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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

LaSalle County Station, Units 1 and 2
Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Subject: Response to Request for Additional Information Concerning the Conversion to Improved Standard Technical Specifications (ITS)

Reference: Letter from R. M. Krich (ComEd) to U. S. NRC Document Control Desk, "Request for Technical Specifications Changes for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2, to Convert to Improved Standard Technical Specifications," dated March 3, 2000

Commonwealth Edison (ComEd)) Company, currently Exelon Generation Company (EGC), LLC, in a letter dated March 3, 2000, proposed changes to the Technical Specifications (TS) of Facility Operating License Nos. DPR-19, DPR-25, NPF-11, NPF-18, DPR-29, and DPR-30 for Dresden Nuclear Power Station, Units 2 and 3, LaSalle County Station, Units 1 and 2, and Quad Cities Nuclear Power Station, Units 1 and 2.

In discussions, the NRC requested that additional information be provided concerning the proposed LaSalle County Station Improved Technical Specifications (ITS) Section 3.5.1, "ECCS [Emergency Core Cooling System] Operating." Specifically, the NRC asked EGC to provide additional technical justification for a proposed change to the value of the Reactor Coolant System (RCS) pressure from 122 psig to 150 psig when establishing operability requirements for the Automatic Depressurization System (ADS) Valves.

A001

The requested additional information is provided in the Attachment to this letter. Should you have any questions concerning this letter, please contact Mr. J. V. Sipek at (630) 663-3741.

Respectfully,

R. M. Krich
Director - Licensing
Mid-West Regional Operating Group

Attachment: Response to Request for Additional Information

cc: Regional Administrator - NRC Region III
NRC Senior Resident Inspector - LaSalle County Station
Office of Nuclear Facility Safety - Illinois Department of Nuclear Safety

ATTACHMENT

LaSalle County Station, Units 1 and 2
Response to Request for Additional Information

**Response to NRC Question Regarding
Automatic Depressurization System Operability
At 150 psig for LaSalle County Station**

Background

The proposed change to increase the value of the Reactor Coolant System (RCS) pressure from 122 psig to 150 psig at which to require the Automatic Depressurization System (ADS) function of six safety relief valves to be OPERABLE is described in "Enclosure B" of the referenced letter. Specifically, the proposed change is described in Less Restrictive Technical Change # L.1 to Improved Technical Specifications (ITS) Section 3.5.1 "ECCS [Emergency Core Cooling System] Operating."

The RCS pressure at which ADS is required to be OPERABLE, as specified in the Current Technical Specifications Section 3.5.1 "ECCS Operating APPLICABILITY and ACTIONS e.1 and e.2 is increased from 122 psig in ITS Section 3.5.1 to 150 psig to provide consistency of OPERABILITY requirements for all ECCS and Reactor Core Isolation Cooling (RCIC) systems equipment. The consequences of small break loss of coolant accidents (LOCAs) at low pressure (i.e., between 122 psig and 150 psig) are bounded by analyses performed at higher pressures.

Discussion

Successful core cooling for small line breaks (i.e. small break LOCAs) is depicted in Updated Safety Analysis Report (UFSAR) Chapter 7, Figure 7.3-2. The depressurization phase is mitigated by the High Pressure Injection (HPCS) system, ADS A, or ADS B.

Two reactor vessel low water level trip settings are used to initiate the ECCS. The first low water level setting, which is the higher of the two, initiates the HPCS. The second low water level setting, which is lower, initiates the Low Pressure Coolant Injection (LPCI) system, Low pressure Core Spray (LPCS) system, and ADS. This setting also closes the main steamline isolation valves.

The ADS automatic relief valves are installed on the main steamlines inside the primary containment drywell. The valves can be actuated in three ways; they will relieve pressure by pressure switches, or by mechanical actuation on high reactor pressure, or by actuation of ADS trip logic via an electric-pneumatic control system. Relief valve operation may be controlled manually from the main control room to hold the desired reactor vessel pressure. The depressurization by automatic blowdown is intended to reduce reactor vessel pressure during a LOCA.

Two ADS trip sub-systems (i.e., channels) are provided, as described in UFSAR Figure 7.3-6, ADS A and ADS B. Division 1 sensors for low reactor vessel water level and high drywell pressure initiate ADS A, and Division 2 sensors initiate ADS B. The high drywell pressure signal can be automatically bypassed as discussed in UFSAR Section 7.3.1.2.2.3.

