FRAMATOME ANP

BAW-10242(NP)
Revision 0

ZPPT Modifications for B&W Designed Reactors

August 2003

Framatome ANP, Inc.

Non-Proprietary

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
Rev. 0		Initial issue.

I. PURPOSE

The purpose of this report is twofold: (1) to describe the evaluations performed by Framatome ANP to justify a revised zero power physics testing program for Babcock & Wilcox (B&W) designed reactors, and (2) to gain NRC acceptance for the revised program.

Zero power physics testing (ZPPT) is required for PWRs following completion of a refueling outage. The required testing involves a number of tests performed at zero (very low) power prior to power escalation. Significantly reducing the time required to perform these tests will increase the efficiency of post-refueling activities, since ZPPT is performed on the critical path.

The ZPPT program and the reload physics startup program (including power escalation testing) are discussed herein. Framatome-ANP (FANP) specifically seeks approval for the change to the control rod worth testing (item (1) in Section II below). The remaining items are modifications to FANP's testing program which are already in practice at other operating U.S. PWRs and for which NRC approval is not required. These modifications are included herein for information only.

II. SUMMARY

Most of the modifications to the ZPPT program for B&W-designed reactors outlined herein are minor changes to the current scope of testing. These changes consist of modifications to test techniques and approaches that result in gathering the same data as the previous ZPPT program. NRC approval is sought for item (1) below. The remaining items are modifications to FANP's testing program which are included herein for information only. The changes are as follows:

1) The measurement of Control Rod Groups (CRGs) 5, 6, and 7 to determine worth has been changed to measure only CRGs 6 and 7. This change is discussed in Section III. A. NRC approval is sought for this modification.

2) The all rods out critical boron concentration (AROCBC) test has been changed from 100 percent withdrawal of CRG 7 to a minimum of 80 percent withdrawal. This change is discussed in Section III. B.

3) The test for determining the all rods out temperature coefficient (α_T) has been changed to perform two reactor coolant system temperature changes. This change is discussed in Section III. C.

4) The differential boron worth (DBW) test has been changed as follows:

   - Boron equilibrium (between the RCS and pressurizer and between the RCS and the makeup tank) is no longer required following the completion of rod worth measurements.

- A measured DBW will be obtained by taking the ratio of the reactivity rate of change (from the reactivity computer) to the boron rate of change from measured boron samples at specific time intervals.

- The measured DBW results will be considered information only.

These changes are discussed in Section III. D.

III. DETAILED TEST DESCRIPTIONS AND JUSTIFICATION FOR REVISED TESTING PROCEDURE

The purpose of this section is to describe the evaluations performed by FANP to justify a revised ZPPT Program for B&W-designed Reactors.

A. CRG Worths

FANP proposes that only CRGs 6 and 7 be measured for worth rather than CRGs 5, 6, and 7.

The primary reason for discontinuing the measurement of CRG 5 is that present-day physics codes for predicted CRG worth have demonstrated the ability to calculate individual CRG worth. Table 1 contains comparisons of measured CRG worths to predicted values for recent startups for B&W-designed reactors. The measured worths are determined using the boron swap (boron dilution) method. The acceptance criterion for the allowable % deviation ({Pred – Meas} / Pred * 100 %) for an individual CRG is ±15%. Table 1 demonstrates the accuracy of the CRG worth calculations.

Table 2 shows that "total" CRG worth % deviations (the differences compared to predicted for the sum of all measured CRGs) would be nearly identical if only CRGS 6 and 7 are measured versus the current practice of measuring CRGs 5, 6, and 7.

Below are additional justifications for this change:

1) The ANS 19.6.1 Standard distinguishes between "control rod groups" and "safety groups" based on normal practice. While CRG 5 is still considered a control rod group, CRG 5 is very rarely inserted during normal power operations, such that it is essentially a safety group.

2) Control rod worth testing is performed to assess whether or not the core is operating as designed – not to measure the worth of every control rod. The assessment that the core is operating as designed (and that shutdown margin-related acceptance criteria can be met) can be accomplished by measuring CRGs 6 and 7 as accurately as the determination can be made by measuring CRGs 5, 6, and 7.

3) The presence of the fixed incore detector system at the B&W-designed reactors and associated on-line computing software determines the presence of unlatched control rods (or other potential core anomalies) at low power levels. The measured power distribution is provided as low as 8 %FP at six-minute intervals at B&W-designed reactors. These power distribution measurements are continuously available at a low power level for the B&W-designed reactors.

4) The ANS 19.6.1 Standard states that the rods measured should be "radially representative" of the core. CRG 5 locations are in close enough proximity to the CRG 6 and 7 locations such that significant additional information relative to the zero-power power distribution is not acquired by measuring CRG 5 worth, as illustrated by Figures 1 and 2.

