with high reliability that the safeguard systems will function properly if required to do so.

2) Allowances of sufficient time to effect repairs using safe and proper procedures.

3) The degree of functional redundancy among safeguards systems over the range of break sizes.

FSAR Accident Analysis Concerns

With all other ECCS pumps operable (as well as their standby AC and DC power supplies), more than the required degree of functional redundancy of safeguards systems over a range of break sizes will be met. Analyses in the FSAR assume the operability of only one charging pump, one safety injection pump, and one RHR pump.

The ability of a charging pump to function as designed makes a significant contribution to the mitigation of the following events:

1) Small break LOCA

2) Design basis main steamline break

3) Small (less than Design Basis) Steam Generator Tube Rupture

Zion Station's current small LOCA analysis bounds the present plant configuration of 1B charging pump out-of-service and no single failure. With the worst single failure assumption of the loss of the operable 1A charging pump train (loss of Bus 149), ECCS flow can still be delivered from two SI pump trains and one RHR pump train.

Westinghouse has evaluated the effect of a small break LOCA at Zion Station with no charging pumps available and ECCS flows being delivered from two SI pumps and one RHR pump.

With no charging pumps available, there is no ECCS flow at pressures greater than 1500 psia. The Westinghouse Topical Report, WCAP-9600, "Report on Small Break LOCA's in W NSSS Systems", has demonstrated that the RCS depressurizes to a pressure where ECCS flow equilibriates to the break flow for design basis small LOCA's with auxiliary feedwater available. Therefore, for small break sizes that relied only on charging flow for inventory makeup, additional depressurization will naturally occur, resulting in makeup from the SI pump head. ECCS flow above 1500 psia is not required for core cooling of any size design basis LOCA's. Further, the absence of ECCS flow at RCS pressures above approximately 1500 psia would have no affect on the peak clad temperature (PCT) for the worst small break, as analyzed in the FSAR.

ECCS flow delivered in the range of RCS pressure from 600 to 1200 psia does influence the small break calculated PCT. Two SI pumps would deliver approximately 25% less flow in this pressure range than would one SI plus one charging pump, as assumed in the FSAR analysis. Established sensitivity studies applicable to Zion have indicated that such a degradation would result in as much as a 250°F small LOCA PCT increase.

Zion Station's small break analysis does not use the latest NRC approved Westinghouse small LOCA Evaluation Model. The small LOCA analysis in the Zion FSAR has utilized the March, 1974 small LOCA version, and resulted in a PCT of 1747°F. A reanalysis with the latest approved model (October, 1975 version) would result in a maximum PCT of approximately 1200°F. The latest model is based on a 4-loop plant analyzed in WCAP-8970-P.A., "Westinghouse Emergency Core Cooling System Small Break October, 1975 Model" and applies to the Zion Unit. Therefore, the expected small LOCA PCT assuming loss of both charging pumps and the latest small break model is bounded by current analysis, and remains well below 10CFR 50.46 limits.

Additionally, credit for conservative assumptions in the small LOCA FSAR analysis can mitigate the PCT penalty. Following is a summary of some of those assumptions, and estimates of their impact on PCT.

1. ANS Decay Heat + 20% - A best estimate decay heat function would reduce PCT by 200°F.

2. Small LOCA analysis assumed peaking factor of 2.32 - Large break limited maximum F_Q of 2.17 would reduce small break PCT by 100°F.

3. Analysis assumed loss of steam dump - steam dump availability would reduce PCT by 100°F or more.

4. Degraded SI pump performance - best estimate performance would reduce PCT by 50°F.

5. Other probable assumptions, such as operator initiated cooldown, would also mitigate consequences.

Since Unit 1 is near BOL (approximately 1/8 the way thru Cycle 7) the consequences in the unlikely event of a Design Basis Main Steamline Break are greatly reduced from that assumed in the worst case FSAR analyses due to the much less negative moderator temperature coefficient at this time in core life compared to EOL.

Compensatory Actions

To ensure the high reliability of the safeguard system, the following compensatory actions will be taken during the period the 1B charging pump is out of service:

1) The other safeguards charging pump (1A charging pump) will be running continuously to ensure pump operability at all times.

2) The remaining ECCS pumps, including their associated standby AC and DC power supplies, will be demonstrated to be operable daily, thus ensuring they are capable of providing sufficient injection flow consistent with FSAR assumptions.

To provide assurance to the extent practical that the probability of occurrence of the accidents analyzed in the FSAR will be minimized, the following additional measures will be taken while 1B charging pump is out of service:

1) RCS leakage surveillance will be increased. The performance of PT-21 (Reactor Coolant System Leakage Surveillance) will be required shiftly instead of daily. PT-21 monitors for RCS leakage by a variety of diverse methods including:

   a. a quantitative calculation of RCS leakage

   b. monitoring of containment area particulate and gas radiation monitors

   c. monitoring of containment temperature, pressure, and humidity

   d. monitoring of containment sump water accumulation

   e. the Reactor Vessel Head Flange leak detection system

   f. the Reactor Vessel leak detection system

   In addition, shiftly monitoring of Reactor Coolant Drain Tank and Pressurizer Relief Tank input will be required. This increased surveillance will provide additional assurance of RCS integrity.

2) A surveillance will be conducted daily to provide additional assurance of main steam system integrity. In addition to the normal daily inspection inside containment for observable leaks, the main steam system outside containment will be inspected daily. Water accumulation in the containment sump will be trended.

3) A recent steam generator tube leak rate test performed on all four Unit 1 steam generators on September 17, 1982 confirmed zero leakage. To increase our confidence in continued steam generator tube integrity, the following additional interim measures will be taken:

1. Steam generator activity levels will be monitored shiftly.

2. Steam generator blowdown radiation monitor 1RE-0019, and condenser off-gas radiation monitor 1RE-0015, will be trended shiftly.

If any significant findings are observed in any of the above interim surveillance, the results will be evaluated. If the evaluation shows a significant decrease in system integrity, then a unit shutdown shall be commenced immediately.

The required fee for this Amenoment will be submitted at the earliest opportunity.

Please address questions regarding this matter to this office.

Very truly yours,

F. G. Lentine

F. G. Lentine
Nuclear Licensing Administrator

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cc: Zion Resident Inspector

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