APPLICATION FOR AMENDMENT

TO

FACILITY OPERATING LICENSE NO. NPF-3

FOR

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NO. 1

Attached are requested changes to the Davis-Besse Nuclear Power Station, Unit No. 1 Facility Operating License No. NPF-3. Also included are the Safety Evaluation and Significant Hazards Consideration.

The proposed changes (submitted under cover letter Serial No. 1490) concern:

License Condition 2.C.(3)(d)
Technical Specification Section 3/4.4.2, Safety Valves-Shutdown, Figures 3.4-2a and 3.4-2b
Technical Specification Section 3/4.4.9.1, Pressure/Temperature Limits - Reactor Coolant System, and Figures 3.4-2, 3.4-3, 3.4-4, Table 4.4-5
Technical Specification Bases Section 3/4.4.9, Bases Table 4-1 and Bases Figures 4-1 and 4-2

By _____
D. C. Shelton, Vice President, Nuclear

Sworn and subscribed before me this 30th day of March, 1988.

_____
Notary Public, State of Ohio

My Commission expires 5/15/91

LAURIE A. HINKLE
Notary Public, State of Ohio
My Commission Expires May 15, 1991

The following information is provided to support issuance of the requested changes to the Davis-Besse Nuclear Power Station, Unit No. 1 Operating License No. NPF-3, License Condition 2.C.(3)(d), Appendix A Technical Specifications Sections 3/4.4.2, 3/4.4.9.1; Figures 3.4-2a, 3.4-2b, 3.4-2, 3.4-3, and 3.4-4; Table 4.4-5; and Bases Section 3/4.4.9, Bases Table 4-1 and Bases Figures 4-1 and 4-2.

A. Time required to Implement: This change will be implemented by the licensee by the end of the fifth refueling outage. However, issuance by June 1, 1988 is requested to support operator training prior to startup from the fifth refueling outage.

B. Reason for Change: Revise the Technical Specifications to: 1) change License Condition 2.C.(3)(d) to reflect the validity of the Reactor Coolant System Low Temperature Overpressure protection to 10 Effective Full Power Years (EFPY), 2) delete the reactor vessel material surveillance requirements and surveillance specimen withdrawal schedule from Technical Specifications and reflect the governing of this program by Babcock and Wilcox Report BAW-1543A, "Integrated Reactor Vessel Materials Surveillance Program", and 3) incorporate new reactor coolant system (RCS) pressure/temperature limits and curves (for heatup, cooldown and hydrostatic testing) and pressurizer level/RCS pressure limit curves (for new Low Temperature Overpressure protection considerations) to reflect reactor vessel material properties to 10 EFPY.

C. Safety Evaluation: See attached Safety Evaluation (Attachment No. 1).

D. Significant Hazards Consideration: See attached Significant Hazards Consideration (Attachment No. 2).

E. License Condition 2.C.(3)(d) revision and Technical Specification Change Pages (Attachment No. 3)

SAFETY EVALUATION

PURPOSE

The purpose of this Safety Evaluation is to review proposed changes to the Davis-Besse (DB) Nuclear Power Station, Unit No. 1, Operating License, and its Appendix A, Technical Specifications, as described in FCR 87-0117, Rev. B. Changes are being made in Sections 3.4.2, 3.4.9.1, and 4.4.9.1.2; Figures 3.4-2a, 3.4-2b, 3.4-2, 3.4-3, and 3.4-4; Table 4.4-5; Bases Section 3/4.4.9, Bases Table 4-1; and Bases Figures 4-1 and 4-2.

The proposed revisions are being made to reflect the changes in the reactor vessel metal material properties due to exposure to high energy neutrons. Presently, the Davis-Besse Operating License requires that Toledo Edison provide the NRC a reanalysis and proposed modifications, as necessary to ensure continued means of protection against low temperature reactor coolant system overpressure (LTOP) events prior to operation beyond five Effective Full Power Years (EFPY). The current Technical Specifications reflect the material properties of the vessel after exposure to five Effective Full Power Years (EFPY) of neutron exposure. The proposed revisions will reflect the material properties with exposure equivalent to ten EFPY of reactor operation. The reactor vessel will have approximately 4.3 EFPY of exposure at the end of the present fuel cycle (Cycle 5).

The specific changes to be made to the Technical Specifications are:

1. Revision of the Reactor Coolant System (RCS) pressure-temperature operating limits during heatup, cooldown, and inservice leak and hydrostatic tests (ISLH), Figures 3.4-2, 3.4-3, 3.4-4, to reflect vessel neutron exposure.

2. Revision of the maximum allowable heatup and cooldown rates permitted by Limiting Condition for Operation (LCO) 3.4.9.1 to reflect the rates assumed when calculating the pressure-temperature limitation curves.

3. Revision of the evaluation required by the Action Statement of 3.4.9.1 to determine the effects of exceeding the pressure-temperature limits. Exceeding the limits does not necessarily affect the reactor vessel material properties, however; the reactor vessel's integrity may have been affected due to flaw initiation or propagation. The proposed wording reflects the need for the evaluation to determine this effect.

4. The Action Statement of LCO 3.4.9.1 is being revised to require the plant to be placed in COLD SHUTDOWN rather than requiring Tavg to be less than 200°F and pressure less than 500 psig since this could put the plant outside the acceptable region of operation and still allow compliance. By specifying COLD SHUTDOWN (Tavg < 200°F) the LCO of the Technical Specification is not superseded and full compliance is assured.

5. Revision of Surveillance Requirement 4.4.9.1.2 and deletion of Table 4.4-5: Reactor Vessel Material Irradiation Surveillance Schedule to reflect the NRC's acceptance of BAW-1543A (Reference 1) as an adequate method of ensuring reactor vessel material surveillance samples are withdrawn and tested as required in 10 CFR 50, Appendix H.

6. Revision of Bases Section 3/4.9 to delete Figures 4-1 and 4-2 and Table 4-1, and modification of associated text to reflect their incorporation into BAW-10046A (Reference 2), BAW-1543A (Reference 1), and BAW-1701 (Reference 3), respectively. The text of this Bases Section is also being modified to reflect the change in the P-T curves to allow operation to 10 EFPY, to reflect changes in theory regarding vessel material property composition sensitivity from phosphorus to nickel, and the fact that steam generator tubes have been determined to no longer restrict plant P-T limits. An editorial change to revise "reactor vessel line surface conditions" to "reactor vessel beltline conditions" is also being made.