5) From a shutdown margin perspective, measuring just two groups by dilution is consistent with how the rods are inserted during normal plant operation Measuring control rod worth for each reload is to verify the uncertainty used in the cycle-specific shutdown margin analyses, it follows that a more direct verification of this uncertainty is obtained by measuring rod worth by dilution – even if only two CRGs are measured.

6) Also, from a shutdown margin and operations perspective, measuring CRG 5 worth by dilution usually means that Technical Specification MODE 2 Physics Test Exceptions are declared to allow for CRG 4 (safety group) insertion to account for possible over-dilution. Additionally, having the reactor critical with all the CRGs (CRGs 5-7) inserted, places the core in a configuration where the maximum ejected rod worth exists. Therefore, eliminating the CRG 5 worth measurement will result in the operational convenience of not declaring a MODE 2 Physics Test Exception and will result in less probability for a limiting reactivity insertion accident.

7) Since the test criteria are not altered to reflect the smaller sample size, measuring fewer control rods provides a more severe test of neutronics models underlying the predictive and engineering analysis of the core.

NOTE 1:

Reference 2 documents the NRC acceptance of as low as a 5% uncertainty when using the approved NEMO code (Reference 3) to calculate the total rod worth. Testing during the startup of each reload cycle confirms the validity of this uncertainty. Measuring one less CRG does not impact the conclusions reached in that document.

NOTE 2:

The rod worth evaluations herein were performed using predicted data from both the NEMO and SIMULATE-3 nodal codes. Both codes have been approved for reload licensing calculations. They utilize similar advanced nodal methods to determine the core reactivity and power distribution. Differences in cross section treatment have been verified to be accurate for each code system. Some examples of equivalent rod worth results are provided in Table 4. The results of this analysis are valid for predicted data calculated from either the NEMO or SIMULATE-3 nodal code, or any other code used in the future that has been adequately benchmarked (to the level demonstrated herein) and approved for use in the reload design process by the NRC.

Table 1
Individual Group Worth Comparisons

Plant	Cycle	Group 7 Worth (pcm)			Group 6 Worth (pcm)			Group 5 Worth (pcm)			Total		
		Predicted	Measured	% Dev	Predicted	Measured	% Dev	Predicted	Measured	% Dev	Predicted	Measured	% Dev
Crystal River 3	10	891	943.5	-5.9	864	832.7	3.6	1399	1522.4	-8.8	3154	3298.5	-4.6
	11	885	907.1	-2.5	845	867.3	-2.6	1309	1338.5	-2.3	3039	3112.9	-2.4
	12	918	958.6	-4.4	843	812.5	3.6	1475	1470.6	0.3	3236	3241.6	-0.2
	13	875	873.1	0.2	976	956.6	2.0	1248	1192.2	4.5	3099	3021.9	2.5
Davis Besse	10	1114	1101.2	1.1	717	719.0	-0.3	1554	1565.2	-0.7	3385	3385.4	0.0
	11	860	894.4	-4.0	768	740.5	3.6	1258	1356.3	-7.8	2886	2991.3	-3.6
	12	830	856.9	-3.2	807	796.8	1.3	1314	1378.8	-4.9	2951	3032.5	-2.8
	13	795	830.3	-4.4	913	921.6	-0.9	1173	1197.8	-2.1	2881	2949.7	-2.4
ANO-1	15	851	869.8	-2.2	894	890.3	0.4	1625	1543.2	5.0	3370	3303.3	2.0
	16	861	875.1	-1.6	881	868.1	1.5	1364	1293.1	5.2	3106	3036.2	2.2
	17	913	935.5	-2.5	825	841.7	-2.0	1409	1351.2	4.1	3147	3128.4	0.6
TMI-1	10	952	956.0	-0.4	735	713.0	3.0	1377	1400.0	-1.7	3064	3069.0	-0.2
	11	1023	1078.5	-5.4	812	807.5	0.6	1188	1219.5	-2.7	3023	3105.5	-2.7
	12	881	909.8	-3.3	754	741.9	1.6	1227	1263.0	-2.9	2862	2914.6	-1.8
	13	901	934.1	-3.7	853	868.2	-1.8	1486	1499.6	-0.9	3240	3301.9	-1.9
	14	951	1000.8	-5.2	870	825.3	5.1	1166	1160.3	0.5	2987	2986.4	0.0
Oconee 1	17	841	868.7	-3.3	932	941.2	-1.0	1139	1192.3	-4.7	2912	3002.2	-3.1
	18	902	933.9	-3.5	950	950.7	-0.1	1093	1173.1	-7.3	2945	3057.7	-3.8
	19	763	819.7	-7.4	905	897.3	0.9	1178	1242.2	-5.4	2846	2959.1	-4.0
	20	808	843.7	-4.4	846	860.7	-1.7	1094	1206.6	-10.3	2748	2910.9	-5.9
Oconee 2	16	871	925.6	-6.3	790	818.0	-3.5	1350	1465.3	-8.5	3011	3208.9	-6.6
	17	797	840.5	-5.5	745	800.2	-7.4	1248	1289.8	-3.3	2790	2930.5	-5.0
	18	776	830.0	-7.0	794	805.9	-1.5	1222	1306.6	-6.9	2792	2942.5	-5.4
	19	859	878.4	-2.3	779	843.1	-8.2	1267	1348.4	-6.4	2905	3069.9	-5.7
Oconee 3	16	800	806.9	-0.9	873	883.3	-1.2	1255	1302.6	-3.8	2928	2992.8	-2.2
	17	881	898.0	-1.9	877	898.0	-2.4	1428	1445.0	-1.2	3186	3241.0	-1.7
	18	896	907.2	-1.2	798	785.9	1.5	1150	1178.3	-2.5	2844	2871.4	-1.0
	19	927	973.4	-5.0	912	936.8	-2.7	1218	1360.6	-11.7	3057	3270.8	-7.0

