Docket No. 50-346
License No. NPF-3
Serial No. 1516
Enclosure

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. 1516) concern:

Technical Specification 2.0, Safety Limits and Limiting Safety System Settings;
Technical Specification 3/4.1, Reactivity Control Systems;
Technical Specification 3/4.2, Power Distribution Limits;
Technical Specification 3/4.3, Instrumentation;
Technical Specification 3/4.4, Reactor Coolant System;
Technical Specification 3/4.5, Emergency Core Cooling Systems (ECCS);
Technical Specification Basis 3/4.1, Reactivity Control Systems;
Technical Specification Basis 3/4.5, Emergency Core Cooling Systems (ECCS); and
Technical Specification 5.0, Design Features.

By: T. J. Myers, Nuclear Licensing Director

For: D. C. Shelton
Vice President, Nuclear

Sworn and subscribed before me this 18th day of May, 1988.

Laurie A. Hinkle
Notary Public, State of Ohio

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

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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, Appendix A, Technical Specifications

A. Time Required to Implement: This change will be implemented prior to entry into Mode 2 from the current refueling outage, presently scheduled for September 2, 1988.

B. Reason for Change (FCR No. 84-0067 Rev. B): Cycle 6 Reload Report.

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

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

E. Babcock & Wilcox Topical Report BAW-2038, "Davis-Besse Nuclear Power Station, Unit No. 1, Cycle 6-Reload Report", April 1988 (Attachment No. 3)

SAFETY EVALUATION

DESCRIPTION OF PROPOSED ACTIVITIES

This License Amendment application proposes the loading of new fuel assemblies (FAs) and burnable poison rod assemblies (BPRAs), the shuffling of FAs and control rod assemblies (CRAs) and the replacement of eight "black" axial power shaping rods (APSRs) to facilitate nuclear power generation for Cycle 6 in accordance with the limits and analysis presented in BAW-2038, April 1988, Davis-Besse Nuclear Power Station, Unit No. 1, Cycle 6-Reload Report (Attachment 3). The reference cycle for this reload report is Cycle 5. The design and methodology for the Cycle 6 analysis include other changes such as a reduced physics testing program, a new power imbalance detector correlation (PIDC) method, the elimination of the regenerative neutron sources, a different computer code for generating the core physics parameters and different computer codes for the thermal-hydraulic analysis. This License Amendment application also proposes to revise several Technical Specifications to reflect the above changes.

SYSTEMS AFFECTED

Reactor Core

Reactor Protection System Setpoints

CRAs

DOCUMENTS AFFECTED

USAR Appendix 4B and Figure 7.2-2

Technical Specifications 3.1.2.8, 3.1.2.9, 3.1.3.6, 3.1.3.9, 3.2.1, 3.2.4, 3.2.5, 3.4.1.1, 3.5.4, 4.1.2.8, 4.1.2.9, 4.2.5.1, 4.4.1.2, 4.5.4, 5.3.2

Technical Specifications Figures 2.1-1, 2.1-2, 2.2-1, 3.1-1, 3.1-2a, 3.1-2b, 3.1-2c, 3.1-2d, 3.1-3a, 3.1-3b, 3.1-3c, 3.1-3d, 3.1-4, 3.1-5a, 3.1-5b, 3.1-5c, 3.1-5d, 3.1-5e, 3.1-5f, 3.1-5g, 3.2-1 (new), 3.2-1a, 3.2-1b, 3.2-1c, 3.2-1d, 3.2-2 (new), 3.2-2a, 3.2-2b, 3.2-2c, 3.2-2d

Technical Specification Tables 2.2-1, 3.2-1, 3.2-2, 4.3-1

Technical Specification Bases 2.1.1, 2.1.2, 2.2.1, 3/4.1.2, 3/4.5.4, Figure 2.1

SAFETY FUNCTION OF THE SYSTEMS AFFECTED

The safety function of the Reload Report and the affected Technical Specifications is to ensure operation of the core within safety limits. The safety function of the Reactor Protection System (RPS) setpoints is to trip the reactor when an unsafe condition is approached. The safety function of the CRAs is to provide SCRAM and shutdown margin capability and to maintain peaking within acceptable limits. The function of the core is to generate power for a specified duration. The impact on safety is to arrange the core such that safety limits are not violated.