7. Revision of Figures 3.4-2a and 3.4-2b: Reactor Coolant System Pressure versus Pressurizer Level in Modes 4 and 5 curves for an inoperable Decay Heat Removal System (DHRS) relief valve DH-4849 to reflect the lower pressure limits due to reactor vessel neutron exposure.

8. Revision of License Condition 2.C.(3)(d) to require reanalysis and any necessary modifications to prevent LTOP events prior to operation beyond 10 EFPY.

SYSTEMS AFFECTED

Reactor Coolant System/Reactor Vessel

DOCUMENTS AFFECTED

Davis-Besse Nuclear Power Station, Unit 1 Operating License Appendix A: Technical Specifications

Davis-Besse Updated Safety Analysis Report

Davis-Besse Nuclear Power Station System Procedures SP 1505.03

SAFETY FUNCTIONS AFFECTED

The Reactor Coolant System consists of the mechanical piping and the motive force to circulate reactor coolant between the reactor, where energy is added to the coolant, and the Steam Generator, where energy is removed from the coolant. The Reactor Vessel is an integral part of the RCS. In order for the reactor coolant to reach the core, the integrity of the vessel must be assured. As discussed below, in order to ensure the vessel's integrity is maintained, its material properties must be reflected in the maximum allowable stress which is permitted due to pressure and thermal gradients.

EFFECTS ON SAFETY

10 CFR Part 50, Appendix G, Fracture Toughness Requirements, provides pressure vessel toughness requirements which are meant to protect the pressure vessel from brittle fracture and fatigue. The Davis-Besse (D-B) Technical Specifications include Limiting Condition for Operation (LCO) 3.4.9.1, and pressure-temperature limit curves in Figures 3.4-2 (for heatup and criticality), 3.4-3 (for cooldown), and 3.4-4 (for heatup and cooldown during inservice leak and hydrostatic (ISLH) tests). This LCO, with its associated pressure-temperature limit curves meets the requirements of 10CFR50, Appendix G, thereby ensuring protection of the Reactor Coolant System (RCS) pressure boundary integrity from nonductile failures during any anticipated operational occurrences. Consequently, reactor vessel brittle fracture is considered to be an incredible occurrence, and loss of coolant accidents (LOCA's) are limited to ruptures of primary coolant piping. The methods and criteria employed to establish the LCO and operating pressure and temperature limits are described in B&W Topical Report BAW-10046A (Reference 2).

When the low-alloy ferritic steels used in the fabrication of the D-B reactor vessel (RV) are exposed to fast neutrons (E > 1 MeV), their resistance to brittle fracture is decreased. More specifically, these low-alloy steels show a rise in the temperature of transition from brittle to ductile fracture (RTndt or Reference Nil Ductility Transition Temperature) and an accompanying reduction in the upper shelf impact toughness.

Because of the decrease in the non-ductile failure resistance of these low-alloy steels with increased fast neutron fluence (fast neutron flux integrated over time), the effects of fast neutron exposure must be accounted for when determining the maximum allowable stresses in the reactor vessel due to pressure and temperature transients in the RCS. In order to determine the vessel's exposure and the effects of the exposure on material properties, surveillance specimen of vessel material have been installed in the reactor vessel. Periodically some of these samples are withdrawn and tested to determine shifts in material properties. The D-B Reactor Vessel Material Surveillance Program is prepared in accordance with the requirements of 10 CFR Part 50, Appendix H, Reactor Vessel Material Surveillance Program Requirements, and ASTM Standard Practice E185-82.

B&W Report BAW-1701, Analysis of Capsule TE1-F (Reference 3), reports the results of the examination of the first material surveillance specimen capsule, removed at the end of the first Davis-Besse fuel cycle in accordance with the Davis-Besse Reactor Vessel Material Surveillance Program. The material properties of the reactor vessel metal were conservatively revised to values which reflect actual expected values based on experimental data rather than the analytically predicted values previously used.

The procedures of Regulatory Guide 1.99, Rev. 2, Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials, were then used to predict the radiation induced increase in RTndt as a function of the material's copper and nickel content and neutron fluence. Previously, it was believed that the amount of phosphorus contained in low-alloy steels played a major role in determining the metal's fracture toughness. Through continuing research, it has been found that nickel, rather than phosphorus, plays a greater role in determining the fracture toughness of low-alloy steels. Consequently, nickel has replaced phosphorus in the latest revision of Regulatory Guide 1.99.

Since the plant was originally designed, research on the fracture toughness of the steam generator tubes has been completed. This work demonstrated that the tube material has a fracture toughness sufficiently high to warrant eliminating consideration of tube integrity during heatups and cooldowns. Consequently, the regions of the heatup, cooldown, and ISLH curves originally limited by the steam generator tubes have risen to the values calculated for the next limiting component.

During Mode 1 and 2 operations, the RCS is above the Reference Transition Temperature of the reactor vessel and brittle fracture is not of concern. Pressurized Thermal Shock (PTS) issues are satisfactorily addressed within the plant Emergency Procedures and are not of concern for Technical Specification 3.4.9.

For a limited region when the plant is in Mode 3, the operator, in conjunction with Administrative Controls, provides the primary means of maintaining the plant within the limits of the applicable P-T curve. In this region, the potential causes of an LTOP event, which is viewed as low probability transient, are inadvertent initiation of High Pressure Injection (HPI) flow into the RCS and failure of the Make Up Control valve to a full open position. Inadvertent initiation of HPI flow to the RCS is not a credible event since it would require multiple failures within the Safety Features Actuation System to both establish a flow path and start the HPI pumps. Additionally, even if HPI were injected into the RCS, the maximum potential RCS pressure is the shut-off head of the HPI pump. Although this pressure exceeds the pressure limitations of the heat-up curve, it is still below the maximum allowable pressure of the ISLH curve. Since the ISLH curve maintains a significant margin of safety from the maximum pressure required to initiate or propagate a flaw, all USAR analyses are unaffected and still valid. Administrative controls will

require the reactor operator to maintain the RCS pressure and pressurizer level within a region where, should the Make Up Control valve fail open, the operator would have at least ten minutes to respond to the situation without exceeding the pressure limitations of the heat-up or cooldown curves. This situation exists until the RCS temperature is high enough (about 330°F) for the Power Operated Relief Valve (PORV) to provide automatic overpressure protection. This method of operation is acceptable since it only occurs for a limited period during plant startup and shutdown.

