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Subject: Revision to Technical Specification Bases to Reflect Cycle 7 Core Reload (TAC Number M75850)

Gentlemen:

The attached provides changes to the Davis-Besse Nuclear Power Station (DBNPS), Unit Number 1, Operating License, Appendix A Technical Specification Bases which will result from the core reload for Cycle 7 during the sixth refueling outage. The Bases sections affected are Section 2.1.1 and 2.1.2, Reactor Core; 2.2.1, Reactor Protection System Instrumentation Setpoints; 3/4.2, Power Distribution Limits; 3/4.2.5, DNB Parameters; and 3/4.4.1, Reactor Coolant Loops.

The changes to the Bases sections cited above reflect analyses contained in BAW-2096, November 1989, "Davis-Besse Nuclear Power Station Unit 1, Cycle 7 Reload Report" (Attachment 3). Specifically, the changes to the Bases incorporate the BWC Critical Heat Flux correlation, in addition to the B&W-2 correlation previously referenced, for Departure from Nucleate Boiling Ratio limit determination and the deletion of the references to Batch 1 fuel which no longer resides in the core. Other changes presented in the Reload Report will be incorporated in the Core Operating Limits Report. Utilization of the Core Operating Limits Report has been approved by the Nuclear Regulatory Commission in License Amendment Number 144, dated January 11, 1990.

In addition, an administrative correction is being made to Bases Section 2.2.1, "Reactor Protection System Instrumentation Setpoints - RC Pressure Low, High, and Pressure Temperature," to reflect the Reactor Coolant System (RCS) high pressure trip setpoint of 2355 psig to be consistent with the Amendment Number 128 (Log Number 2795, dated December 28, 1988) authorized increase in the RCS high pressure trip setpoint from 2300 psig. As the safety evaluation contained in Serial Number 1464 (dated February 1, 1988) and the Safety Evaluation Report accompanying the License Amendment provided the basis and justification for this change, a further evaluation for this correction is not included herein.

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Attachment 1 provides the Technical Assessment of the Reload Report and supporting Bases changes. Attachment 2 provides the marked-up Technical Specification Bases pages which incorporate the changes cited above and in the Reload Report.

It is requested that these changes be issued by May 1, 1990 to facilitate startup following the sixth refueling outage.

If you have any questions or comments, please contact Mr. R. W. Schrauder, Manager - Nuclear Licensing at (419) 249-2366.

Very truly yours,

RMC/ssg

Attachments

cc: P. M. Byron, DB-1 NRC Senior Resident Inspector
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# Technical Assessment

## DESCRIPTION OF PROPOSED ACTIVITIES

Based on a 10CFR50.59 review of the Cycle 7 core reload report, this submittal proposes revisions to Technical Specification (TS) Bases Sections 2.1.1 and 2.1.2, Reactor Core; 2.2.1 Reactor Protection System Instrumentation Setpoints; 3/4.2, Power Distribution Limits; 3/4.2.5, DNB Parameters; and 3/4.4.1, Reactor Coolant Loops for the Cycle 7 core reload which includes the loading of new fuel assemblies (FAs) and burnable poison rod assemblies (BPRAs) and the shuffling of FAs to facilitate nuclear power generation presented in BAW-2096, November 1989, "Davis-Besse Nuclear Power Station Unit 1, Cycle 7 Reload Report" (Attachment 2). The reference cycle for this reload report is Cycle 6. The design and methodology for the Cycle 7 analysis include other changes such as the use of the Mark-B8A FA, the use of anti-straddle lower end fittings, reduced fuel rod pre-pressure and the use of the BWC Critical Heat Flux (CHF) correlation.

## SYSTEMS AFFECTED

Reactor Core

## SAFETY FUNCTION OF THE SYSTEMS AFFECTED

The safety function of the Reload Report and the affected Core Operating Limits Report (COLR) is to ensure operation of the core within safety limits. The function of the core is to generate power for a specified duration. The impact on safety is the assurance of the arrangement of the core such that safety limits are not violated.

## EFFECTS ON SAFETY

The referenced cycle for the nuclear and thermal hydraulic design of Cycle 7 is Cycle 6. The Cycle 7 physics parameters are based on a 387 effective full power day (EFPD) Cycle 6 length including Axial Power Shaping Rod (APSR) withdrawal and coastdown. Safety analyses for Cycle 7 are valid for a Cycle 6 length of 402 EFPD. Core follow monitoring and analyses ensured that there have been no anomalies during Cycle 6 which would adversely affect fuel performance during Cycle 7 as designed. Consistent with previous cycles, cross-core shuffling of fuel assemblies is minimized in this cycle. The Cycle 7 design is characterized by only 12 FAs being cross-core shuffled so as to minimize any carryover effects from radial power differences accrued from previous cycles.