EFFECTS ON SAFETY

The reference cycle for the nuclear and thermal-hydraulic design of Cycle 6 is Cycle 5. The Cycle 6 physics parameters are based on a 400 effective full power day (EFPD) Cycle 5 length including APSR withdrawal and coastdown. There have been no anomalies during Cycle 5 which would adversely affect fuel performance during Cycle 6 as designed. Consistent with previous cycles, cross-core shuffling of fuel assemblies is minimized in this cycle. The Cycle 6 design is characterized by only 16 FAs being cross core shuffled so as to minimize any carryover effects from tilts encountered in previous cycles.

The Cycle 6 loading includes 64 new FAs (Batch 8) at 3.13 w/o U-235 and the reinsertion of one (Batch 1) previously discharged FA. This loading, characterized as Batch 8, is comprised of the MK-B5 design which is the same as the Batch 7 design currently in use. Due to the design length of Cycle 6 (405 EFPD), additional reactivity is necessary. This increased reactivity will be controlled, in part, by 64 new BPRAs (of the same physical design as used in Cycle 5) located in the fresh fuel. The reactivity is also controlled by soluble boron and 53 full-length Ag-In-Cd CRAs. These CRAs are the same ones used in previous cycles. However, the rod group designations differ from Cycle 5 in order to increase the worth of group 4 to facilitate control during physics testing and to decrease the worth of group 7 to be compatible with the new gray APSR imbalance control capability.

The gray APSRs are of the same design as those previously approved by the NRC and currently in use at other B&W operating plants but are a change from the black APSRs used previously at Davis-Besse. APSRs are used for the additional control of the axial power distribution. The poison material in the black APSRs is a 36" long rod composed of Ag-In-Cd. The poison material in the gray APSRs is a 63" long Inconel 600 rod. Due to the reduced neutron absorbing capability of Inconel, the gray APSRs provide a more even flux shape which reduces the axial power peak and brings about a reduction in the local change in the kw/ft during APSR movement. This helps minimize the duty placed on the fuel, thus reducing the chances of failed fuel.

The gray APSRs are designed for improved creep life in order to extend their life. The APSRs cladding stress and strain have been analyzed and shown to have sufficient cladding and weld stress margins and no cladding strain is induced due to thermal expansion or irradiation swelling of the Inconel. Therefore, Toledo Edison believes the gray APSRs to be both safe and an improvement over the black APSRs.

The Cycle 6 core physics parameters represent a change in the analytical methods from the previous cycle(s). The Cycle 5 parameters were generated using the PDQ07 (Babcock & Wilcox Version of PDQ Users Manual BAW-10117 P-A) computer code, whereas, the NOODLE code (See topical report BAW-10152A) was used for Cycle 6. The calculational differences between the models are negligible when compared to measured data as shown in Babcock and Wilcox (B&W) topical reports BAW-10152A NOODLE, a Multi-Dimensional Two-Group Reactor Simulator and BAW-10120 Comparison of Core Physics Calculations with Measurement. The differences in the Cycle 6 parameters when compared to Cycle 5 are attributable to the increased cycle lifetime, the increase in the BPRA poison concentration, the variation in the loading pattern, the second transition cycle to the BPRA low leakage core design, the revised control rod groupings and the replacement of the black APSRs with gray APSRs. Since the gray APSRs are non-trippable CRAs, they are not explicitly used in the shutdown margin calculation. Implicit in the calculation is the axial power profile which they influence.

The last change in the nuclear design of Cycle 6 is the removal of the two regenerative neutron sources (RNS) from the core. This was addressed in a separate 10CFR50.59 review (FCR 85-0100) where it was determined that after three cycles of operation the RNS are no longer necessary. This is due to the presence of a sufficient number of neutron emitters naturally produced during the normal burnup process being able to produce the 0.5 counts per second reading at the source range detectors before approach to criticality. This is consistent with the requirements of Revision 2 to Reg. Guide 1.68. Toledo Edison believes this to be safe for Cycle 6 and all future cycles.