During Mode 4 and Mode 5 operations, the susceptibility of low-alloy ferritic steel to brittle fracture increases dramatically. Consequently, the operating pressure limits are particularly rigorous below 280°F. The NRC first raised the issue of low temperature overpressure protection of the reactor coolant system at Davis-Besse in Reference 5. This letter requested that Toledo Edison describe what steps would be taken to ensure reactor coolant system pressure boundary integrity would be maintained while at low temperatures. In order to resolve this issue, Limiting Condition for Operation 3.4.2 and Figures 3.4.2-a & 3.4.2-b were added to the Technical Specifications in License Amendments No. 28 and No. 57.

As described in the NRC Safety Evaluation Report supporting License Amendment No. 57, Davis-Besse Unit 1 is protected from overpressure events at low temperatures (Modes 4 and 5) in two ways:

1. Decay Heat Removal System (DHRS) 4 inch relief valve DH4849 in the DHRS suction line provides active protection. DH4849 is put in service below 280°F by opening isolation valves DH11 and DH12, and removing power to their operators.

   This relief valve is sized such that it can fully relieve any overpressure transient condition which could occur during shutdown as long as the maximum allowable pressure is above its setpoint.

2. The plant design and operating philosophy prohibits the plant from being operated in a water solid condition, except for system hydrotests, thereby providing passive protection. A steam bubble is present in the pressurizer at RCS pressures above 30 to 50 psig. A nitrogen bubble of 30 to 50 psig is kept in the pressurizer when the pressurizer temperature can no longer sustain a 30 to 50 psig steam bubble.

   When the plant is in Mode 5 at a temperature (120°F during cooldown for the proposed curves) where the maximum allowable pressure is below the setpoint of DH4849 there is no automatic overpressure protection available. However, when an RCS temperature of 140°F is reached, the Makeup system is procedurally shutdown and the RCS pressure is procedurally reduced to and controlled at less than 75 psig. Consequently, the only potential source of overpressurization has been eliminated. This and other credible sources of overpressurization are further discussed in the following paragraphs.

Should relief valve DH4849 become inoperable, there are two postulated events which could cause a low temperature overpressurization at Davis-Besse Unit 1. These are inadvertent actuation of the High Pressure Injection (HPI) System and the makeup control valve (MU32) failing open, with continuous makeup pump operation. All other possible causes are considered incredible in view of existing operating procedures and interlocks. The actions to be taken if DH4849 is not operable are specified in the Action Statement of Technical Specification 3.4.2 and deal with these two causes:

- HPI actuation is excluded by disabling both HPI pumps.

- Makeup control valve failure is prevented from causing RCS overpressurization by limiting the total amount of water injected into the system to the most that can be accommodated by the pressurizer. This is done by disabling automatic transfer of makeup pump suction to the Borated Water Storage Tank (BWST) on low makeup tank level, limiting the makeup tank level to 73 inches or less, and by limiting the pressurizer level and RCS pressure in the acceptable region of Figure 3.4-2a (Mode 4) or Figure 3.4-2b (Mode 5).

Note that, as discussed earlier, inadvertent actuation of HPI is not a credible source of pressurization since it involves multiple failures.

During Mode 6 operation, the RCS is open to the atmosphere and, consequently, overpressurization is not feasible.

The changes in the heatup, cooldown, and in-service leak and hydrostatic test (ISLH) curves and rates, as documented in BAW-2011 (Reference 6) incorporate the changes in the material properties due to the neutron exposure from 10 EFPY of reactor operation. Since these curves satisfactorily account for the changes in material properties, the revision of the curves does not affect the safety function of the RCS/Reactor Vessel.

In comparing the proposed curves with the existing curves several changes are noted. The Heatup curve generally has shifted downward for any given temperature. It does remain, however, above the setpoint of DH4849 at all RCS temperatures below 280°F so that during plant heatup, DH4849 will provide overpressure protection, as before. In the range of about 235°F to 245°F the new pressure limit is actually greater than the old limits. This is due to the change in the maximum allowable heatup rate from 100°F/hr to 50°F/hr and the elimination of concern about the pressure-temperature capabilities of the steam generator tubes. Previously, there was only a very limited range of temperatures (about 280 to 285°F) where Administrative Controls provided overpressure protection. This range now covers RCS temperatures from 280°F to 335°F. This is deemed acceptable based on the previous discussion of Mode 3 operations. The minimum pressure-temperature for criticality curve has shifted to the right for any given pressure due to the changes in material properties. Also, the knee in the curve has been eliminated due to the improved understanding of steam generator tube capabilities. In actuality this curve is not germane since criticality below 525°F is prohibited by the Technical Specifications.

The Cooldown curve has generally shifted downward in the maximum permissible pressure for a given temperature due to the increased brittleness of the vessel metal. The amount of shift at low temperatures has been minimized by lowering the maximum allowable cooldown rate from -100°F/hr to -50°F/hr when below 270°F. At or above 270°F the cooldown rate remains at -100°F/hr, as before. The allowable pressure during cooldown above about 250°F increased due to the elimination of steam generator tube P-T concerns. The range of temperatures (from 280°F to about 375°F) where overpressure protection is provided by Administrative Controls has increased (previously from 280°F to 360°F). This change is acceptable since the amount of time spent in this range is limited and the Administrative Controls provide appropriate overpressure protection against a Makeup system failure. Since the maximum allowable pressure is well above the shutoff head of the HPI pumps, this event is not of concern. Also as noted above, the actual allowable pressure in this range is actually higher than previously allowed so that operators have more time to respond to any pressure increase prior to the limit being exceeded. The range where the maximum allowable pressure is below the setpoint of DH4849 has increased from less than about 80°F to less than 120°F. However, as discussed above, the Administrative Controls on plant operation in this temperature range make this increase acceptable.

