

NUREG-1433, "Standard Technical Specifications General Electric Plants, BWR/4," Revision 2, dated June 2001.

*Basis for proposed no significant hazards consideration determination:* As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in probability or consequences of an accident previously evaluated.

The proposed change to the Technical Specifications (TS) Surveillance Requirement (SR) 3.6.1.3.8 will incorporate into the SR, wording specified in NUREG-1433, "Standard Technical Specifications General Electric Plants, BWR/4," Revision 2, dated June 2001. The proposed change will specify that the testing required by SR 3.6.1.3.8 is applicable to reactor instrumentation line excess flow check valves (EFCVs). The performance of TS surveillance testing is not a precursor to any accident previously evaluated. Thus, the proposed change does not have any effect on the probability of an accident previously evaluated.

The function of reactor instrumentation line EFCVs, in combination with other accident mitigation features, is to limit fission product release. The surveillance testing specified in SR 3.6.1.3.8 will provide assurance that the reactor instrumentation line EFCVs will perform as designed. Thus, the radiological consequences of any accident previously evaluated are not increased.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not affect the control parameters governing unit operation or the response of plant equipment to transient conditions. The proposed change does not introduce any new equipment, modes of system operation or failure mechanisms.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed changes do not involve a significant reduction in a margin of safety.

NUREG-1433, Rev. 2, provided licensees with the latest NRC recommended content and format for TS. The NUREG-1433 SR for testing EFCVs, SR 3.6.1.3.10, specifies that this testing is associated with reactor instrumentation line EFCVs. The Basis to SR 3.6.1.3.10 in NUREG-1433, Rev. 2, provides a reference to NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," dated June 2000. NEDO-32977-A was approved for use by licensees in a NRC letter dated March 14, 2000. NEDO-32977-A states the following on the scope of TS testing associated with EFCVs:

EFCVs in instrument lines which connect to the reactor coolant pressure boundary (RCPB) are normally tested during refueling outages to meet Technical Specification requirements. Instrument lines that connect to the containment atmosphere, such as those which measure drywell pressure, or monitor the containment atmosphere or suppression pool water level, are considered extensions of primary containment. A failure of one of these instrument lines during normal operation would not result in the closure of the associated EFCV, since normal operating containment pressure is not sufficient to operate the valve. Such EFCVs will only close with a downstream line break concurrent with a Loss of Coolant Accident (LOCA). Since these conditions are beyond the plant design basis, EFCV closure is not needed and containment atmospheric instrument line EFCVs need not be tested.

The proposed change will incorporate the wording from NUREG-1433 into LaSalle County Station SR 3.6.1.3.8 to limit the scope of TS required testing to EFCVs that are directly connected to the RCPB.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based upon the above, Exelon Generation Company concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the requested amendments involve no significant hazards consideration.

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*FirstEnergy Nuclear Operating Company, Docket No. 50-346, Davis-Besse Nuclear Power Station, Unit 1, Ottawa County, Ohio*

*Date of amendment request:* May 14, 2003.

*Description of amendment request:* The proposed amendment would modify the Technical Specifications by allowing entry into Mode 3 operation (shutdown with reactor coolant system temperature equal to or greater than 280 degrees Fahrenheit) during the current outage only with neither high pressure injection (HPI) pump capable of taking suction from the low pressure injection system trains when aligned for containment sump recirculation. The HPI system will otherwise be operable.

*Basis for proposed no significant hazards consideration determination:*

As required by 10 CFR 50.91(a), the licensee has provided their analysis of the issue of no significant hazards consideration, which is presented below:

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

*Response:* No.

The proposed change allows the plant to operate in Mode 3 in support of RCS [reactor coolant system] leakage inspection activities conducted during the ongoing Thirteenth Refueling Outage, utilizing a limited exception to Limiting Condition for Operation (LCO) 3.5.2. This LCO applies in plant operational Modes 1 (Power Operation), 2 (Startup), and 3 (Hot Standby). Under the proposed exception, for entry into Mode 3, both HPI trains would be required to be operable except for the capability of maintaining suction from the containment emergency sump during the recirculation phase.

The ability of the HPI pumps to draw suction from the containment emergency sump (via the LPI [low pressure injection] pumps) is a design feature credited by the Davis-Besse Nuclear Power Station Updated Safety Analysis Report (USAR) for mitigation of various types of loss-of-coolant accidents (LOCAs). Due to the potential susceptibility to damage from debris contained in the pumped fluid, the existing HPI pumps may not be capable of maintaining suction from the containment emergency sump without an increased probability for malfunction. However, the current plant conditions are unique in that decay heat generation rate in the reactor core is extremely low due to the fact that the plant has not operated in more than 14 months and 76 unirradiated fuel assemblies have been loaded into the core, replacing irradiated fuel assemblies.

A LOCA evaluation has been performed considering the current reactor core decay heat generation rate. The evaluation shows that in the unlikely event that a LOCA did occur while operating in Mode 3 under the proposed exception, the accident can be mitigated without crediting HPI flow during the recirculation phase, while crediting additional operator actions not presently credited in the USAR. In addition, a risk evaluation has been performed and shows that the increase in core damage frequency, accounting for human error probability for the additional operator actions, is very small. Also, in the unlikely event that a LOCA did occur while operating in Mode 3 under the proposed exception, radiological consequences would be very small compared to the accident analyses results of record, given the fission product decay over the extended plant shutdown. Therefore, the proposed change would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

*Response:* No.