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United States Nuclear Regulatory Commission
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Subject: Supplemental Information Regarding the License Amendment Application to Revise Technical Specifications 3/4.3.2 and 3/4.6.3 (TAC No. 65685)

Gentlemen:

In response to your Request for Additional Information dated March 24, 1988 (Log No. 2528) Toledo Edison is providing additional information to assist in the review of the subject License Amendment application. This License Amendment application was submitted to the NRC on August 7, 1987 (Serial No. 1400), supplemented on March 21, 1988 (Serial No. 1500), and discussed during a meeting in Rockville, Maryland between Toledo Edison representatives, Mr. A. W. DeAgazio (NRC/NRR Davis-Besse Project Manager), and other members of the NRC Staff. This License Amendment proposes removing closure time requirements for valves connected to the secondary side of the steam generators listed in Technical Specification Table 3.3-5, Safety Features System Response Times, and Table 3.6-2, Containment Isolation Valves. Each NRC question, followed by Toledo Edison's response, is listed below:

1. Question: The proposed changes to the plant Technical Specifications (TS) in the licensee's letter dated August 7, 1987 will revise Table 3.3-5, Safety Features System Response Times to delete reference to the main steam warmup drain valves and atmospheric vent valves (AVV) receiving a high containment pressure or low reactor coolant system pressure SFAS automatic signal. It was indicated that the purpose of this change was to improve reliability and availability of the Main Feedwater System by reducing the chance of plant trips resulting from an inadvertent SFAS. The primary justification for this change was that those valves are normally closed during power operations. The SFAS signal serves to provide only a backup to procedural requirements for maintaining the valves in a closed position.

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The staff has two concerns with the above proposed TS change:

(1) These valves are normally closed, and an automatic closure of these valves does not isolate the feedwater system. Therefore, how can the elimination of the SFAS automatic signal for MS warmup drain valves and AVVs improve reliability of the main feedwater system?

(2) It is required in NUREG-0737, Item II.E.4.2 that following an accident all nonessential systems penetrating containment be automatically isolated. No credit can be given for operator action. By eliminating the SFAS state how this requirement is satisfied, or justify why those containment isolation valves can be granted a deviation from this requirement.

Response: (1) In the Safety Evaluation submitted with the License Amendment application, Toledo Edison stated that the primary purpose of removing the Safety Features Actuation System (SFAS) closure signal to the steam generator secondary isolation valves is to improve the reliability and availability of the Main Feedwater System and to minimize challenges to the Auxiliary Feedwater (AFW) System. The reliability of the Main Feedwater (MFW) System is impacted by the large valves affected by the proposed application, specifically the Main Steam Isolation Valves (MSIV) and the Main Feedwater Isolation Valves. The position of other valves (AVV's, MSIV bypass valves and MS warmup drain valves) is inconsequential from an accident analysis standpoint if the MSIV remains open and does not receive a closure signal.

Deleting SFAS closure of the AVV's simplifies operator response and control during a Steam Generator Tube Rupture (SGTR). Specifically, with an SFAS Level II closure of the AVV's and the Turbine Bypass System unavailable during a SGTR, the operators would have to override the AVV SFAS signal in order to cool down and depressurize the plant to below the Main Steam Safety Valve set pressure. Removing the SFAS closure of the MS warmup drain valves benefits the Human Factors engineering of the Control Room by maintaining consistency in the manner by which the Steam Generator secondary side is isolated following postulated accidents. Provision of one status and control location for major secondary side valves simplifies operator response to transients.

(2) In reviewing the valves and systems affected by this change, it has been concluded that for small break Loss of Coolant Accidents (SBLOCA), the availability of the Main Steam, the Main Feedwater, and the Auxiliary Feedwater Systems is desirable for event mitigation and, therefore, these systems should not be isolated during a SBLOCA. During a large break LOCA, automatic isolation will occur when the

Steam Feedwater Rupture Control System (SFRCS) low steam line pressure condition occurs. Until the SFRCS induced isolation is completed, the secondary side of the steam generator becomes effectively isolated during a large break LOCA due to the pressure gradients which will develop between the Reactor Coolant System (RCS) and the steam generator secondary side, and the containment vessel to steam generator differential pressures. Automatic closure through the SFRCS, given a Main Steam Line Break or Main Feedwater Line Break, where such isolation is indeed required, will continue to be available. For the reasons cited in the response above, it is also desired to make all associated valves respond consistently to a postulated accident. It is noted that the proposed design is consistent with General Design Criterion 57. Consequently, it is concluded that the requirements of 10CFR50, Appendix A and NUREG 0737, Item II.E.4.2 will still be met.