The thermal-hydraulic design of Cycle 6 also represents a change in the analytical methods. Cycle 6 was analyzed using the LYNX1 Reactor Fuel Assembly Thermal Hydraulic Analysis Code (BAW-10129-A), LYNX2 Subchannel Thermal Analysis Program (BAW-10130-A), and LYNXT Core Transient Thermal Hydraulic Program (BAW-10156-A) crossflow codes which can predict the flow redistribution effects in an open lattice reactor core. The crossflow methodology provides significant DNBR improvements (by allowing the coolant to mix) over the traditional closed-channel methodology. Because of the new methodology, the reactor coolant flow, bypass flow and design axial flux shape were revised for the Cycle 6 analysis. The reactor coolant flow requirement, although approximately equal to the existing (Cycle 5) Technical Specification requirement, is based on a bounding analysis using 52 BPRAs. The Cycle 6 core design includes 64 BPRAs. This is bounded by the 52 BPRA case since 64 BPRAs result in less core bypass flow, i.e., more coolant to the fuel than with 52 BPRAs. In addition, with use of the LYNX analysis the design axial peak was increased from 1.50 to 1.65. The RPS pressure-temperature trip setpoint has been recalculated based on pressure-temperature limits computed with LYNXT. The flux/flow setpoint was raised from 1.07 to 1.08 based on the limit determined with LYNXT.

Also, reanalysis of the locked rotor transient, chosen since it produces the worst DNBR, shows that the minimum DNBR for this event is greater than 1.30. Toledo Edison believes this change to be not only safe but a more accurate representation of the actual conditions within the core.

All accidents analyzed in Chapter 15 of the Updated Safety Analysis Report have been reexamined, with respect to Cycle 6 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 6 is bounded for main steam line break or any other over-cooling transient. The radiological dose consequences of the SAR

Chapter 15 accidents have been evaluated using conservative radionuclide source terms that bound the cycle specific source term for the longer Cycle 6 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 NUREG-0800.

The bounding values for the allowable LOCA maximum linear heat rates (LHRs) were reviewed for Cycle 6. Based on an additional calculation at the 6 foot elevation for Davis-Besse, the values previously utilized for Cycle 5 at the 4, 8 and 10 foot elevations have been reviewed and found to be acceptable as previously reviewed by the NRC for Cycle 5 at Davis-Besse.

The pertinent Technical Specifications in the Reload Report have been revised for Cycle 6 operation to account for changes in power peaking and control rod worths. The Technical Specifications have also been revised to incorporate the changes due to the crossflow methodology previously discussed. Additional Technical Specifications being revised as a part of the Reload Report, either for clarification or as a result of additional analysis, are discussed below in order of occurrence in the Technical Specifications.

First, the shutdown borated water sources Technical Specification 3.1.2.8 was reviewed. The existing boron capability requirement to provide a 1% shutdown margin while cooling from 200°F appeared excessively high. Therefore, this requirement was reanalyzed for a cooldown from 200°F to 70°F with the results as presented in the proposed Technical Specification. In addition, prior to the reanalysis it became evident that the contained volume in the borated water storage tank (BWST) is not the same as the available volume in the tank because of the location of the suction piping in the tank. Although previously any unavailable water volume was calibrated out, the word "contained" is being changed to "available" with an explanation added in BASES 3/4.5.4 in order to avoid any future confusion. The remaining Technical Specifications 3.1.2.9, 3.5.4 and Surveillance Requirements associated with the BWST are also being changed to reflect the above.

The quadrant power tilt (QPT) limit Technical Specification 3.2.4 for Cycle 6 incorporates methods used to revise the power imbalance detector correlation (PIDC). (This revision to the PIDC method is addressed in a subsequent paragraph). The QPT alarm setpoint was widened by reducing the incore system error by utilizing the statistical combination of the incore system measurement and observability errors methodology. This methodology (See Letter: J. H. Taylor, B&W to H. R. Denton, NRR dated July 30, 1986) and the new PIDC, initiated thru the B&W Owners Group (BWOG), have been previously approved by the NRC and have already been safely implemented at other B&W facilities.

