Due to recent anomalous mechanical behavior of the burnable poison rod assemblies (BPRAs) and orifice rod assemblies (ORAs), the licensee deems it prudent to remove all of the BPRAs and all but two of the ORAs from the Davis Besse Unit 1 (DB1) before the completion of the first cycle of operation. The removal of the BPRAs and ORAs will result in a change in various nuclear parameters as well as resulting in an increase in core bypass flow. Changes to various technical specifications are required as a result of the changes in nuclear parameters and core bypass flow. In referencing BAW-1489 the licensee describes the technical specification changes and provides analyses supporting the changes. Our review of the request and supporting documentation follows.

BPRAs are used in the first cycle of B&W reactors to control part of the initial excess reactivity and to flatten the radial power distribution. The reactivity controlled by burnable poison reduces the amount which must be controlled by soluble boron and prevents the occurrence of a positive moderator coefficient above 95 percent of full power. The Davis Besse Unit 1 reactor has achieved a first cycle burnup of 87 effective full power days (EFPDs) and some of the burnable poison has been burned out. However, sufficient burnable poison remains to require core changes in order to offset the effect of its removal. These core changes were:

1. Interchange of four intermediate (2.63 w/o) enrichment bundles near the center of the core with four low (1.98 w/o) enrichment bundles near the core periphery.

2. Rearrangment of the control rod groupings and decoupling of group seven from the withdrawal sequence. In the regrouping control rod group seven has been shifted toward the periphery and remains in the core until a burnup of 145 EFPDs has been reached. This arrangement serves to further flatten the radial power distribution and to replace some of the fixed poison in the core and thus prevent the moderator coefficient from becoming positive.

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B&W has performed an analysis of the modified core, assuming that the modification occurred at 80 EFPD and that the cycle length is increased from 433 to 485 EFPD. The analysis was performed using the same calculational methods and techniques that have been employed in the design of other B&W reactors--including Davis Besse Unit 1. The core physics parameters have been calculated for the modified cycle--80 to 145 EFPD with groups 5 and 6 essentially out of the core and group 7 completely inserted followed by 145 to 485 EFPD with groups 5 through 7 nearly out of the core. The recalculated parameters included shutdown margins, rod bank worths, ejected and dropped rod worths, stuck rod worth, Doppler coefficient, moderator coefficient, xenon worth, boron worth, and critical boron concentration.

During removal of the BPRAs it was discovered that sufficient wear was present on the holddown devices for the orifice rod assemblies (ORAs) to warrant their removal.

All of the ORAs will be removed with the exception of two modified orifice rod assemblies which are used with a primary neutron source. The removal of the ORAs increases the flow through the guide tubes but does not significantly alter the physics parameters. Thus, the analyses presented in BAW-1489 remain in effect.

We have reviewed the information presented in BAW-1489 for the values of the physics parameters and core flow and their effect on the safety analyses for Davis Besse Unit 1. For the rod withdrawal transients at full and zero powers, the control rod misoperation transient, the rod ejection accident, the moderator dilution transient, cold water accident, steam line failure accident, loss of coolant accident, and loss of normal feedwater transient, the significant parameters are shown to be bounded by those used in the Final Safety Analysis Report analysis. Thus, the consequences of these transients and accidents will not be greater than those described in the Final Safety Analysis Report.

The loss of electric power transient and the steam generator tube failure transient are independent of the significant parameter changes and the Final Safety Analysis Report analyses are, therefore, applicable for these transients.

By reference 1, revised analyses were submitted for the loss of flow transient and the feedwater system malfunction transient. The minimum DNBR transient is the one-pump loss of flow transient which results in a minimum DNBR of 1.45. It should be noted that the Davis Besse Unit 1 Final Safety Analysis Report and BAW-1489 indicated that the most limiting loss of flow transient was a four-pump loss of flow transient. The one-pump loss of flow transient became the most limiting transient when the power imbalance/flow reactor trip was adjusted to decrease inadvertent power imbalance/flow reactor trips. This trip adjustment was made prior to operation of Davis Besse Unit 1. It should also be pointed out that incorporating margin to compensate for fuel rod bow results in a minimum required DNBR of 1.445, thus the limiting loss of flow transient resulting in a minimum DNBR of 1.45 is acceptable.

