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COMPARISON OF PREDICTED AND MEASURED
NUCLEAR DESIGN PARAMETERS AT
BEGINNING OF LIFE OF CYCLE 1
IN THE SAN ONOFRE REACTOR

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ABSTRACT

A program of reactor physics tests has been carried out at beginning-of-life of the first fuel cycle in the San Onofre Nuclear Generating Station in support of the nuclear design effort which preceded construction and operation of the station. Final design methods and analytically derived predictions are reviewed and the results of the experimental program, spanning reactor operations from first criticality to initial operation at 395 MWe, are presented. Finally, comparison is made between predicted and measured values and conclusions concerning the validity of the analytical procedures are drawn.

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ERRATA

The following pages in WCAP-3269-66 should be changed as indicated below:

1. Page 33 Opposite Power Level Correction:

$$+ \left\{ -0.7 \times \frac{Q(\%)}{100} \right\} \left(\frac{\text{pcm}}{^{\circ}\text{F}} \right)$$

Opposite Depletion Correction:

Refer to Figure 3.10a attached

2. Page 106 Opposite Power Level Correction:

$$+ \left\{ -0.7 \times \frac{Q(\%)}{100} \right\} \left(\frac{\text{pcm}}{^{\circ}\text{F}} \right)$$

Opposite Depletion Correction:

Refer to Figure 3.10a attached

3. Page 132 Table 6.5

Power Level Correction $(-0.7 \times 60/100)$	-0.4 pcm/ $^{\circ}\text{F}$
Burnup Correction	-0.1 pcm/ $^{\circ}\text{F}$
Predicted Value for Conditions of Measurement	+2.5 pcm/ $^{\circ}\text{F}$

Isothermal Temperature Coefficient Correction Term, $\Delta \frac{\partial \rho}{\partial T} |_{Bu}$ (pcm/°F)

Figure 3.10a

Effect of Burnup on Isothermal Temperature Coefficient

I. INTRODUCTION

Motivation for preparing and executing a program of reactor physics measurements during the initial startup of the San Onofre Unit 1 reactor is to be found in several areas:

- 1) It is essential that the validity of the values of nuclear design parameters assumed in conducting the plant Safeguards Analysis be demonstrated.
- 2) It is necessary to show that the actual values of specified parameters fall within limits established in the San Onofre Unit 1 License and Technical Specifications.
- 3) It is highly desirable that values of operationally significant parameters such as control rod worth, for example, be developed under actual operating conditions for subsequent use by the plant operating staff.
- 4) It is correspondingly desirable that the validity of nuclear design procedures be verified, especially with respect to those parameters for which innovations in design methods have been effected. Introduction of the rod cluster control concept as an extension of the conventional cruciform control rod approach to reactor control rod shutdown is an instance where verification of design methods is of particular significance.

Surveys were made of the particular requirements arising in each of these areas and comprehensive programs of reactor physics testing under zero power and power escalation conditions, respectively, were devised, reviewed and adopted. The usual difficulties encountered in operating a power reactor within the close boundaries of an experimental program forced minor modifications and compromises but no significant deletions or changes.

In succeeding pages, consideration is given to the basic methods of both analytical prediction and experimental measurement employed to obtain the specific predicted and measured results. Comparison is made between measured values and analytical predictions, and compilations of significant operational data are presented.

For purposes of consistency and convenience in intercomparison of data all reactivity values are quoted in units of thousandths of one percent (pcm or 10^{-5}) in reactivity and all boron concentrations are quoted in terms of parts per million (ppm) of elemental boron.

II. REVIEW OF NUCLEAR DESIGN METHODS

A. Methods of Analytical Calculation

In the following sections are reviewed, in general terms, the analytical methods and models used to predict the nuclear characteristics of the San Onofre reactor. Successive sections summarize the means by which power distributions are determined, integral and differential control rod worths and differential boron worths are calculated, and kinetics coefficients (of moderator temperature and power level) are predicted.

It should be emphasized that the methods considered here are those in current use when the San Onofre reactor was designed and are relevant up to the beginning of actual plant startup. More recent evolution and refinement of methods, prompted in part by material presented in this report, has led to significant improvement in the ability to anticipate core behavior. Remarks relevant to these improvements are presented in Section VII.