Another change to the Technical Specification is the introduction of the power dependent QPT limit. QPT has the effect of increasing the power in one quadrant of the core while decreasing it in another. The FAs with the increased power are the ones of interest. As the tilt, i.e., the power in the FA increases, the margin to DNB decreases. Clearly, the higher the initial power in the FA, the less tilt it can tolerate. Cycle 6, has been analyzed using a break point at 50%FP to take advantage of the lower power and to make low power operation easier. The wider QPT limit is valid for reactor power of 50%FP or less. The tighter QPT limit is valid at any licensed power. This change is considered by Toledo Edison to be safe and of benefit to operations by avoiding unnecessary entries into an action statement.

The remaining changes to the QPT and DNB Technical Specifications are administrative in nature. Currently, the two tables in the Technical Specifications are correctly referenced but are numerically out of sequence. To correct this, Table 3.2-2, Quadrant Power Tilt Limits is being re-numbered as Table 3.2-1. Similarly, Table 3.2-1, DNB Margin, is being re-numbered as Table 3.2-2 and the references changed accordingly.

The PIDC test calibrates the excore detectors to the incore detector measurements of core offset (core offset equals the ratio of the core power in the top of the core minus the core power in the bottom of the core to the actual core thermal power). Previously, the relationship between the excore and the incore offset was conservatively set with the excore detectors biased high. The new PIDC correlates the excore detectors more closely to the incore detectors by removing the unnecessary conservatism. This is acceptable based on the consequent tightening of the excore detector recalibration criteria as noted below. In order to preserve the required measurement system error allowance which results from reducing the conservatism, the criteria which required the excore detectors to be recalibrated has been tightened to require the excore detector offset to be within 2.5%FP of the incore detector offset (down from 3.5%FP). This is reflected in Technical Specification Table 4.3-1.

Table 4.3-1 is also being revised to increase the power level at which it is required to compare the excore detector offset to the incore detector offset. Using the same reasoning as for the QPT, imbalance has the effect of increasing the power in either the top or the bottom of the core. The lower the initial power the greater the imbalance that can be tolerated. Below 50%FP the maximum imbalance which can be induced produces peak powers within acceptable limits. Therefore, below 50%FP it is acceptable not to calibrate the excore detectors to trip the reactor on imbalance.

Technical Specification 3.4.1.1, pertains to reactor coolant circulation. The action statement, which restricts thermal power for three reactor coolant pumps (RCPs) to 79.7%, is being increased to 80.6%. This is acceptable as a result of the crossflow methodology. The wording changes clarify this section of the Technical Specification. When only three RCPs are operating the corresponding flux/delta flux/flow limit curve is reduced, by the Reactor Protection System (RPS), by the ratio of the new three RCPs measured flow to the old four RCPs measured flow. This means the actual limit could be higher than shown in Technical Specification 2.2.1 if the actual flow is greater than the flow assumed in the example shown. This does not represent a change in the manner in which the RPS responds, it is only a clarification. Also, surveillance requirement 4.4.1.2 which refers back to the action statement is being revised. This is justified as follows. Following removal of one RCP from service the high flux limit is manually reduced and the RPS automatically reduces the flux/delta flux/flow curve to a new limit. This reduction in the two setpoints is to be verified within four hours to ensure the reactor will trip when required with three RCPs running. When the fourth RCP is put back into service the four hour verification is not required. If the high flux limit is not increased or if the RPS fails to readjust the limit curve upward, an unsafe condition would not exist as power is increased. This is because of the fact that the reactor would trip before full power could be reached. If the RPS responds as designed for the flux/delta flux/flow limit, there is no reason to assume the gain settings would change from their original four RCPs setting. Therefore, the limit curve would return to the original values. If for any reason the gain settings are adjusted while operating with three RCPs, procedures require the gain settings to be verified upon return to four RCPs. Consequently, the wording of the surveillance requirement is being changed to clarify that the four hour requirement applies only when the total number of running RCPs is reduced from four to three and is not required when the number of running RCPs is increased from three to four.