After completion of the core modification, startup tests will be performed to assure that the various physics parameters are bounded by those in the Final Safety Analysis Report. Tests will be performed on rod drop times, critical boron concentration, temperature coefficients, control rod worths, power distributions, and power coefficients. Successful completion of tests at each power level will be required before proceeding to the next higher power level.

On the basis of the use of approved calculation methods, and of the proposed startup tests we find the analysis of the physics parameters of the core modification to be acceptable. We require, however, that the moderator temperature coefficient measurement be repeated at 145 EFPDs after removing group 7.

Because of the modification of core loading, some changes have been made in power distributions in the core. These changes necessitate changes in the technical specifications. Further changes are necessitated by the reprogramming of the rod groups.

The new technical specifications have been established using procedures which have been previously employed. New safety limits (Spec. 2.1.2) and Trip Setpoints (Fig. 2.2-1) and Allowable Values (Fig. 2.2-2) have been specified. New rod insertion limits (Spec. 3.1.2.6) have been specified along with new axial imbalance limits (Spec. 3.2.1) to ensure that peaking factor limits used as input to the LOCA-ECCS analysis are not exceeded. The rod program description has been changed (Spec. 3.1.7) to reflect the modification in group assignments. The maximum boration capability requirements (page B3/4 1-2) has been changed to reflect the reactivity changes resulting from the removal of the BPRAs and the relocation of the fuel assemblies.

The procedures used to establish the technical specifications on power distribution limits have been previously reviewed and approved. Based on this review and approval we find the technical specifications changes described above to be acceptable.

A further technical specification change, unrelated to the core modification, is requested. This request concerns the modification of alarm setpoints on quadrant tilt to accommodate a recently discovered increase in the measurement error associated with this quantity. The original uncertainty evaluation was performed in 1974 based on data obtained from prototype detectors. Observations of anomalies in operating reactors led to the reevaluation of this error. The licensee submitted (letter, Taylor to Reid, dated May 11, 1978) a document describing the methods used to perform the statistical analysis of the uncertainties and giving revised quadrant tilt alarm setpoints for Davis Besse Unit 1. We have reviewed the document and conclude that the analysis method is acceptable. We have not reviewed the data base used to obtain numerical results but we know of no data that would make the application of the method to Davis Besse Unit 1 nonconservative. We, therefore, find the revised alarm setpoints on quadrant tilt to be acceptable.

Removal of all the BPRAs and all but two of the ORAs from the core results in a calculated increase of 4.7% in the maximum core bypass flow (from 6.04% to 10.75%). A previous letter (Reference 2) requested that the minimum allowable reactor coolant flow be increased by 5% over the FSAR design flow to compensate for potential effects of fuel rod bowing. Therefore, modified operating conditions have been proposed to compensate for both the increased bypass flow and the potential effects of rod bow on the core thermal safety margin. An analysis has been performed based on a minimum allowable flow rate of 110% of design flow and a slightly adjusted trip limit curve (Technical Specification Figure 2.1-1) for reactor coolant core outlet pressure and outlet temperature. The analysis results indicate that operation at the proposed limits with BPRAs and ORAs removed would not result in violation of acceptable fuel design limits. Reactor coolant system flow measurements have indicated an actual system flow rate of at least 113% of the previous limit (measurement errors not included).

In a B&W designed nuclear power plant, Gentile flowmeters are used to measure loop 1 and loop 2 reactor coolant flow rates (B&W plants have two loops with two pumps each). These primary loop flowmeters are not calibrated prior to installation. Loop 1 and 2 feedwater flow rates are measured with calibrated flowmeters and a plant heat balance is used to calibrate the Gentile flowmeters.

The total reactor coolant flow rate for Davis-Besse, Unit 1, as determined from a plant heat balance, is 113.2% of the design flow rate. Based on the accuracies of primary and secondary side measurements reported in Table 1, the licensee calculated the reactor coolant flow rate accuracy to be ± 2.2%.

Technical Specification changes which are proposed to reflect the modified operating limits, including measurement uncertainties, are described in Section 7 of BAW-1489.