The ISLH curve has generally shifted downward in allowable pressure for any given temperature for the same reasons the heatup and cooldown curves have shifted, as discussed above. This is because it represents the lowest allowable pressure from either the heatup curve or the cooldown curve, as further modified for a safety factor of 1.5, as specified in the ASME Code, Appendix G. The split in the curves has been eliminated due to the removal of concerns over steam generator tube fracture toughness.

The Reactor Vessel Material Surveillance Program which is used to determine the material properties of the reactor vessel has been evaluated by the NRC and found acceptable. In a letter dated May 8, 1985 from H. L. Thompson, Jr. of the NRC to H. B. Tucker, Chairman of the BWOG the NRC recommended removing specific details of the Surveillance Program from the Technical Specifications. The proposed changes do not affect the safety function of the RCS/Reactor Vessel since the currently approved requirements will be retained by referencing a document approved by the NRC.

The changes proposed for the RCS Pressure vs. Pressurizer level curves are being made to reflect the change in the P-T curves. Since they are calculated based on the 10 EFPY curves, as documented in B&W Document No. 32-1170349-00 they preserve the same conditions as the previous curves for operation up to 5 EFPY. Therefore, they help maintain the safety function of the reactor vessel.

It should be noted that the curves to be included in the Technical Specification do not include the effects of instrument string uncertainty. The curves which are to be put into the operating procedures will be modified so as to conservatively correct for these uncertainties.

The revision of the License Condition 2.C.(3)(d) is administrative in nature and does not directly affect the safe operation of the plant. By preserving the same License Condition, but requiring a review of LTOP concerns before the end of 10 EFPY ensures that Toledo Edison will again evaluate the effects of reactor vessel neutron embrittlement and take whatever action is necessary to prevent overpressurization of the RCS.

UNREVIEWED SAFETY QUESTION EVALUATION

Removing the reactor vessel materials surveillance requirements and specimen withdrawal schedule from the Technical Specifications, so that controlling authority is transferred to BAW-1543A ...

1. Does not increase the probability of an accident previously evaluated in the USAR because it is an administrative change that does not influence any accident analyses, since material properties will still be verified as assumed. (10CFR50.59(a)(2)(i))

2. Does not increase the consequences of an accident previously evaluated in the USAR because it does not change any assumptions or requirements of previous accident analyses. (10CFR50.59(a)(2)(i))

3. Does not increase the probability of a malfunction of equipment important to safety previously evaluated in the USAR because it is an administrative change that does not directly affect any safety function. (10CFR50.59(a)(2)(i))

4. Does not increase the consequences of a malfunction of equipment important to safety previously evaluated in the USAR because it does not affect the safety function of any equipment. (10CFR50.59(a)(2)(i))

5. Does not create the possibility of an accident of a different type than any evaluated previously in the USAR because the Davis-Besse Reactor Vessel Material Surveillance Schedule is still controlled as required by 10 CFR 50, Appendix H. (10CFR50.59(a)(2)(ii))

6. Does not create the possibility of a malfunction of equipment of a different type than any evaluated previously in the USAR because no equipment functions are affected in any way. (10CFR50.59(a)(2)(ii))

7. Does not reduce any margin of safety as defined in the basis for any Technical Specification because the basis for the Reactor Vessel Material Surveillance Program has not changed, and will still be adequately controlled with NRC approval, as required by 10 CFR 50, Appendix H. (10CFR50.59(a)(2)(iii))

Modification of the operating pressure and temperature limit curves to extend their validity to 10 EFPY, and of the pressurizer level-RCS pressure limit curves that are applicable when DHRS relief valve DH4849 is inoperable ...

1. Does not increase the probability of an accident previously evaluated in the USAR because the same level of protection of RCS pressure boundary integrity under all postulated circumstances is preserved. (10CFR50.59(a)(2)(i))

2. Does not increase the consequences of an accident previously evaluated in the USAR because revision of these curves does not affect any assumption made in the accident analysis of the limiting design basis event, which is the instantaneous failure of primary coolant piping at full power operation. (10CFR50.59(a)(2)(i))

3. Does not increase the probability of a malfunction of equipment important to safety previously evaluated in the USAR because the level of protection of equipment in the RCS under all postulated circumstances is preserved. (10CFR50.59(a)(2)(i))

4. Does not increase the consequences of a malfunction of equipment important to safety previously evaluated in the USAR because revision of these curves does not affect any assumptions made in the accident analyses of small and large breaks of primary coolant piping. (10CFR50.59(a)(2)(i))

5. Does not create the possibility of an accident of a different type than any evaluated previously in the USAR because adequate protection of the Reactor Vessel from postulated brittle fracture is preserved under all postulated circumstances, and therefore reactor vessel failure remains not credible. (10CFR50.59(a)(2)(ii))

6. Does not create the possibility of a malfunction of equipment of a different type than any evaluated previously in the USAR because all equipment in the RCS is as adequately protected as before from failure due to brittle fracture under all postulated circumstances. (10CFR50.59(a)(2)(ii))

7. Does not reduce any margin of safety as defined in the basis for any Technical Specification because this change maintains the present margin of safety from brittle fracture required by 10 CFR 50, Appendix G under all postulated circumstances. (10CFR50.59(a)(2)(iii))

Revising the License Condition 2.C.(3)(d) to require a reanalysis of LTOP events and proposal of modifications required to prevent LTOP events prior to operating beyond 10 EFPY ...