The planned startup physics test program has been reduced in scope from that performed for previous cycles. As a result of a Davis Besse undertaking with the BWOG, the zero power ejected rod worth and all-rods-in temperature coefficient measurements and one of the intermediate core power distribution test plateaus have been eliminated. The reduced scope physics testing as previously reviewed by the NRC has been utilized at other B&W facilities and has proven to be sufficient to demonstrate that the core will perform within the assumptions of the safety analysis.

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 (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. Also, there have been no mechanical changes made to the FAs, CRAs or BPRAs that could increase the probability of their malfunction (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 (10CFR59.59(a)(2)(i)).

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 Technical Specifications have been 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.

SIGNIFICANT HAZARDS CONSIDERATION

DESCRIPTION OF PROPOSED ACTIVITIES

This License Amendment application proposes the loading of new fuel assemblies (FAs) and burnable poison rod assemblies (BPRAs), the shuffling of FAs and control rod assemblies (CRAs) and the replacement of eight "black" axial power shaping rods (APSRs) to facilitate nuclear power generation for Cycle 6 in accordance with the limits and analysis presented in BAW-2038, April 1988, "Davis-Besse Nuclear Power Station, Unit No. 1, Cycle 6-Reload Report" (Attachment 3). The reference cycle for this reload report is Cycle 5. The design and methodology for the Cycle 6 analysis include other changes such as a reduced physics testing program, a new power imbalance detector correlation (PIDC) method, the elimination of the regenerative neutron sources, a different computer code for generating the core physics parameters and different computer codes for the thermal-hydraulic analysis. This License Amendment application also proposes to revise several Technical Specifications to reflect the above changes.

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Reactor Core

Reactor Protection System Setpoints

CRAs

DOCUMENTS AFFECTED

USAR Appendix 4B and Figure 7.2-2

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Technical Specifications Figures 2.1-1, 2.1-2, 2.2-1, 3.1-1, 3.1-2a, 3.1-2b, 3.1-2c, 3.1-2d, 3.1-3a, 3.1-3b, 3.1-3c, 3.1-3d, 3.1-4, 3.1-5a, 3.1-5b, 3.1-5c, 3.1-5d, 3.1-5e, 3.1-5f, 3.1-5g, 3.2-1 (new), 3.2-1a, 3.2-1b, 3.2-1c, 3.2-1d, 3.2-2 (new), 3.2-2a, 3.2-2b, 3.2-2c, 3.2-2d

Technical Specification Tables 2.2-1, 3.2-1, 3.2-2, 4.3-1

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SAFETY FUNCTION OF THE SYSTEMS AFFECTED

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EFFECTS ON SAFETY

The reference cycle for the nuclear and thermal-hydraulic design of Cycle 6 is Cycle 5. The Cycle 6 physics parameters are based on a 400 effective full power day (EFPD) Cycle 5 length including APSR withdrawal and coastdown. There have been no anomalies during Cycle 5 which would adversely affect fuel performance during Cycle 6 as designed. Consistent with previous cycles, cross-core shuffling of fuel assemblies is minimized in this cycle. The Cycle 6 design is characterized by only 16 FAs being cross core shuffled so as to minimize any carryover effects from tilts encountered in previous cycles.

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The accidents analyzed in Chapter 15 of the Updated Safety Analysis Report have been reexamined, with respect to Cycle 6 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 6 is bounded for main steam line break or any other over-cooling transient. The radiological dose consequences of the SAR Chapter 15 accidents have been evaluated using conservative radionuclide source terms that bound the cycle specific source term for the longer Cycle 6 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 NUREG-0800.

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First, the shutdown borated water sources Technical Specification 3.1.2.8 was reviewed. The existing boron capability requirement to provide a 1% shutdown margin while cooling from 200°F appeared excessively high. Therefore, this requirement was reanalyzed for a cooldown from 200°F to 70°F with the results as presented in the proposed Technical Specification. In addition, prior to the reanalysis it became evident that the contained volume in the borated water storage tank (BWST) is not the same as the available volume in the tank because of the location of the suction piping at the tank. Although, previously, any unavailable water volume was calibrated out, the word "contained" is being changed to "available" with an explanation added in BASES 3/4.5.4 in order to avoid any future confusion. The remaining Technical Specifications 3.1.2.9, 3.5.4 and Surveillance Requirements associated with the BWST are also being changed to reflect the above.