%Dev = (Pred – Meas) / Pred * 100

Table 2
Groups 6 and 7 Combined Comparisons

Plant	Cycle	Total %Dev	Grp 7 + Grp 6 %Dev
Crystal River 3	10	-4.6	-1.2
	11	-2.4	-2.6
	12	-0.2	-0.6
	13	2.5	1.1
Davis Besse	10	0.0	0.6
	11	-3.6	-0.4
	12	-2.8	-1.0
	13	-2.4	-2.6
ANO-1	15	2.0	-0.9
	16	2.2	-0.1
	17	0.6	-2.3
TMI-1	10	-0.2	1.1
	11	-2.7	-2.8
	12	-1.8	-1.0
	13	-1.9	-2.8
	14	0.0	-0.3
Oconee 1	17	-3.1	-2.1
	18	-3.8	-1.8
	19	-4.0	-2.9
	20	-5.9	-3.0
Oconee 2	16	-6.6	-5.0
	17	-5.0	-6.4
	18	-5.4	-4.2
	19	-5.7	-5.1
Oconee 3	16	-2.2	-1.0
	17	-1.7	-2.2
	18	-1.0	0.1
	19	-7.0	-3.9

Table 3
CRG 7 Worth from 60 to 100 %WD

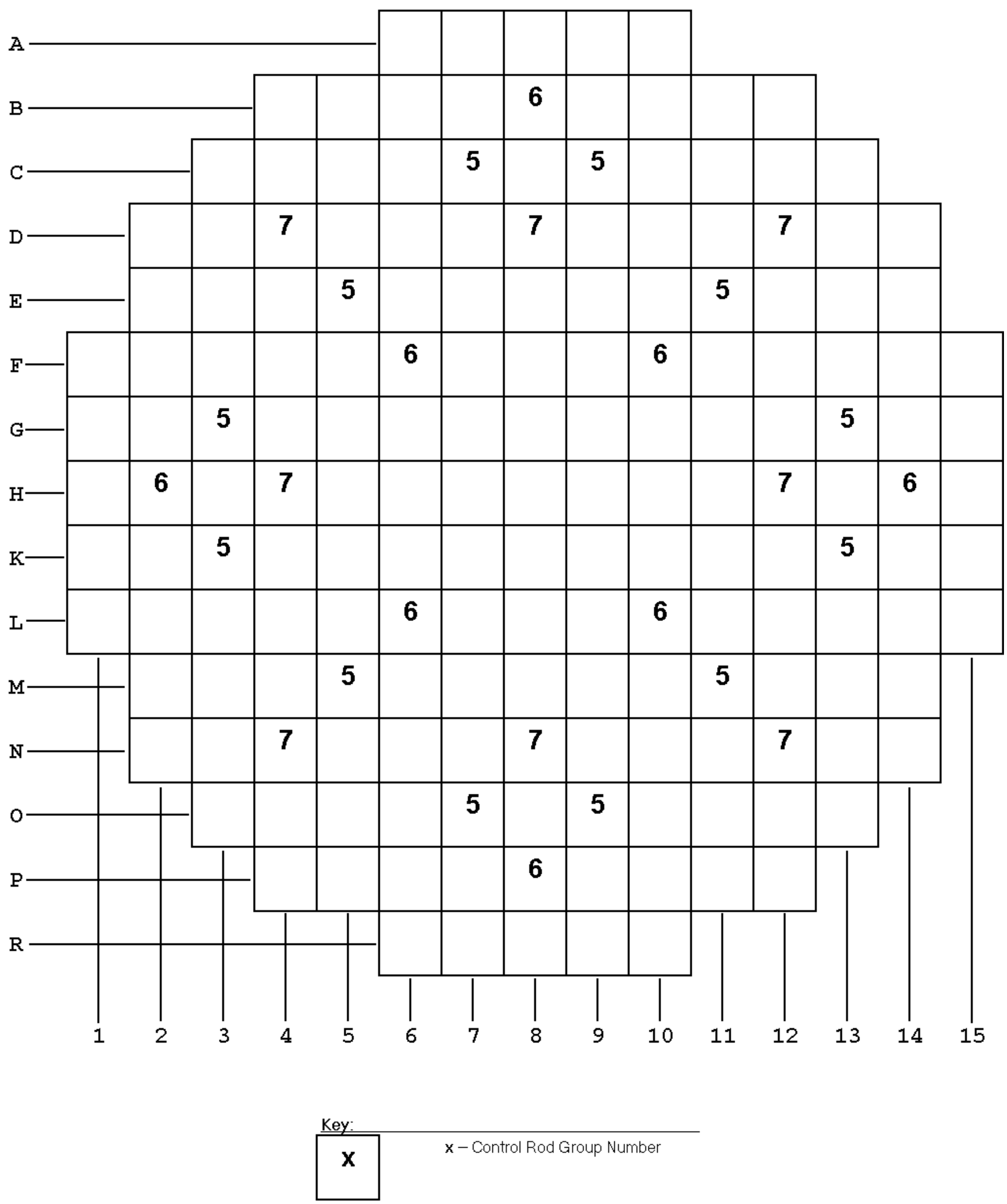
Plant	Cycle	Group 7 (60 to 100)		
		Predicted	Measured	% Dev