The design length of Cycle 7 is 415 EFPD. Cycle 7 reactivity will be controlled by 60 new BPRAs (of the same physical design as used in Cycle 6) located in the fresh fuel, soluble boron and the same 53 full-length Ag-In-Cd control rod assemblies (CRAs) used in previous cycles. The Cycle 7 loading includes 60 new FAs (Batch 9) at 3.38 w/o U-235. This feed batch loading, characterized as Batch 9, is comprised of the Mark-B8A design. The difference between the Mark-B8A design and the Mark-B5A design currently in use are described below.

The Mark-B8A FA is an improved Mark-B5A FA with design features to permit easy reconstitution if defective fuel rods are experienced, allow for high burnup and to provide protection against debris fretting damage to the fuel rods. To permit easy reconstitution, the upper end fitting (UEF) is designed to be removable in the spent fuel pool. The top spacer grid has been enhanced to provide extra stiffness and the top spacer skirt has been removed. To provide for high burnups, the lower end fitting (LEF) has been shortened by approximately 0.7 inches and the guide and instrument tubes lengthened by approximately 0.7 inches. The fuel rod was lengthened by 0.4 inches. These changes provide sufficient growth room for the fuel rod to preclude its contact with the UEF and LEF at high burnup. Also, for higher burnup capability, the cold worked guide tubes previously employed have been replaced with annealed guide tubes that grow at a slower rate as a function of burnup. To prevent misalignment when inserting FAs into the core, anti-straddle bars have been added to the lower end fitting. These bars prevent the FA from being fully inserted unless it is in the proper position over the lower grid assembly of the core support internals.

To protect against debris induced fretting failure of the fuel rod, the solid portion of the lower end plug was extended in length. Also, the lower spacer grid location was lowered so that the solid end plug extends through the lower spacer grid. This change is intended to trap any debris capable of fuel rod fretting below the bottom spacer grid. The solid lower end plug will maintain the rod's integrity since debris wear of the solid plug would not breach the rod's pressure boundary.

The solid end plug, however, results in a loss of plenum volume which would result in a higher fuel rod pressure with burnup and associated fission gas release. Part of the gas volume reduction was offset by the 0.4 inch increase in rod length. In addition, in order to obtain similar mechanical and thermal performance, when compared with the Mark-B5A, the fill gas pressure (prepressure) was also reduced by approximately 100 psi. This initial prepressurization has been verified by B&W, and independently by Toledo Edison, to not result in excessive clad creepdown and its resulting clad strain.

The Mark-B8A design incorporates zircaloy in place of inconel for the six intermediate spacer grids. This was done to improve uranium utilization as a result of zircaloy's smaller thermal neutron absorption cross section. The zircaloy spacer grid design was approved by the NRC in 1984 and, as required by the NRC Safety Evaluation Report (SER) for the use of the zircaloy design (Rancho Seco Nuclear Generating Station - Evaluation of Mark BZ Fuel Assembly Design, U. S. Nuclear Regulatory Commission, Washington, D. C., November 16, 1984), a combined seismic and Loss of Coolant Accident (LOCA) loads analysis has been completed (Attachment 4).

The Mark-B8A design has been or will be used at the following plants: Oconee 2 Cycle 11 which commenced in June 1989, Oconee 3 Cycle 12 which is scheduled to start in January 1990, and TMI-1 Cycle 7 which is scheduled to start in March 1990.

The analytical methods used to calculate the Cycle 7 core physics parameters are unchanged from Cycle 6. Toledo Edison has independently verified Cycle 7's fuel cycle and physics design.

For the thermal-hydraulic design of Cycle 7 there were no changes in the analytical methods. The introduction of the zircaloy grids, which have a higher pressure drop, was investigated for the possibility of a "transition core penalty". The higher pressure drop has the effect of reducing the flow to the zircaloy grid assemblies' (Mark-B8A) fuel rods. The transition core was determined to be bounded by previous analysis which used a higher bypass flow assumption than exists in Cycle 7. A higher bypass flow also has the effect of reducing the flow to those assemblies' fuel rods. Thus, no penalty is required. Also, though not new methodology, the use of zircaloy grids requires the use of the BWC critical heat flux correlation instead of B&W-2. This correlation was previously approved by the NRC with the 1984 incorporation of the zircaloy spacer grid design (BAW-10143P-A, "BWC Correlation of Critical Heat Flux," April 1985).