2. Question: The proposed TS change will revise TS 3/4.3.2, Table 3.3-5 to delete reference to the atmospheric vent valves, main steam warmup drain valves, main steam isolation valves, main feedwater stop valves, and main steam line warmup valves receiving a manual SFAS. It was indicated in a telecon of March 3, 1988 between the licensee and staff that those valves were also listed in Table 3.6-2, Containment Isolation Valves, under TS section 3/4.6.3. Therefore, the licensee considered it redundant and unnecessary to list those valves in Table 3.3-5.

The staff finds that the surveillance requirements under TS 3/4.3.2 are not the same as the requirements under TS 3/4.6.3. For example, a monthly CHANNEL FUNCTIONAL TEST is required by TS 3/4.3.2 but not required by TS 3/4.6.3. Identify the differences between these TS requirements and justify your proposed TS for the above valves.

Response: It is noted that the channel check, channel functional test and channel calibration requirements stipulated in Technical Specification Surveillance Requirements (SR) 4.3.2.1.1 and 4.3.2.2.1 only apply to instrument channels and not the actuated equipment (e.g., valves) except for the 18 month response time measurement which does require surveillance testing of the actuated equipment. With the proposed change the only instrumentation system applicable to automatic isolation of the affected valves will be SFRCS.

The Surveillance Requirements for the SFRCS provided in SR 4.3.2.2.1, therefore, replace the requirements of SR 4.3.2.1.1 for instrument string and output logic surveillance. This will continue to ensure that the sensors and logic channels which are depended upon for containment isolation are still tested in a manner and on a schedule comparable to that which now exists.

Actuated equipment will continue to be tested at least every eighteen months in fulfillment of SR 4.6.3.1.2. The time response requirement will be consistent with the most limiting value used in Safety Analysis Report (SAR) analyses. Where no specific time assumption was made in the SAR, the only response that is required to be verified is that the valve closes in response to its automatic containment isolation initiation signal. This is consistent with the existing requirements.

3. <u>Question</u>: The proposed change will revise TS section 3/4.3.6, Table 3.6-2 to delete isolation time requirements for the MSIV, MS warmup valves, MFW stop valves, AVV, MS warmup drain valves, and steam generator blowdown valves, along with the deletion of SFAS actuation. The licensee's evaluation of unreviewed safety question has been focused on large break LOCA and MSLB. The licensee should verify whether there are any other unidentified safety concerns or accident analyses that may be impacted by the proposed changes? For example, confirm the dose consequences for a steam generator tube rupture accident are within acceptable limits. Confirm that the environmental effects for a small MSLB inside or outside containment are not adversely affected. Verify that the small break LOCA accident analysis is not adversely affected by the proposed change. Provide additional discussion and/or analysis to justify that there is no unreviewed safety question resulting from the proposed change.

<u>Response</u>: The Safety Evaluation submitted by Serial No. 1400 provides discussion and rationale for the primary focus on large break LOCAs and Main Steam Line Breaks (MSLB). With the present plant configuration, the only accident which can cause an SFAS Level 4 signal is a large break LOCA. Consequently, when comparing the proposed configuration to the existing plant configuration for analysis impact, only the large break LOCA need be considered for the majority of the valves. The Safety Evaluation also discusses the valves that receive an SFAS Level 2 confirmatory close signal, and why that signal is inconsequential when the MSIV's are still open. For SBLOCAs, steam generator isolation is not desired to aid in accident mitigation. As stated in the March 21, 1988 supplemental letter (Serial No. 1500), the dose consequences of a Steam Generator Tube Rupture (SGTR) are not affected by this change. Per the Davis-Besse Updated Safety Analysis Report (USAR), Section 15.4.2, the consequences of any steam line break which is beyond the capability of the Integrated Control System are bounded by the MSLB analysis. Chapters 3 and 6 of the USAR fully examine the environmental effects of steam line breaks inside and outside of containment. The only mitigating

isolation features assumed in these analyses are from SFRCS. These features are not affected by the proposed change. Since there is no change being made to the SFRCS which mitigates such Steam Line Breaks, there is no impact on the USAR analyses.

The applicable accidents have, therefore, been reviewed and, as summarized in the Safety Evaluation contained in Serial No. 1400, an unreviewed safety question does not exist.

Toledo Edison believes the above addresses the NRC request. If you have any additional questions, please contact Mr. R. W. Schrauder, Nuclear Licensing Manager, at (419) 249-2366.

Very truly yours,

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cc: DB-1 Resident Inspector
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