1. Does not increase the probability of an accident previously evaluated in the USAR because it is administrative in nature and does not affect the probability of experiencing any accidents previously analyzed. (10CFR50.59(a)(2)(i))

2. Does not increase the consequences of an accident previously evaluated in the USAR because the change is administrative in nature and does not affect any previously analyzed accidents. (10CFR50.59(a)(2)(i))

3. Does not increase the probability of a malfunction of equipment important to safety previously evaluated in the USAR because the change is administrative in nature and does not directly affect any equipment. (10CFR50.59(2)(a)(i))

4. Does not increase the consequences of a malfunction of equipment important to safety previously evaluated in the USAR because the change is administrative in nature and does not directly affect any plant equipment or USAR analyses, including equipment failure analyses. (10CFR50.59(2)(a)(i))

5. Does not create the possibility of an accident of a different type than any evaluated previously in the USAR because the change is administrative in nature and does not change the way the plant is operated. (10CFR50.59(a)(2)(ii))

6. Does not create the possibility of a malfunction of equipment of a different type than any evaluated previously in the USAR because the change is administrative in nature and does not affect any equipment or the way the plant is operated. (10CFR50.59(a)(2)(ii))

7. Does not reduce the margin of safety as defined in the basis for any Technical Specification because the change is administrative in nature and does not affect any of the analyses which form the foundation of the Technical Specifications. (10CFR50.59(a)(2)(iii))

In conclusion, no unreviewed safety question exists.

REFERENCES

1. B&W Owners Group Material Committee Report BAW-1543A, Rev. 2, Integrated Reactor Vessel Materials Surveillance Program, February 1984, approved by NRC May 1985.

2. BAW-10046A, Methods of Compliance with Fracture Toughness and Operational Requirements of Appendix G to 10 CFR 50, October 1975.

3. BAW-1701, Analysis of Capsule TE1-F, January 1982.

4. Letter dated October 1, 1976, from John F. Stolz (NRC) to Lowell E. Roe (TECo).

5. Letter dated October 1, 1976 from John Stolz, USNRC, to Lowell E. Roe, TECo.

6. BAW-2011, Pressure Temperature Limits for 10 EFPY, November, 1987.

SIGNIFICANT HAZARDS CONSIDERATION

PURPOSE

The purpose of this Significant Hazards Consideration is to review proposed changes to the Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS), Operating License, and its Appendix A, Technical Specifications. Changes are being proposed to License Condition 2.C(3)(d) and Technical Specification Sections 3.4.2, 3.4.9.1, and 4.4.9.1.2; Figures 3.4-2a, 3.4-2b, 3.4-2, 3.4-3, and 3.4-4; Technical Specification Table 4.4-5; Technical Specification Bases Section 3/4.4.9 and Bases Table 4-1; and Technical Specification Bases Figures 4-1 and 4-2.

The proposed revisions are being made to reflect the changes in the reactor vessel metal material properties due to exposure to high energy neutrons. Presently, the DBNPS Operating License requires that Toledo Edison provide the NRC a reanalysis and proposed modifications, as necessary, to ensure continued means of protection against low temperature reactor coolant system overpressure (LTOP) events prior to operation beyond five Effective Full Power Years (EFPY). The current Technical Specifications reflect the material properties of the vessel after exposure to five Effective Full Power Years (EFPY) of neutron exposure. The proposed revisions will reflect the material properties with exposure equivalent to ten EFPY of reactor operation. The reactor vessel will have approximately 4.3 EFPY of exposure at the end of the present fuel cycle (Cycle 5).

The specific changes to be made to the Technical Specifications are:

1. Revision of the Reactor Coolant System (RCS) pressure-temperature operating limits during heatup, cooldown, and inservice leak and hydrostatic tests (ISLH), Figures 3.4-2, 3.4-3, 3.4-4, to reflect vessel neutron exposure.

2. Revision of the maximum allowable heatup and cooldown rates permitted by Limiting Condition for Operation (LCO) 3.4.9.1 to reflect the rates assumed when calculating the pressure-temperature limitation curves.

3. Revision of the evaluation required by the Action Statement of LCO 3.4.9.1 to determine the effects of exceeding the pressure-temperature limits. Exceeding the limits does not necessarily affect the reactor vessel material properties, however; the reactor vessel's integrity may have been affected due to flaw initiation or propagation. The proposed wording reflects the need for the evaluation to determine this effect.

4. The Action Statement of LCO 3.4.9.1 is being revised to require the plant to be placed in COLD SHUTDOWN rather than requiring Tavg to be less than 200°F and pressure less than 500 psig since this could put the plant outside the acceptable region of operation and still allow compliance. By specifying COLD SHUTDOWN (Tavg < 200°F), the LCO is not superseded and full compliance is assured.

5. Revision of Surveillance Requirement 4.4.9.1.2 and deletion of Table 4.4-5, Reactor Vessel Material Irradiation Surveillance Schedule, to reflect the NRC's acceptance of BAW-1543A (Reference 1) as an adequate method of ensuring reactor vessel material surveillance samples are withdrawn and tested as required in 10 CFR 50, Appendix H.

6. Revision of Bases Section 3/4.9 to delete Figures 4-1 and 4-2 and Table 4-1, and modification of associated text to reflect their incorporation into BAW-10046A (Reference 2), BAW-1543A (Reference 1), and BAW-1701 (Reference 3), respectively. The text of this Bases Section is also being modified to reflect the change in the P-T curves to allow operation to 10 EFPY, to reflect changes in theory regarding vessel material property composition sensitivity from phosphorus to nickel, and the fact that steam generator tubes have been determined to no longer restrict plant P-T limits. An editorial change to revise "reactor vessel line surface conditions" to "reactor vessel beltline conditions" is also being made.

7. Revision of Figures 3.4-2a and 3.4-2b, Reactor Coolant System Pressure - Pressurizer Level Limits for Inoperable Decay Heat Removal System Relief Valve (DH-4849) in Modes 4 and Mode 5, respectively, to reflect the lower pressure limits due to reactor vessel neutron exposure. In addition, Action A.2.b.2 of Technical Specification 3/4.4.2 is being administratively revised to correct the numbers of these figures in order to maintain consistency in Technical Specification figure numbering.

8. Revision of License Condition 2.C.(3)(d) to notify the NRC of a reanalysis and necessary modifications to prevent LTOP events prior to operation beyond 10 EFPY.