All accidents analyzed in Chapter 15 of the Updated Safety Analysis Report (USAR) have been reexamined with respect to Cycle 7 parameters to ensure that the thermal performance during the hypothetical transients has not been degraded. The hot full power moderator and Doppler coefficients remain negative such that Cycle 7 is bounded for the main steam line break or any other over-cooling transient. The radiological dose consequences of the USAR Chapter 15 accidents have been evaluated using conservative radionuclide source terms that bound the cycle specific source term for Cycle 7 and future reload cycles. The results of the dose evaluations show that the offsite radiological doses for each accident are below the respective acceptance criteria values in the current NRC Standard Review Plan (SRP), NUREG-0800. The use of the NRC SRP acceptance criteria was accepted by the NRC in the NRC SER for Cycle 6 (Log Number 2719 dated October 3, 1988). The bounding values for the allowable LOCA maximum linear heat rates (LHRs) have been reanalyzed for Cycle 7. The reanalysis was submitted to the NRC by letter Serial Number 1720 dated October 16, 1989.

Cycle 7 is the first cycle to use the Core Operating Limits Report (COLR). Currently, the COLR does not include Section 2 of the TS, i.e., the safety limits. There are no changes to the safety limits for Cycle 7. The COLR figures for rod index and axial power imbalance and the table for quadrant power tilt will be updated to reflect the Cycle 7 values. The slight differences from the Cycle 6 TS values are to account for small changes in power peaking and control rod worths.

This reload also requires several TS Bases changes. Except for one change to delete a reference to batch 1, since no reinserted FAs are used in Cycle 7, all Bases changes are to include the use of the BWC critical heat flux correlation. As stated above, the BWC correlation is to be used for the Batch 9 fuel due to the zircaloy spacer grids.

The planned startup physics test program is unchanged in scope from Cycle 6.

## UNREVIEWED SAFETY QUESTION EVALUATION

The proposed action would not increase the probability of occurrence of an accident previously evaluated in the USAR because the probability of any accident which is presently analyzed in the Davis-Besse USAR is independent of the core loading pattern and is also independent of the other modifications discussed above. (10CFR50.59(a)(2)(i))

The proposed action would not increase the consequence of an accident beyond the present acceptance criteria previously evaluated in the USAR. All accidents have been reviewed to ensure that they are bounded by the existing analysis. The results of the USAR dose evaluations performed showed that the offsite radiological doses for each accident in the USAR were below the respective acceptance criteria values in the current NRC Standard Review Plan, NUREG-0800, as accepted by the NRC in the Cycle 6 SER (Log Number 2719 dated October 3, 1988). (10CFR50.59(a)(2)(i))

The proposed action would not increase the probability of occurrence of a malfunction of equipment important to safety because the probability of a malfunction of equipment is independent of the core loading pattern and is also independent of the other modifications discussed above. (10CFR50.59(a)(2)(i))

The proposed action would not increase the consequence of a malfunction of equipment important to safety beyond the present acceptance criteria because the results of the USAR dose evaluations performed showed that the offsite radiological doses for each accident in the USAR were below the respective acceptance criteria values in the current NRC Standard Review Plan, NUREG-0800, as accepted by the NRC in the Cycle 6 SER (Log Number 2719 dated October 3, 1988). (10CFR50.59(a)(2)(ii))

The proposed action would not create a possibility for an accident of a different type than any evaluated previously in the USAR because there have been no hardware changes or design modifications which would create the possibility for an accident of a different type than previously evaluated. (10CFR50.59(a)(2)(ii))

The proposed action would not create a possibility for a malfunction of equipment of a different type than any evaluated previously in the USAR because there have been no hardware changes or design modifications which would create the possibility for a malfunction of equipment of a different type than previously evaluated. (10CFR50.59(a)(2)(ii))

The proposed action would not reduce the margin of safety as defined in the basis for the Technical Specifications because, where required, the COLR will be changed to ensure the margin of safety. (10CFR50.59(a)(2)(iii))

## CONCLUSION

Pursuant to the above, it is concluded that the changes proposed as described in the Reload Report do not involve an unreviewed safety question.