Staff calculations of bypass flow through the guide tubes with the BPRAs and ORAs removed give approximate agreement with the value reported by B&W. Therefore, an increase in the reactor vessel flow of about 5% is sufficient to compensate the increased bypass flow. Also, as reported in a separate evaluation (Reference 5) an additional 5% in design flow provides sufficient margin to compensate for the potential effects of fuel rod bow on DNBR. The above considerations tend to confirm the analysis results for the modified operating limits.

Measurement accuracies for primary and secondary side measurements used for calculation of reactor coolant flow rate are shown in Table 1. Except for the pressure uncertainty and flow ΔP uncertainty, these values are reasonable and consistent with industry practice. The most significant terms in calculating accurate values of reactor flow rate are reactor coolant temperatures and feedwater flowmeter differential pressures.

The measurement accuracy reported for reactor coolant pressure is $\pm$ 0.77%; staff experience indicates that $\pm$ 1% is more reasonable. The change to $\pm$ 1% pressure measurement accuracy does not affect the final reactor coolant flow accuracy as given to three significant digits.

The measurement accuracy reported for reactor coolant flow rate ΔP (± 1.046%) is for the ΔP transmitter only. It is the staff's opinion that a drift allowance for the flow element (Gentile tube) is also needed. Therefore the staff has re-evaluated the reactor coolant flow measurement accuracy using a value of ± 2% for the reactor coo ant flow rate ΔP measurement. The effect of this change is to increase the total flow rate measurement accuracy from ± 2.2% to ± 2.5%.

An important element in the error analysis is the assumed independence of the uncertainties in measurement of feedwater flow for the two loops. The major potential source of dependency for the feedwater flow measurement uncertainties is crud buildup in the flow elements. Although crud buildup has been observed in the feedwater venturi's for at least one reactor vendor, the once-through steam generator feedwater chemistry control minimizes the increase of contaminants into the system and the buildup of crud on the flow elements for Davis Besse. Therefore, it is reasonable to assume that the feedwater flow measurement accuracies are i dependent.

Flow requirements given in Table 3.2-1 of the proposed Technical Specification revision (attachment to letter No. 439, 5/26/78) include a measurement uncertainty of ± 2.2% factored into the 110% design flow required for potential rod bow effects and increased bypass flow. Because the staff has assessed the measurement accuracy at ± 2.5%, a revised Table 3.2-1 is included with this evaluation.

Based on the analyses presented in the report BAW-1489 and previously documented analyses, Davis Besse can be operated safely during Cycle 1 without BPRAs and ORAs at the rated core power level of 2772 MWt. Revised Technical Specification limits necessary for the safe operation at that power level are included in that report. The minimum flow required to assure a design flow of 110% original design flow, considering measurement uncertainties, is included in the revised Table 3.2-1 of this evaluation.

We have also reviewed the modified orifice rod assembly (MORA) for acceptability. A MORA is a standard ORA modified for use with a primary neutron source. During the initial core operation of Davis Besse Unit 1, two primary neutron sources are located in individual guide tubes of two fuel assemblies. Each source is held in a shroud tube which rests on the bottom of a guide tube. A solid stainless steel rod is placed on top of the source to hold it down against hydraulic lift. To provide further assurance that the source will not come out of the guide tube during postulated accidents, an ORA is latched to the top of the fuel assembly. The rods of the ORA plug the top of each guide tube including the guide tube containing the source.

To prevent the MORA from causing wear of the fuel assembly end fitting and coming loose, Toledo Edison and B&W propose to modify the primary source capturing arrangement. Firstly, twelve of the rods in each of the two ORAs remaining in the core are being removed, leaving only the rod above the source and the three symmetrically located rods. Secondly, a retainer is to be placed over the hub of the modified ORA and held down by the reactor internals.

The design and testing of this retainer device are described in reference 3. From a mechanical design standpoint, the basic concern is whether the retainer provides enough holddown force to preclude loosening of the MORAs. From analyses of the static and dynamic stresses on the retainer spring load arm and housing, results of prototype testing in a flow test facility, and in-air mechanical tests, criteria for use of the BPRA retainer device with modified ORAs have been established. The primary criterion is that the margin to component lift with the retainer, taking into account the hydraulic forces acting on the MORA, the MORA weight, and the retainer holddown force, should be greater than 30 pounds. This criterion is met with acceptable margin by the fact that when the retainer device is used with the modified ORA, the holddown force is greater than 35 pounds with all four reactor coolant pumps operating. A second criterion, which is related to fuel assembly growth, is based on a fuel assembly burnup design value that is used as a basis for the retainer design. Since the maximum burnup used in one cycle of operation will be less than the burnup used as a design basis, the fuel assemuly growth criterion is met (note that the retainer will be used for only one cycle of operation).