SYSTEMS AFFECTED

Reactor Coolant System/Reactor Vessel

DOCUMENTS AFFECTED

Davis-Besse Nuclear Power Station, Unit No. 1 Operating License, and Operating License Appendix A, Technical Specifications

Davis-Besse Nuclear Power Station Updated Safety Analysis Report, July 1987

Davis-Besse Nuclear Power Station System Procedure SP 1505.03, "Surveillance Capsule Withdrawal Schedule"

SAFETY FUNCTIONS AFFECTED

The Reactor Coolant System (RCS) consists of the mechanical piping and the motive force to circulate reactor coolant between the reactor, where energy is added to the coolant, and the Steam Generator, where energy is removed from the coolant. The Reactor Vessel is an integral part of the RCS. In order for the reactor coolant to reach the reactor core, the integrity of the vessel must be assured. As discussed below, in order to ensure the vessel's integrity is maintained, its material properties must be reflected in the maximum allowable stress which is permitted due to pressure and thermal gradients.

EFFECTS ON SAFETY

10CFR50, Appendix G, Fracture Toughness Requirements, provides pressure vessel toughness requirements which are meant to protect the pressure vessel from brittle fracture and fatigue. The DBNPS Technical Specifications include Limiting Condition for Operation (LCO) 3.4.9.1, and pressure-temperature limit curves in Figures 3.4-2 (for heatup and criticality), 3.4-3 (for cooldown), and 3.4-4 (for heatup and cooldown during inservice leak and hydrostatic (ISLH) tests). This LCO, with its associated pressure-temperature limit curves meets the requirements of 10CFR50, Appendix G, thereby ensuring protection of the RCS pressure boundary integrity from nonductile failures during any anticipated operational occurrences. Consequently, reactor vessel brittle fracture is not considered to be a credible occurrence, and loss of coolant accidents (LOCA's) are limited to ruptures of primary coolant piping. The methods and criteria employed to establish the LCO and operating pressure and temperature limits are described in Babcock and Wilcox (B&W) Topical Report BAW-10046A (Reference 2).

When the low-alloy ferritic steels used in the fabrication of the DBNPS reactor vessel are exposed to fast neutrons (E > 1 MeV), their resistance to brittle fracture is decreased. More specifically, these low-alloy steels show a rise in the temperature of transition from brittle to ductile fracture (RTndt or Reference Nil Ductility Transition Temperature) and an accompanying reduction in the upper shelf impact toughness.

Because of the decrease in the non-ductile failure resistance of these low-alloy steels with increased fast neutron fluence (fast neutron flux integrated over time), the effects of fast neutron exposure must be accounted for when determining the maximum allowable stresses in the reactor vessel due to pressure and temperature transients in the RCS. In order to determine the reactor vessel's exposure and the effects of the exposure on material properties, surveillance specimens of reactor vessel material have been installed in the reactor vessel. Periodically some of these samples are withdrawn and tested to determine shifts in material properties. The DBNPS Reactor Vessel Material Surveillance Program is prepared in accordance with the requirements of 10CFR50, Appendix H, Reactor Vessel Material Surveillance Program Requirements, and ASTM Standard Practice E185-82.

B&W Report BAW-1701, Analysis of Capsule TE1-F (Reference 3), reports the results of the examination of the first material surveillance specimen capsule, removed at the end of the first DBNPS fuel cycle in accordance with the DBNPS Reactor Vessel Material Surveillance Program. Based on this examination, the material properties of the reactor vessel metal were conservatively revised to values which reflect actual expected values based on experimental data rather than the analytically predicted values previously used.

The procedures of Regulatory Guide 1.99, Rev. 2, Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials, were then used to predict the radiation induced increase in RTndt as a function of the material's copper and nickel content and neutron fluence. Previously, it was believed that the amount of phosphorus contained in low-alloy steels played a major role in determining the metal's fracture toughness. Through continuing research, it has been found that nickel, rather than phosphorus, plays a greater role in determining the fracture toughness of low-alloy steels. Consequently, nickel has replaced phosphorus in the latest revision of Regulatory Guide 1.99.

Since the plant was originally designed, research on the fracture toughness of the steam generator tubes has been completed. This work demonstrated that the tube material has a fracture toughness sufficiently high to warrant eliminating consideration of tube integrity during heatups and cooldowns. Consequently, the regions of the heatup, cooldown, and ISLH curves originally limited by the steam generator tubes have risen to the values calculated for the next limiting component.

During Mode 1 and 2 operations, the RCS is above the Reference Transition Temperature of the reactor vessel and brittle fracture is not of concern. Pressurized Thermal Shock (PTS) issues are satisfactorily addressed within the plant Emergency Procedures and are not of concern for Technical Specification 3.4.9.

For a limited region when the plant is in Mode 3, the operator, in conjunction with Administrative Controls, provides the primary means of maintaining the plant within the limits of the applicable P-T curve. In this region, the potential causes of an LTOP event, which is viewed as a low probability transient, are inadvertent initiation of High Pressure Injection (HPI) flow into the RCS and failure of the Make Up Control valve to a full open position. Inadvertent initiation of HPI flow to the RCS is not a credible event since it would require multiple failures within the Safety Features Actuation System to both establish a flow path and start the HPI pumps. Additionally, even if HPI were injected into the RCS, the maximum potential RCS pressure is the shut-off head of the HPI pump. Although this pressure exceeds the pressure limitations of the heat-up curve, it is still below the maximum allowable pressure of the ISLH curve. Since the ISLH curve maintains a significant margin of safety from the maximum pressure required to initiate or propagate a flaw, USAR analyses are unaffected and remain valid. Administrative controls will require the reactor operator to maintain the RCS pressure and pressurizer level within a region where, should the Make Up Control valve fail open, the operator would have at least ten minutes to respond to the situation without exceeding the pressure limitations of the heat-up or cooldown curves. This situation exists until the RCS temperature is high enough (about 330°F) for the Pilot Operated Relief Valve (PORV) to provide automatic overpressure protection. This method of operation is acceptable as it only occurs for a limited period during plant startup and shutdown.

During Mode 4 and Mode 5 operations, the susceptibility of low-alloy ferritic steel to brittle fracture increases dramatically. Consequently, the operating pressure limits are particularly rigorous below 280°F. The NRC first raised the issue of low temperature overpressure protection of the reactor coolant system at DBNPS in Reference 4. This letter requested that Toledo Edison describe what steps would be taken to ensure RCS pressure boundary integrity would be maintained while at low temperatures. In order to resolve this issue, LCO 3.4.2 and Figures 3.4.2a and 3.4.2b were added to the Technical Specifications in License Amendment Nos. 28 and No. 57.