Crystal River 3	10	244	355.6	-45.7
	11	261	362.0	-38.7
	12	274	382.3	-39.5
	13	295	291.0	1.3
Davis Besse	10	290	319.9	-10.3
	11	277	283.7	-2.4
	12	288	323.0	-12.2
	13	285	303.8	-6.6
ANO-1	15	316	339.0	-7.3
	16	326	341.3	-4.7
	17	329	366.7	-11.5
TMI-1	10	265	302.4	-14.1
	11	314	382.1	-21.7
	12	310	311.4	-0.4
	13	359	387.0	-7.8
	14	345	368.7	-6.9
Oconee 1	17	290	258.8	10.7
	18	277	288.0	-4.0
	19	265	287.6	-8.5
	20	279	271.7	2.6
Oconee 2	16	299	290.1	3.0
	17	275	247.7	9.9
	18	277	282.8	-2.1
	19	269	258.4	3.9
Oconee 3	16	257	225.2	12.4
	17	281	256.6	8.7
	18	253	244.2	3.5
	19	307	273.1	11.0

Avg	-6.2
Std Dev	14.3

Figure 1

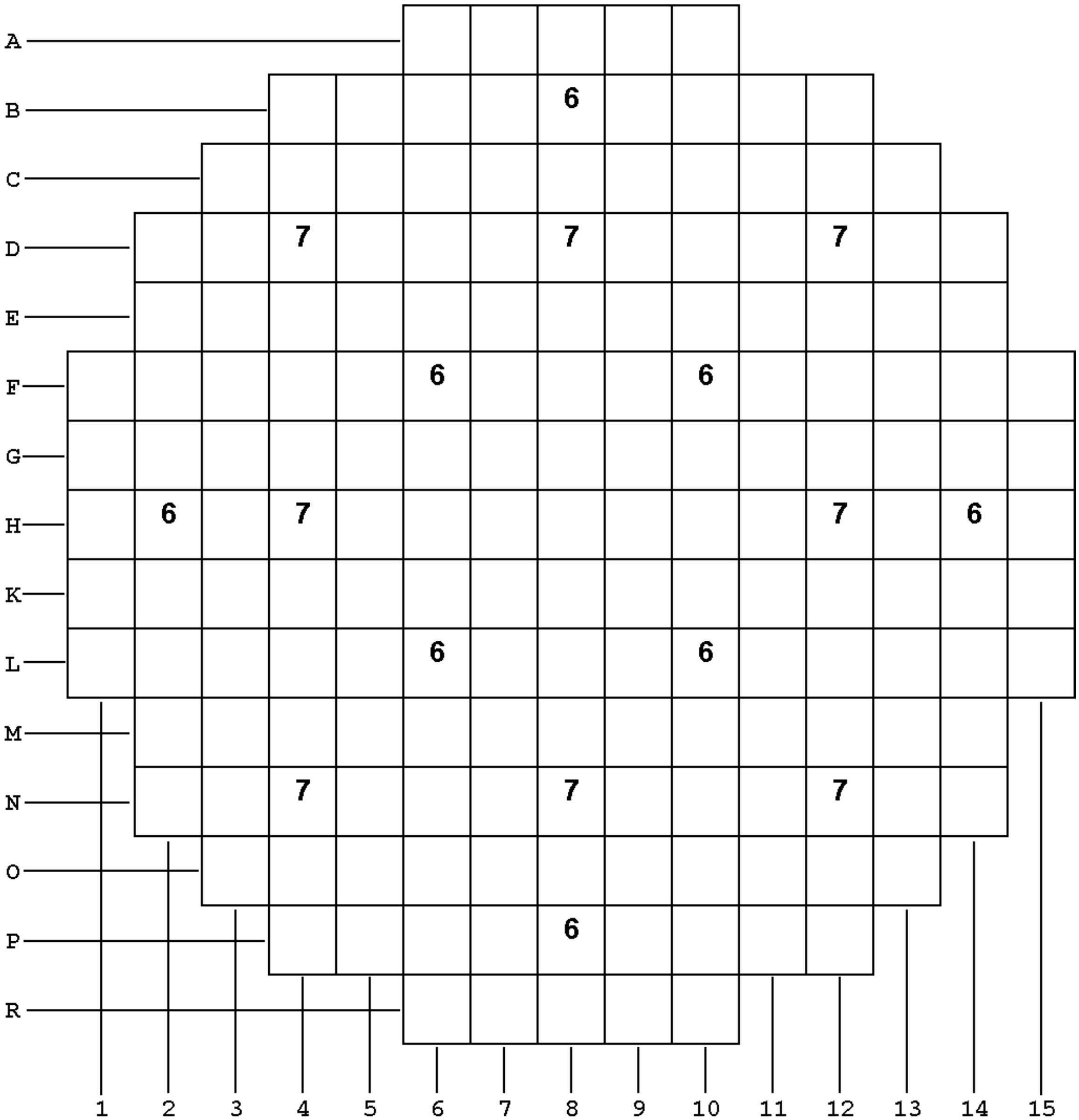
Typical Control Rod Group Location in 177-FA Core