The quadrant power tilt (QPT) limit Technical Specification 3.2.4 for Cycle 6 incorporates methods used to revise the power imbalance detector correlation (PIDC). (This revision to the PIDC method is addressed in a subsequent paragraph). The QPT alarm setpoint was widened by reducing the incore system error by utilizing the statistical combination of the incore system measurement and observability errors methodology. This methodology (See Letter: J. H. Taylor, B&W to H. R. Denton, NRR dated July 30, 1986) and the new PIDC, initiated thru the B&W Owners Group (BWOG), have been previously approved by the NRC and have already been implemented at other B&W facilities.

Another change to the Technical Specification is the introduction of the power dependent QPT limit. QPT has the effect of increasing the power in one quadrant of the core while decreasing it in another. The FAs with the increased power are the ones of interest. As the tilt, i.e., the power in the FA increases, the margin to DNB decreases. Clearly, the higher the initial power in the FA, the less tilt it can tolerate. Cycle 6, has been analyzed using a break point at 50%FP to take advantage of the lower power and to make low power operation easier. The wider QPT limit is valid for reactor power of 50%FP or less. The tighter QPT limit is valid at any licensed power. This change is considered by Toledo Edison to be safe and of benefit to operations by avoiding unnecessary entries into an action statement.

The remaining changes to the QPT and DNB Technical Specifications are administrative in nature. Currently, the two tables in the Technical Specifications are correctly referenced but are numerically out of sequence. To correct this, Table 3.2-2, Quadrant Power Tilt Limits is being re-numbered as Table 3.2-1. Similarly, Table 3.2-1, DNB Margin, is being re-numbered as Table 3.2-2 and the references changed accordingly.

The PIDC test calibrates the excore detectors to the incore detector measurements of core offset (core offset equals the ratio of the core power in the top of the core minus the core power in the bottom of the core to the actual core thermal power). Previously, the relationship between the excore and the incore offset was conservatively set with the excore detectors biased high. The new PIDC correlates the excore detectors more closely to the incore detectors by removing the unnecessary conservatism. This is acceptable based on the consequent tightening of the excore detector recalibration criteria as noted below. In order to preserve the required measurement system error allowance which results from reducing the conservatism, the criteria which required the excore detectors to be recalibrated has been tightened to require the excore detector offset to be within 2.5%FP of the incore detector offset (down from 3.5%FP). This is reflected in Technical Specification Table 4.3-1.

Table 4.3-1 is also being revised to increase the power level at which it is required to compare the excore detector offset to the incore detector offset. Using the same reasoning as for the QPT, imbalance has the effect of increasing the power in either the top or the bottom of the core. The lower the initial power the greater the imbalance that can be tolerated. Below 50%FP the maximum imbalance which can be induced produces peak powers within acceptable limits. Therefore, below 50%FP it is acceptable not to calibrate the excore detectors to trip the reactor on imbalance.