As described in the NRC Safety Evaluation Report supporting License Amendment No. 57, DBNPS is protected from overpressure events at low temperatures (Modes 4 and 5) in two ways:

1. Decay Heat Removal System (DHRS) 4 inch relief valve DH4849 in the DHRS suction line provides active protection. DH4849 is put in service below 280°F by opening isolation valves DH11 and DH12, and removing power to their operators.

   This relief valve is sized such that it can fully relieve any overpressure transient condition which could occur during shutdown as long as the maximum allowable pressure is above its setpoint.

2. The plant design and operating philosophy prohibits the plant from being operated in a water solid condition, except for system hydrotests, thereby providing passive protection. A steam bubble is present in the pressurizer at RCS pressures above 30 to 50 psig. A nitrogen bubble of 30 to 50 psig is kept in the pressurizer when the pressurizer temperature can no longer sustain a 30 to 50 psig steam bubble.

   When the plant is in Mode 5 at a temperature (120°F during cooldown for the proposed curves) where the maximum allowable pressure is below the setpoint of DH4849 there is no automatic overpressure protection available. However, when an RCS temperature of 140°F is reached, the Makeup system is procedurally shutdown and the RCS pressure is procedurally reduced to and controlled at less than 75 psig. Consequently, the only potential source of overpressurization has been eliminated. This and other credible sources of overpressurization are further discussed in the following paragraphs.

Should relief valve DH4849 become inoperable, there are two postulated events which could cause a low temperature overpressurization at DBNPS. These are inadvertent actuation of the High Pressure Injection (HPI) System and the makeup control valve (MU32) failing open, with continuous makeup pump operation. Other possible causes are considered incredible in view of existing operating procedures and interlocks. The actions to be taken if DH4849 is not operable are specified in the Action Statement of Technical Specification 3.4.2 and deal with these two causes:

- HPI actuation is excluded by disabling both HPI pumps.

- Makeup control valve failure is prevented from causing RCS overpressurization by limiting the total amount of water injected into the system to the most that can be accommodated by the pressurizer. This is done by disabling automatic transfer of makeup pump suction to the Borated Water Storage Tank (BWST) on low makeup tank level, limiting the makeup tank level to 73 inches or less, and by limiting the pressurizer level and RCS pressure in the acceptable region of Figure 3.4-2a (Mode 4) or Figure 3.4-2b (Mode 5).

As discussed earlier, inadvertent actuation of HPI is not a credible source of pressurization since it involves multiple failures.

During Mode 6 operation, the RCS is open to the atmosphere and, consequently, overpressurization is not feasible.

The changes in the heatup, cooldown, and in-service leak and hydrostatic test (ISLH) curves and rates, as documented in BAW-2011 (Reference 5) incorporate the changes in the material properties due to the neutron exposure from 10 EFPY of reactor operation. Since these curves satisfactorily account for the changes in material properties, the revision of the curves does not affect the safety function of the RCS/Reactor Vessel.

In comparing the proposed curves with the existing curves, several changes are noted. The Heatup curve generally has shifted downward for any given temperature. It does remain, however, above the setpoint of DH4849 at RCS temperatures below 280°F so that during plant heatup DH4849 will provide overpressure protection as before. In the range of about 235°F to 245°F the new pressure limit is greater than the old limits. This is due to the change in the maximum allowable heatup rate from 100°F/hr to 50°F/hr and the elimination of concern about the pressure-temperature capabilities of the steam generator tubes. Previously, there was only a very limited range of temperatures (about 280 to 285°F) where Administrative Controls provided overpressure protection. This range now covers RCS temperatures from 280°F to 335°F. This is deemed acceptable based on the previous discussion of Mode 3 operations. The minimum pressure-temperature for criticality curve has shifted to the right for any given pressure due to the changes in material properties. Also, the knee in the curve has been eliminated due to the improved understanding of steam generator tube capabilities. In actuality this curve is not germane since criticality below 525°F is prohibited by Technical Specification 3.1.1.4.

The Cooldown curve has generally shifted downward in the maximum permissible pressure for a given temperature due to the increased brittleness of the vessel metal. The amount of shift at low temperatures has been minimized by lowering the maximum allowable cooldown rate from 100°F/hr to 50°F/hr when below 270°F. At or above 270°F the cooldown rate remains at 100°F/hr, as before. The allowable pressure during cooldown above about 250°F increased due to the elimination of steam generator tube P-T concerns. The range of temperatures (from 280°F to about 375°F) where overpressure protection is provided by Administrative Controls has increased (previously from 280°F to 360°F). This change is acceptable since the amount of time spent in this range is limited and the Administrative Controls provide appropriate overpressure protection against a makeup system failure. Since the maximum allowable pressure is well above the shutoff head of the HPI pumps, this event is not of concern. Also, as noted above, the actual allowable pressure in this range is higher than previously allowed so that operators have more time to respond to a pressure increase prior to the limit being exceeded. The range where the maximum allowable pressure is below the setpoint of DH4849 has increased from less than about 80°F to less than 120°F. However, as discussed above, the Administrative Controls on plant operation in this temperature range make this increase acceptable.

The ISLH curve has generally shifted downward in allowable pressure for any given temperature for the same reasons the heatup and cooldown curves have shifted, as discussed above. This is because it represents the lowest allowable pressure from either the heatup curve or the cooldown curve, as further modified for a safety factor of 1.5, as specified in the ASME Code, Appendix G. The split in the curves has been eliminated due to the removal of concerns over steam generator tube fracture toughness.

The Reactor Vessel Material Surveillance Program which is used to determine the material properties of the reactor vessel has been evaluated by the NRC and found acceptable. In a letter dated May 8, 1985 from H. L. Thompson, Jr. of the NRC to H. B. Tucker, Chairman of the B&W Owners Group (BWOG), the NRC recommended removing specific details of this Surveillance Program from the Technical Specifications. The proposed changes do not affect the safety function of the RCS/Reactor Vessel since the currently approved requirements will be retained by referencing a document approved by the NRC.