Full Core Layout -- CRGs 5 & 6 & 7



Typical Control Rod Group Location in 177-FA Core

Full Core Layout -- CRGs 6 & 7



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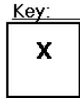

x – Control Rod Group Number

Figure 2

Table 4
NEMO vs SIMULATE Predicted Rod Worths

Plant	Cycle	Group 7 Worth (pcm)			Group 6 Worth (pcm)			Group 5 Worth (pcm)			Total		
		SIMULATE	NEMO	% Dev	SIMULATE	NEMO	% Dev	SIMULATE	NEMO	% Dev	SIMULATE	NEMO	% Dev
ANO-1	15	828	851	-2.8	860	894	-4.0	1584	1625	-2.6	3272	3370	-3.0
	16	835	861	-3.1	850	881	-3.6	1327	1364	-2.8	3012	3106	-3.1
	17	886	913	-3.0	795	825	-3.8	1358	1409	-3.8	3039	3147	-3.6
TMI-1	9	833	856	-2.8	889	822	7.5	1051	1117	-6.3	2773	2795	-0.8
	10	913	952	-4.3	708	735	-3.8	1318	1377	-4.5	2939	3064	-4.3
	11	996	1023	-2.7	791	812	-2.7	1155	1188	-2.9	2942	3023	-2.8
D-B	11	881	860	2.4	754	768	-1.9	1315	1258	4.3	2950	2886	2.2
	12	838	830	1.0	806	807	-0.1	1344	1314	2.2	2988	2951	1.2
	13	817	795	2.7	930	913	1.8	1194	1173	1.8	2941	2881	2.0
Oconee-1	20	808	823	-1.9	846	873	-3.2	1094	1083	1.0	2748	2779	-1.1
Oconee- 3	16	800	837	-4.6	873	905	-3.7	1255	1268	-1.0	2928	3010	-2.8

B. Critical Boron Concentration

The all rods out critical boron concentration (AROCBC) measures overall core reactivity. At a given rod configuration (near the all-regulating rods out condition), the boron concentration is measured and small corrections are made to correct to the ARO condition. The resulting measured ARO critical boron concentration is compared to a predicted value.

The primary correction made to correct to the ARO condition is the inserted rod worth of the lead control rod group (CRG). B&W plant owners have historically measured the inserted worth of CRG 7 to 100 %WD. Since the measurement is limited by the startup rate allowed and due to the uncertainty of obtaining critical conditions for a new core, there have been many instances where a boron adjustment was required to position CRG 7 closer to the ARO condition. The additional accuracy achieved by obtaining the critical configuration that would allow pulling CRG 7 to ARO is small and an inefficient use of time. To document this assertion, measured AROCBC values were determined for startups that had critical, equilibrium conditions present with a deeper CRG 7 position than desired.

