ATTACHMENT B, Proposed Change to Technical Specification for Quad Cities Nuclear Power Station Units 1 and 2, Page 1 of 2

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3.7 - LIMITING CONDITIONS FOR OPERATION

2. With one or more reactor instrumentation line excess flow check valves inoperable, operation may continue and the provisions of Specification 3.0.C are not applicable, provided that within 4 hours either:

   a. The inoperable valve is restored to OPERABLE status, or

   b. The instrument line is isolated and the associated instrument is declared inoperable.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

a. At least once per 31 days by verifying the continuity of the explosive charge.

b. At least once per 18 months by removing at least one explosive squib from an explosive valve such that each explosive squib will be tested at least once per 90 months, and initiating the removed explosive squib(s). The replacement charge for the exploded squib(s) shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life or operating life, as applicable.

6. At the frequency specified by the Primary Containment Leakage Rate Testing Program, verify leakage for any one main steam line isolation valve when tested at P_t (25 psig) is $\leq$11.5 scfh.

insert

leakage tests). The acceptance criteria were established during initial air lock and primary containment OPERABILITY testing. The periodic testing requirements verify that air lock leakage does not exceed the allowed fraction of the overall primary containment leakage rate. The Frequency is required by the Primary Containment Leakage Rate Testing Program. The surveillance requirements have been annotated such that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Additional annotation is provided to require the results of air lock leakage tests being evaluated against the acceptance criteria applicable to the surveillance requirements. This ensures that the air lock leakage is properly accounted for in determining the combined Type B and Type C primary containment leakage.

3/4.7.D Primary Containment Isolation Valves

The OPERABILITY of the primary containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The containment is also penetrated by a large number of small diameter instrument lines which contact the primary coolant system. A program for periodic testing and examination of the flow check valves in these lines is performed by blowing down the instrument line during an inservice leak or hydrostatic test and observing conditions which verify that the flow check valve is operable, e.g., a distinctive 'click' when the poppet valve seats, or an instrumentation high flow that quickly reduces to a slight trickle.

The main steam line isolation valves are tested at lower pressures, per an approved exemption, but the leakage rate is included in the Type B and C test totals. The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10CFR Part 50 with the exception of approved exemptions. (Ref: Exemption Request Approval, Mr. D. B. Vassallo (NRC) to Mr. D. L. Farrar (CECo) dated June 12, 1984.)

3/4.7.E Suppression Chamber - Drywell Vacuum Breakers ← Bases INSERT

The function of the suppression chamber to drywell vacuum breakers is to relieve vacuum in the drywell. These internal vacuum breakers allow air and steam flow from the suppression chamber to the drywell when the drywell is at a negative pressure with respect to the suppression chamber. Each vacuum breaker is a self-actuating valve, similar to a check valve.

The safety analysis assumes that the internal vacuum breakers are closed initially and are fully open at a differential pressure of 0.5 psid. Additionally, three of these internal vacuum breakers

INSERT IN PLACE OF THE CURRENT 4.7.D.6:

6. In accordance with the methods and at the frequency specified by the Primary Containment Leakage Rate Testing Program, verify total maximum pathway leakage for all main steam isolation valves (MSIVs) is ≤ 46 scfh when tested at P_t (25 psig).

INSERT AT THE END OF BASES SECTION 3/4.7.D:

The individual main steam isolation valve (MSIV) leakage limit has been replaced by the aggregate leakage limit of ≤ 46 scfh for all MSIVs. The leakage will be determined for the maximum pathway leakage in accordance with the Primary Containment Leakage Rate Testing Program. This is a very conservative total for MSIV leakage because it takes the MSIV with the maximum leakage in each steam line and sums the leakage for each of those valves to determine the maximum pathway leakage.

ComEd has evaluated this proposed amendment for Quad Cities Nuclear Power Station, Units 1 and 2 and determined that it involves no significant hazards consideration. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

Involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated;

Create the possibility of a new or different kind of accident from any previously analyzed; or

Involve a significant reduction in a margin of safety.

ComEd proposes to amend Appendix A, Technical Specification, of Facility Operating Licenses DRP-29, DPR-30. The determination that the criteria set forth in 10 CFR 50.92 are met for this amendment request is indicated below:

Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed changes to the Technical Specifications, Appendix A, modifies the allowed leakage limit to an aggregate value with no change to the total allowed leakage rate. This change does not affect either the automatic or manual features that would close the MSIVs. There are no physical changes to the plant and plant operations remain unchanged. Therefore, this proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The safety function of the MSIVs is to provide a timely steam line isolation to mitigate the release of radioactive steam and limit reactor inventory loss under certain accident and transient conditions. The MSIVs are designed to automatically close whenever plant conditions warrant main steam line isolation. Changing the leakage limits to include an aggregate value does not affect the isolation function. No new equipment will be installed or utilized, and no new operating conditions will be initiated as a result of this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

Does the change involve a significant reduction in a margin of safety?

The total allowed leakage rate for all MSIVs remains unchanged at 46 scfh. Therefore, there will be no change in the types or significant increase in the amounts of any effluents released offsite, and, thus, the radiological analyses remain unchanged and within the guidelines of 10 CFR 100 and General Design Criteria 19. Therefore, these changes do not involve a significant reduction in the margin of safety.

Therefore, based upon the above evaluation, ComEd has concluded that these changes involve no significant hazards consideration.

ComEd has evaluated this proposed operating license amendment request against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. ComEd has determined that this proposed license amendment request meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9) and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92(b). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria:

(i) The amendment involves no significant hazards consideration.

As demonstrated in Attachment C, this proposed amendment does not involve any significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

As documented in Attachment C, there will be no significant increase in the amounts, and no significant change in the types, of any effluents released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from this change.