The potential consequences of a retainer failure have also been addressed (Ref. 4), although failure is considered unlikely. The neutronic and thermal-hydraulic consequences are considered insignificant. Interference with control rod motion, for example, would not, according to analyses of stuck-out control rod transients for B&W 177-FA plants, prevent safe shutdown of the plant.

The major concern associated with retainer failure is plant damage and potential outages for repair. This damage should be precluded by the Loose Parts Monitoring System (LPMS). The LPMS is designed to detect a failed retainer in either the reactor vessel or steam generator. Even though the BPRA retainer is designed for only one cycle of operation, The licensee (Ref. 2) will recommend that surveillance inspections be made following retainer use. This should provide additional confirmation of acceptable operation. The licensee stated that definite plans regarding surveillance will be provided to NRC as they are formulated.

In summation, we conclude that, based on (1) analyses and test results on the BPRA retainer device, (2) establishment and meeting of criteria for use of the device with ORAs modified for use with primary neutron sources in Davis Besse Unit 1, (3) analyses which indicate that failure of the retainers, however, unlikely, would not prevent plant safe shutdown and (4) failure detection capability of the Loose Parts Monitoring System, there is reasonable assurance that the proposed use of the BPRA retainer with two MORAs in Davis Besse Unit 1 will pose no significant safety concern.

References

1. Letter and Attachment, Serial No. 443, Lowell, E. Roe to John F. Stolz, June 8, 1978.

2. Letter and Attachment, Serial No. 426, Lowell E. Roe to Roger S. Boyd, April 10, 1978.

3. "BPRA Retainer Design Report," Babcock & Wilcox Report, BAW-1496, May 1968.

4. Telex communication, James H. Taylor (B&W) to Steven A. Varga (NRC), June 7, 1978.

5. "Safety Evaluation of Proposed Change to the 'DNBR Margin' Technical Specification for Davis-Besse, Unit 1," Memorandum from D. F. Ross to D. B. Vassallo.

TABLE 1

ACCURACY OF PRIMARY AND SECONDARY SIDE MEASUREMENTS USED FOR CALCULATION OF TOTAL RC FLOWRATE

PARAMETER	MEASUREMENT ACCURACY %	SPAN	ACCURACY UNITS
RC hot leg temp.	± 0.79	520 to 620F	±0.79 F
RC cold leg temp.	± 0.79	520 to 6[illegible] F	±0.79 F
Steam temp.	± 0.60	0 to 700F	±4.2 F
Feedwater temp.	± 1.13	0 to 600F	±6.8 F
Feedwater pressure	±1.0%	0 to 1500 psig	± 15 psi
Steam pressure	±1.89%	0 to 1200 psig	± 23 psi
RC pressure	± 0.77%	0 to 2500 psig	± 19 psi
Feedwater Flow	± 1.25%	0 to 960 inches (Std. H_2O)	± 12. inches
RC Flowrate	± 1.046	0 to 910 inches (Std. H_2O)	± 9.5 inches

TABLE 3.2-1

DNB MARGIN

	LIMITS		
Parameter	Four Reactor Coolant Pumps Operating	Three Reactor Coolant Pumps Operating	One Reactor Coolant Pump Operating in Each Loop
Reactor Coolant Hot Leg Temperature T_H °F	$\leq$ 611.1	$\leq$ 611.1[1]	$\leq$ 611.1
Reactor Coolant Pressure, psig.[2]	$\geq$ 2062.7	$\geq$ 2058.7[1]	$\geq$ 2091.4
Reactor Coolant Flow Rate, gpm (3)	$\geq$ 396,880	$\geq$ 297.340	$\geq$ 195.760

(1) Applicable to the loop with 2 Reactor Coolant Pumps Operating

(2) Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

(3) These flows include a flow rate uncertainty of 2.5%.