The alternative (new method) AROCBC values were calculated using the following equation:

$$AROCBC\ (ppmB) = Boron_{RCS}\ (ppmB) + \left\{ \frac{Group\ 7\ Worth\ (pcm)}{Predicted\ DBW\ (pcm/ppmB)} \right\}$$

where

$Boron_{RCS}$	=	*RCS boron concentration measured by chemistry samples (equilibrium conditions)*
Group 7 Worth	=	*CRG 7 inserted bank worth (predicted value) based on CRG 7 position (%WD) at boron equilibrium*
Predicted DBW	=	*predicted value of DBW*

Alternate critical condition data for determining AROCBC values using the new method were obtained for 13 startups at B&W-designed reactors. The results are tabulated in Table 5 which demonstrates the adequacy of the new method. The difference between new method AROCBC values determined using predicted CRG 7 worth to correct to the ARO condition and the original AROCBC are negligible.

Table 5
Calculated AROCBC Data Using New Method

Plant	Cy	RCS Boron (ppmB)	Group 7 Position (%WD)	Group 7 Worth (pcm)	DBW (pcm /ppmB)	New Method AROCBC (ppmB)	Original Measured AROCBC (ppmB)	Delta (New Method -Measured) (ppmB)	Predicted AROCBC (ppmB)
TMI	6	1397	45	467.5	9.31	1447	1449	-2	1394
	7	1614	35	605.5	8.74	1683	1691*	-8	1636
	8	1806	56	351.2	7.93	1850	1846	4	1829
	10	2398	81	91	6.62	2412	2406	6	2449
	11	2249	61	304.6	6.54	2296	2295	1	2295
	12	2147	76	163	6.51	2172	2167	5	2195
	13	2134	72	230.6	6.41	2170	2176	-6	2164
CR-3	12	2269	70	179.5	6.43	2297	2299	-2	2297
Oco-1	20	1746	73	151	8.02	1765	1760	5	1760
Oco-2	15	1935	82	114	7.61	1950	1954	-4	1942
Oco-2	16	1984	62	276.4	7.42	2021	2015	6	2003
Oco-3	19	2026	65.6	243.7	7.08	2060	2064	-4	2108
ANO-1	17	2078	75	177.5	7.09	2103	2101	2	2129
						Average		0.58	
						Standard Deviation		4.89	

* A strong case can be made that measured AROCBC was actually 1687 ppmB. The "official" value is used here, but using 1687 would lower the delta to –4 ppmB.

Below are additional justifications for this change:

1) Industry experts (the current membership of ANS 19.6.1) have already endorsed this method and have incorporated this approach at several U.S. Utilities.

2) The approach adopted by B&W plants will typically involve using less than 100 pcm predicted worth, depending on where exact critical conditions are obtained. Table 5 supports corrections using predicted rod worth of greater than 200 pcm in several cases. Hence, the amount of rod worth correction using predicted data will be less for smaller values of CRG worth.

3) The primary contribution of measurement uncertainty for this parameter is the measurement uncertainty of the boron concentration. For most AROCBC measurements performed when the CRG 7 endpoint was measured, more than 99% of the measured AROCBC is determined from the chemistry sample. Similarly, for the revised technique, the percentage of the measured AROCBC that is still chemistry sample is 98 percent.

4) Predicted versus measured comparisons of the upper part of CRG 7 worth are depicted in Table 3. The average percent deviation for this dataset is –6.2 percent

(with a standard deviation of 14.3 percent). The data for CR-3 Cycles 10-12 and for TMI-1 Cycle 11 merit additional discussion. These measurements were taken for reload cycles that observed significant measured versus predicted imbalance differences in the previous fuel cycle. Computer simulations have shown that differences between measured and predicted offset at EOC explain the observed deviation.

To address any potential error with using predicted CRG 7 worth for the AROCBC determination, FANP will recommend that the original method for establishing critical, equilibrium conditions at a CRG 7 position such that the CRG 7 endpoint can be measured by pulling to 100 %WD (usually < 100 pcm) **if** the ± 50 ppm acceptance criterion for the test is being approached. This recommendation will take the following form:

IF predicted rod worth data is used to determine the measured AROCBC, **and** the difference between measured and predicted AROCBC is greater than ± 45 ppm, **then** a boron addition is initiated (if required) such that the endpoint correction consists entirely of measured CRG 7 data.

5) The endpoint correction for the AROCBC has always involved the use of predicted data. The use of the predicted DBW has always been the standard practice for this correction since the measured DBW was not available at the time of the AROCBC test. This approach is already being employed by several U.S. utilities.

C. Temperature Coefficient (α_T)

The test for determining the all rods out temperature coefficient (α_T) has been revised to perform two RCS temperature changes (decrease followed by increase, or increase followed by decrease) of 3-5 °F rather than the original +5/-10/+5 °F approach.

Section III. B. results in the possibility of performing the all rods out α_T test at a deeper CRG 7 position than before. The effects of performing this test with possibly deeper insertion of CRG 7 have been evaluated and determined to be negligible for CRG 7 positions greater than 55 percent withdrawn.