The intended function of the ADS is to assist in depressurizing the reactor vessel so that the low pressure Emergency Core Cooling System (ECCS) pumps (e.g., LPCI and LPCS Systems) can deliver adequate flow into the reactor vessel at an earlier time in the accident sequence. This is especially true for small break LOCAs where the reactor vessel will be at a high pressure for a longer period of time. In the event of a LOCA, the HPCS, LPCS and LPCI systems are initiated either by the reactor vessel low-low water level (LLWL) signals or the low-low-low water level (LLLWL). At approximately 46 sec after LLWL signal, the HPCS injection valves powered by the Emergency Diesel Generator (EDG) are fully opened. The LPCS and LPCI injection valves are powered and fully opened 60 sec after LLLWL signal. Although the valves may be fully opened, low pressure ECCS flow will not start until the reactor vessel has depressurized below the shutoff head of the pumps. The ADS valves will open and depressurize the reactor vessel when the delay timer times out in 120 seconds after LLLWL signal.

In the event of a small break LOCA occurring when the reactor vessel is initially at a pressure below the shutoff head of the pumps, the ECCS may be capable of delivering sufficient flow to the reactor vessel without the assistance of the ADS valves. The following discussion demonstrates that ADS operability is not required when the reactor vessel is operating at or below 150 psig. It demonstrates that at the time when ADS is required to open, the ECCS is already delivering sufficient flow to the core to make up for water inventory loss in blowdown and to recover reactor vessel water level.

At the time when ADS relief valves are required to open, HPCS has already been injecting into the core for 74 seconds plus the time between receipt of the LLWL signal and the LLLWL signal and LPCS and LPCI for 60 seconds. The water level inside the reactor vessel may be above or below the jet pump suction level.

Case 1: Water Level above Jet Pump Suction Level

If the water level inside the reactor core shroud is above the jet pump suction level, the blowdown is saturated liquid and the release rate may be larger than the ECCS flow rate. Water level will continue to decrease and potentially expose the fuel cladding. The potentially exposed part of fuel cladding is protected by ECCS flow from above (e.g., HPCS System) and by steam cooling from below. As the reactor vessel continues to depressurize, the ECCS flow will become greater than the break flow and the water level will recover.

If ECCS flow does not become greater than the break flow, the water level will continue to drop and eventually will drop below the jet pump suction elevation. The consequence of this scenario is the same as if the water level is below the jet pump suction level at the time when ADS relief valves are required to open. The consequence is consistent with Case 2 below.

Case 2: Water Level below Jet Pump Suction Level

If the water level inside the core shroud is below the jet pump suction level at the time when ADS is required to open, the reactor vessel blowdown will be the inventory from

the core and the inventory in the downcomer region. The inventory loss from the core is due to steam generation by decay heat and flashing and it can not exceed the maximum flow through the break. The maximum flow through the limiting small break of 1.1 sq ft, determined from the Moody critical flow model, is approximately 376 lb/sec for an RCS pressure of 150 psig.

The available ECCS flow is conservatively determined using the pump flows at a reactor vessel pressure of 150 psig. This is conservative because the reactor vessel pressure is expected to be below 150 psig and the pump flow rates will be higher. The table below shows that single failure of EDG that powers LPCS pump will result in the least amount of ECCS flow at 150 psig, approximately 8726 gpm. Assuming the ECCS water is at 170° F, the mass flow rate is 1181 lb/sec.

Assumed Failure	ECCS Available	Flow Rate at 150 psig, gpm	Total Flow Rate at 150 psig, gpm (lb/sec)
LPCS/EDG	2 LPCI 1 HPCS	3,326 5,400	8,726 (1,181)
LPCI/EDG	1 LPCI 1 LPCS 1 HPCS	1,663 4,421 5,400	11,484 (1,555)
HPCS/EDG	3 LPCI 1 LPCS	4,989 4,421	9,410 (1,274)

Since the inventory loss from the core (i.e., 376 lb/sec) is less than the worst case ECCS flowrate (i.e., 1,181 lb/sec), water level will be increasing. The ADS is no longer needed to depressurize the reactor vessel so that more ECCS flow can be obtained. If ADS is not available, the time to reflood will be slightly later and the fuel peak cladding temperature (PCT) will be slightly higher. However, the reactor is initially at low power and the PCT will be bounded by the PCT of the limiting small break starting at rated power.

Therefore, revising the minimum required pressure for ADS operability to 150psig is acceptable.