Technical Specification 3.4.1.1, pertains to reactor coolant circulation. The action statement, which restricts thermal power for three reactor coolant pumps (RCPs) to 79.7%, is being increased to 80.6%. This is acceptable as a result of the crossflow methodology. The wording changes cl are operating the corresponding flux/delta flux/flow limit curve is reduced, by the Reactor Protection System (RPS), by the ratio of the new three RCPs measured flow to the old four RCPs measured flow. This means the actual limit could be higher than shown in Technical Specification 2.2.1 if the actual flow is greater than the flow assumed in the example shown. This does not represent a change in the manner in which the RPS responds, it is only a clarification. Also, Surveillance Requirement 4.4.1.2 which refers back to the action statement is being revised. This is justified as follows. Following removal of one RCP from service the high flux limit is manually reduced and the RPS automatically reduces the flux/delta flux/flow curve to a new limit. This reduction in the two setpoints is to be verified within four hours to ensure the reactor will trip when required with three RCPs running. When the fourth RCP is put back into service the four hour verification is not required. If the high flux limit is not increased or if the RPS fails to readjust the limit curve upward, an unsafe condition would not exist as power is increased. This is because of the fact that the reactor would trip before full power could be reached. If the RPS responds as designed for the flux/delta flux/flow limit, there is no reason to assume the gain settings would change from their original four RCPs setting. Therefore, the limit curve would return to the original values. If for any reason the gain settings are adjusted while operating with three RCPs, procedures require the gain settings to be verified upon return to four RCPs. Consequently, the wording of the surveillance requirement is being changed to clarify that the four hour requirement applies only when the total number of running RCPs is reduced from four to three and is not required when the number of running RCPs is increased from three to four.

The planned startup physics test program has been reduced in scope from that performed for previous cycles. As a result of a Davis Besse undertaking with the BWOG, the zero power ejected rod worth and all-rods-in temperature coefficient measurements and one of the intermediate core power distribution test plateaus have been eliminated. The reduced scope physics testing as previously reviewed by the NRC has been utilized at other B&W facilities and has proven to be sufficient to demonstrate that the core will perform within the assumptions of the safety analysis.

SIGNIFICANT HAZARDS CONSIDERATION

The Commission has provided standards in 10CFR50.92(c) for determining whether a significant hazard 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: (1) not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) not create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) not involve a significant reduction in the margin of safety.

Toledo Edison has reviewed the proposed changes and determined that a significant hazards consideration does not exist because the operation of the Davis-Besse Nuclear Power Station, Unit No. 1 in accordance with these changes would:

Not involve a significant increase in the probability or consequences of an accident previously evaluated because the transient evaluation of Cycle 6 is bounded by the USAR accident analysis. The results of the USAR dose evaluations performed show that the offsite radiological doses for each accident in the USAR are below the acceptance criteria established in NUREG-0800, NRC Standard Review Plan. In addition, based on the proposed Technical Specification changes, the final acceptance criteria Emergency Core Cooling system limits will not be exceeded and the thermal design criteria will not be violated (10CFR50.92(c)(1)).

Not create the possibility of a new or different kind of accident from any accident previously evaluated because the transient evaluation of Cycle 6 is bounded by the USAR accident analysis. The results of the USAR dose evaluations performed show that the offsite radiological doses for each accident in the USAR are below the acceptance criteria established in NUREG-0800, NRC Standard Review Plan. In addition, based on the proposed Technical Specification changes, the final acceptance criteria Emergency Core Cooling system limits will not be exceeded and the thermal design criteria will not be violated (10CFR50.92(c)(2)).

Not involve a significant reduction in a margin of safety because the transient evaluation of Cycle 6 is bounded by the USAR accident analysis. The results of the USAR dose evaluations performed show that the offsite radiological doses for each accident in the USAR are below the acceptance criteria established in NUREG-0800, NRC Standard Review Plan. In addition, based on the proposed Technical Specification changes, the final acceptance criteria Emergency Core Cooling system limits will not be exceeded and the thermal design criteria will not be violated (10CFR50.92(c)(3)).

CONCLUSION

On the basis of the above, Toledo Edison has determined that the amendment request does not involve a significant hazards consideration.

REFERENCES

BAW-10117 P-A, Babcock & Wilcox Version of PDQ User's Manual, January 1977

BAW-10152A, NOODLE--A Multi-Dimensional Two Group Reactor Simulator, June 1985

BAW-10120, Comparison of Core Physics Calculations with Measurements, June 1978

BAW-10129-A, LYNX1 Reactor Fuel Assembly Thermal-Hydraulic Analysis Code, July 1985

BAW-10130-A, LYNX2 Subchannel Thermal Analysis Program, July 1985

BAW-10156-A, LYNXT Core Transient Thermal-Hydraulic Program, February 1986

Regulatory Guide 1.68, Revision 2, Initial Test Programs for Water-Cooled Nuclear Power Plants, August 1978