Below are additional justifications for this change:

1) ANS-19.6.1 for ZPPT (Reference 1) endorses the 3-5 °F decrease/increase method. Hence this method is widely employed in the industry by many utilities.

2) 3-5 °F decrease/increase method will provide two α_T values, which are averaged. No additional criterion is applied to the two measured values. For the previous approach measured α_T values were compared to the measured α_T value, and occasionally, the check criterion would not be satisfied. A re-test or lengthy evaluation process would

be required. Neither of these methods would significantly change the measured versus predicted result.

3) Heating up the RCS by 5 °F is inefficient.

D. Differential Boron Worth

The measured differential boron worth (DBW) value for B&W-designed reactors has been obtained during rod worth testing by dividing the measured rod worth (by dilution) by the difference between equilibrium boron samples before and after the rod worth measurements. The DBW test has been modified in the following manner:

Boron equilibrium (between the RCS and pressurizer and between the RCS and the makeup tank) is no longer required following the completion of rod worth measurements.

A measured DBW will be obtained by taking the ratio of the reactivity rate of change (from the reactivity computer) to the boron rate of change from measured boron samples at specific time intervals.

The measured DBW results will be considered information only. This change is consistent with previously approved exceptions to Reference 1 at other U.S. PWRs (Reference 4).

To justify this new approach, the database of DBW measurements at B&W-designed reactors was examined. The results in Table 6 based on using the revised method are comparable to the results from the original method and are more consistent with predicted values than the original method.

Table 6
Comparison of Measured DBW Values for Various Reactor Cycles

Plant	Cycle	Predicted DBW (pcm/ppmB)	Original Method		Revised Method	
			Meas. DBW (pcm/ppmB)	% Deviation	Meas. DBW (pcm/ppmB)	% Deviation
Davis-Besse	11	6.472	6.928	-7.05	6.590	-1.83
	12	6.572	7.015	-6.74	6.399	2.64
	13	6.373	6.683	-4.86	6.283	1.42
Arkansas Nuclear One – Unit 1	16	7.206	7.508	-4.19	7.425	-3.04
	17	7.088	7.461	-5.26	7.082	0.08
Crystal River Unit 3	10	6.774	7.441	-9.85	6.811	-0.54
	11	6.439	6.950	-7.94	6.403	0.55
	12	6.433	7.054	-9.65	6.506	-1.13
	13	6.497	6.923	-6.56	6.157	5.23
Three Mile Island Unit 1	11	6.543	6.927	-5.87	6.212	5.06
	12	6.510	6.754	-3.75	6.105	6.22
	13	6.414	6.449	-0.55	6.088	5.08
	14	6.342	6.431	-1.40	5.631	11.20
			Average	-5.67	Average	2.38
			Std. Dev.	2.79	Std. Dev.	4.01

IV. ADDITIONAL CONSIDERATIONS

Cases Where CRG 5 Worth Will Be Measured

FANP recommends that B&W-designed plants measure CRG 5 worth during reload physics testing for the first fuel cycle following the introduction of new control rod assemblies for CRG 5.

FANP recommends that B&W-designed plants measure CRG 5 worth during reload startup physics testing if any of the rod worth acceptance criteria are failed.

Startup Testing

The entire FANP recommended reload startup physics testing program is presented in this section to demonstrate the continued commitment that licensees of B&W-designed reactors have in verifying that their reload cores are operating as designed.

The purpose of the design analyses of the reload cycle is to ensure that the reference safety analyses remain applicable. The nuclear design analyses are based on modeling the core characteristics using the approved methods, procedures, and computer calculations described in Reference 5. The results of the design analyses show that bounding peaking distributions and bounding nuclear parameters are within the criteria required by the safety analyses. However, there remains an uncertainty related to the accuracy of the design calculations and modeling of the reload cycle characteristics relative to actual measurements. Reload startup physics testing is performed following refueling outages to verify that the core is operating as designed.

The previous cycle design predictions are benchmarked to startup test measurements, and core-follow calculations of the power distributions are also benchmarked to measured data. The previous cycle is the reference cycle for the reload core design. If there are no design changes or changes to the manufacturing specifications, then the conclusion could be reached that the design calculations are completely satisfactory to ensure that the safety parameters have been accurately analyzed. This conclusion is further supported by the topical reports on the computer codes, methods and procedures, and uncertainties, which have shown that the design analyses are sufficiently accurate.

However, prudence suggests that some amount of startup physics testing is important to ensure that the safety evaluations are valid. A small probability exists that the calculations will have larger-than-expected deviations simply because the calculational accuracy was established statistically. Also, a small probability exists that loading or manufacturing deviations may occur. Thus, a startup testing program is part of the reload evaluation process for the nuclear analysis.