The changes proposed for the RCS Pressure versus Pressurizer level curves are being made to reflect the change in the P-T curves. Since they are calculated based on the 10 EFPY curves, as documented in B&W Document No. 32-1170349-00, they preserve the same conditions as the previous curves for operation up to 5 EFPY. Therefore, they help maintain the safety function of the Reactor Vessel.

It should be noted that the curves to be included in the Technical Specification do not include the effects of instrument string uncertainty. The curves which are to be put into the operating procedures will be modified so as to conservatively correct for these uncertainties.

The revision of the Operating License Condition 2.C.(3)(d) is administrative in nature and does not directly affect the safe operation of the plant. In preserving the License Condition by requiring a review of LTOP concerns before the end of 10 EFPY ensures that Toledo Edison will again evaluate the effects of reactor vessel neutron embrittlement and take whatever action is necessary to prevent overpressurization of the RCS.

The revision to Technical Specification 3/4.4.2, Action A.2.b.2, is administrative in nature and changes the figure numbers of Figures 3.4.2-a and 3.4.2-b to 3.4-2a and 3.4-2b, respectively, to make these figure numbers consistent with other figures in the Technical Specifications.

SIGNIFICANT HAZARDS CONSIDERATION

The Commission has provided standards in 10CFR50.92(c) for determining whether a significant hazards consideration exists. A proposed amendment to an Operating License for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: 1) involve a significant increase in the probability or consequences of an accident previously evaluated; 2) create the possibility of a new or different kind of accident from any accident previously evaluated; or 3) involve a significant reduction in the margin of safety. Toledo Edison has reviewed the proposed change and determined that:

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because: 1) the deletion of the reactor vessel material surveillance requirements and surveillance specimen withdrawal schedule from the Technical Specifications is administrative since the verification of material properties is governed by BAW-1543A, which has been approved for this purpose by the NRC, 2) the revision to the pressure/temperature curves and the pressurizer level/RCS pressure limit curves for periods when DH4849 is inoperable will provide the same level of protection of the RCS pressure boundary integrity as was previously evaluated, 3) the revision to License Condition 2.C(3)(d) is administrative to reflect the validity of the present analyses to 10 EFPY and does not affect any previously analyzed accidents, and 4) the requested revision to the figure numbers of Technical Specification 3/4.4.2, Action A.2.b.2, is administrative to maintain numbering consistency in the Technical Specifications and does not affect any previously analyzed accidents. (10CFR50.92(c)(1))

2. The proposed changes do not create the possibility of a new or different type of accident because: 1) the Davis-Besse Reactor Vessel Material Surveillance Schedule is still controlled by and meets the requirements of 10CFR50, Appendix H, 2) the revision to the RCS pressure/temperature curves and the pressurizer level/RCS pressure limit curves provides adequate protection against Reactor Vessel failure due to brittle fracture concerns under all postulated circumstances, 3) the change to License Condition 2.C(3)(d) is administrative to reflect present analyses to 10 EFPY and does not affect any equipment or activities in plant operation to 10 EFPY, and 4) the requested revision to the figure numbers of Technical Specification 3/4.4.2, Action A.2.b.2, is administrative to maintain numbering consistency in the Technical Specifications and does not affect any equipment or activities in plant operation. (10CFR50.92(c)(2))

3. The proposed changes do not involve significant reduction in a margin of safety because: 1) the basis for the Reactor Vessel Material Surveillance Program is unchanged and will still meet 10CFR50, Appendix H requirements, 2) the revision to the RCS pressure/temperature curves and pressurizer level/RCS pressure limit curves maintain the present margin of safety from Reactor Vessel brittle fracture as required by 10CFR50, Appendix G, 3) the change to License Condition 2.C(3)(d) is administrative and does not affect any analyses which provide the basis for the Technical Specifications, and 4) the requested revision to the figure numbers of Technical Specification 3/4.4.2, Action A.2.b.2, is administrative to maintain numbering consistency in the Technical Specifications and does not affect any analyses which provide the basis for any Technical Specification. (10CFR50.92(c)(3))

CONCLUSION

Based on the above, Toledo Edison has determined that the proposed changes do not involve a significant hazard consideration.

REFERENCES

1. B&W Owners Group Material Committee Report BAW-1543A, Rev. 2, Integrated Reactor Vessel Materials Surveillance Program, February 1984, approved by NRC May 1985.

2. BAW-10046A, Methods of Compliance with Fracture Toughness and Operational Requirements of Appendix G to 10 CFR 50, October 1975.

3. BAW-1701, Analysis of Capsule TE1-F, January 1982.

4. Letter dated October 1, 1976, from John F. Stolz (NRC) to Lowell E. Roe (TECo) (Log No. 121).

5. BAW-2011, Pressure Temperature Limits for 10 EFPY, November, 1987.

6. ASTM Standard Practice E185-73, "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels (ANSI/145-73)"

7. Regulatory Guide 1.99, Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials"

8. Davis-Besse Nuclear Power Station, Unit No. 1 License Amendment No. 28 (Log No. 585, dated July 25, 1980)

9. Davis-Besse Nuclear Power Station, Unit No. 1 License Amendment No. 57 (Log No. 1277, dated May 5, 1983)

10. B&W Document No. 32-1170349-00, TED Over Pressure Protection Tech Specs

Docket No. 50-346
License No. NPF-3
Serial No. 1490
Attachment 3

REVISION OF LICENSE CONDITION 2.C(3)(d)
OF DAVIS-BESSE NUCLEAR POWER STATION,
UNIT NO. 1 FACILITY OPERATING LICENSE NPF-3

Presently, License Condition 2.C(3)(d) states:

Prior to operation beyond five Effective Full Power Years, the Toledo Edison Company shall provide to the NRC a reanalysis and proposed modifications, as necessary, to ensure continued means of protection against low temperature reactor coolant system overpressure events.

Serial No. 1490 proposes revising this License Condition to reflect operation to ten Effective Full Power Years as follows:

Prior to operation beyond ten Effective Full Power Years, the Toledo Edison Company shall provide to the NRC a reanalysis and proposed modifications, as necessary, to ensure continued means of protection against low temperature reactor coolant system overpressure events.