Acceptance Criteria

The previous subsections in this nuclear design section have discussed the methodology for performing design analyses to ensure that the characteristics of a reload cycle are bounded by the reference safety analyses. The methodology referenced the calculational codes, models, and procedures that are used to determine the nuclear parameters. The same calculational codes, models, and procedures must be revalidated during the startup of each reload cycle by performing a minimum amount of startup physics tests which compare the resulting measured values to calculational predictions. Design calculations, using the calculational codes, models, and procedures that were used to verify that the nuclear parameters are bounded by the reference safety analyses, shall model startup conditions to produce predictions that can be compared to measurements.

Startup testing requirements should meet the requirements of ANS 19.6.1 (Reference 1). The standard startup physics testing scope for B&W-designed plants complies with ANS 19.6.1 (Reference 1) with the following exceptions:

1) Reference 1 specifies that if the boron dilution method for determining HZP measured rod worth is employed, then measurement of all control rod groups, or at least 3000 pcm is required. Reference 1 also specifies measurement of the entire CRG worth (over the entire range of travel). FANP has justified a ZPPT program that includes measurement of only CRG 7 (partial – at least 80% of the worth of CRG 7 is measured) and CRG 6. This is typically at least 1500 pcm.

2) Reference 1 suggests (the appropriate specification is contained in the Appendix, which is technically not part of the Standard) that the endpoint worth for CRG 7 is measured for the boron equivalent correction to the measured all rods out critical boron concentration (AROCBC). FANP has justified that up to 200 pcm predicted worth can be used for this correction.

3) Reference 1 requires a measured differential boron worth and application of a test criterion to a comparison of measured to predicted values. FANP has developed a modified differential boron worth measurement technique not included in the Appendix of Reference 1. Rather than eliminate the measurement of differential boron worth entirely, this new technique is employed with the results as information only (no test criterion is applied).

The current minimum scope of reload startup physics testing for B&W-designed plants is contained in Table 7.

Table 7
Reload Startup Physics Testing for B&W-Designed Plants

Test	Test Criterion	Notes
All Rods Out Critical Boron Concentration	$\pm$ 50 ppm -- Acceptance (Predicted – Measured) $\pm$ 45 ppm – Review*	Up to 200 pcm predicted worth of CRG 7 allowed for endpoint correction. * - Only applied if predicted worth is used.
Isothermal Temperature Coefficient	$\pm$ 2 pcm/^{o}F (Predicted – Measured)	
Moderator Temperature Coefficient	< Tech Spec Limit	Measured MTC inferred from measured ITC by application of predicted Doppler coefficient.
Individual CRG Worths	$\pm$ 15 % % dev = {(P–M) / P}x 100%	At least 80 % of CRG 7 and all of CRG 6.
Total CRG Worth	$\pm$ X % % dev = {(P–M) / P}x 100%	X = Shutdown margin related uncertainty on rod worth – always between 5-10%, depending on fuel cycle.
Differential Boron Worth	No criterion applied	Ratio of measured rod worth to measured boron differences during CRG worth measurements.
Flux Symmetry Test	| Tilt | < full power limit. Symmetric incore detector readings within $\pm$ 10 %	Both of these criteria are considered "review criteria". Evaluation should be accomplished before physics testing is performed at a higher power level.
Intermediate Power Level Core Power Distribution	Several specific acceptance criteria apply, including the criteria in Reference 1	Between 40-80 %FP
HFP AROCBC	$\pm$ 50 ppm	Difference between the HZP AROCBC P – M delta □and the HFP AROCBC P – M delta.
HFP Core Power Distribution	Several specific acceptance criteria apply, including the criteria in Ref. 1	Between 90-100 %FP

V. CONCLUSIONS

This report documents the technical evaluations performed to justify a revised ZPPT program for B&W-designed reactors. The resulting revised ZPPT program will significantly reduce ZPPT time for future reload fuel cycles at B&W-designed reactors while obtaining equivalent information as the previous ZPPT program.

VI. REFERENCES

1. "Reload Startup Physics Tests for Pressurized Water Reactors", ANS-19.6.1-1997, American Nuclear Society.

2. NRC Letter Robert Jones to J.H. Taylor, "Acceptance of Revised Measurement Uncertainty for Control Rod Worth Calculations", January 26, 1996.

3. BAW-10180-A Rev. 1, "NEMO Nodal Expansion Method Optimized", B&W Fuel Company, Lynchburg, Virginia, March, 1993.

4. USNRC Letter L Raghavan to Harold B. Ray, "San Onofre….Low Power Physics Testing Methodology", October 10, 2000.

5. BAW-10179P Rev. 4, "Safety Criteria And Methodology For Acceptable Cycle Reload Analyses, Framatome-ANP, Lynchburg, Virginia, August, 2001.