1. Radial Power Distributions

All two-dimensional power distributions were calculated with the PDQ-03¹ code. The two-group diffusion constants used as input for the PDQ's were obtained from LEOPARD². The effective neutron temperature as well as the other necessary temperatures required for LEOPARD (pellet average, clad, and moderator temperature) were obtained from a method suggested by Sha.³

The PDQ x-y quarter-core mesh description for these calculations is illustrated in Figure 2.1. Because of the 7500 mesh-point limitation in PDQ-03, the San Onofre core was, at best, rather coarsely described. Each water hole in the Rod Cluster Control (RCC) assembly was given one mesh point at each corner of the water hole cell. Elsewhere, four mesh points described four fuel cells. Even so, because of the size of the San Onofre core, the outer two assemblies on each quarter-core axis were described with only forty-four internal mesh points. This "discrete" model was used for both rodged and unrodged

San Onofre Nuclear Generating Station
Unit 1, Cycle 1 Startup

Figure 2.1 Quarter Core Mesh Pattern for PDQ x-y
Power Distribution Studies

power distributions. The model was also used for checking moderator temperature coefficients and for finding critical boron concentrations for different core parametrical configurations.

The calculations did not incorporate any effects due to xenon and assumed that moderator and fuel (Doppler) temperatures are constant throughout the core.

2. Axial Power Distributions

Axial power distributions were calculated using slab geometry in the FAB code (a modification of the FOG⁴ code). The two group diffusion constants input to the FAB calculations were obtained from Sha's correlation⁵ and LEOPARD. The effect of control rods partially inserted into the core was synthesized by introducing a smeared uniform poison in the corresponding axial region of the core description input to the code. Axial power distribution calculations were carried out for both the hot zero power and full power conditions. Pointwise equilibrium xenon absorption effects were included in the calculations for each rod configuration in the full power condition.

In addition to the conventional axial hot channel factor data, this same series of calculations also yielded a family of "fraction of fission power generated below the rod bank" curves. These are plots of the fraction of power below the rods as a function of rod position and are necessary to calculate the rod position dependent on the total hot channel factor.

Assuming that the radial and axial power distributions are separable, the total hot channel factor (F_Q^N) is composed of two independent components: the enthalpy rise nuclear hot channel factor ($F_{\Delta H}^N$) and the axial nuclear hot channel factor (F_Z^N), which is defined as the ratio of F_Q^N to $F_{\Delta H}^N$. The procedure for calculating F_Z^N has been discussed above, while the method of calculating $F_{\Delta H}^N$ was presented in the preceding section of this chapter. Notice that for a partially inserted rod bank two $F_{\Delta H}^N$'s exist: one for the rodged portion and one for the unrodged portion.

An "average" $F_{\Delta H}^N$ was calculated to determine F_Q^N as a function of rod position by weighting the two $F_{\Delta H}^N$'s by the fraction of power generated above and below the rod bank at a given rod bank position. This average $F_{\Delta H}^N$ was then multiplied by the appropriate F_Z^N to give F_Q^N at the given rod bank position. In this manner then, a curve of F_Q^N vs rod position is generated.

The usual method consisted of following the rod motion which must occur between zero and full power if the boron concentration remains constant. This means that not only do the rods move but also the power level and associated reactivities change. These variations are accounted for by using the maneuvering band limit curves to indicate the rod position at a given power level.

3. Integral Rod Worths

The integral rod worth calculations were performed using the two-group version of the PDQ-03 diffusion code¹. Two core models were used. The first, and more nearly exact, was a quarter-core model employing Sha's⁵ logarithmic derivative representation for the individual RCC pins. This model was capable of calculating the rod worth of the complete rod pattern and, if necessary, the pattern worth with the center rod "stuck", "ejected", or "dropped".

The remaining stuck (ejected) rod analyses were performed with a PDQ-03 full core model. This second model required two different representations of the RCC poison. The outer four rows and columns of assemblies in the core used a full assembly homogenization scheme with the required grossness in mesh description, while the entire fourth quadrant and remaining assemblies used the "three-square-ring" homogenization and the accompanying finer mesh as defined by Sha⁵. The resulting mesh description was such that some assemblies in the second quadrant of the core were described entirely by only four mesh points, while the assemblies in the fourth quadrant were described by twenty-five points.

The full assembly homogenization was accomplished by first running a unit assembly PDQ which described each RCC pin as a two region, absorber and moderator, cell. The PDQ picture for this type of problem is composed of four quadrants of four adjacent assemblies. The water slots between the assemblies then run vertically and horizontally through the center of the picture. This type of picture allows the PDQ to construct a core which will have a complete rod-cluster in half the assemblies and RCC water holes in the remaining assemblies.

For the "exact" RCC description in the unit assembly, the Ag-In-Cd absorber was described as a separate center region, while the absorber clad, the rod guide thimble, and the cell moderator were smeared into the surrounding square annulus (after Sha). A second unit assembly PDQ was then run with the individual RCC cells smeared throughout the fuel. The effect of control rods was approximated by increasing the macroscopic absorption cross sections of the fuel while keeping the fast-to-thermal absorption ratio (t_1) of the added poison the same as that calculated by the exact case. Various amounts of poison were added until the eigenvalue of the smeared unit assembly matched that of the exact assembly. The three-ring homogenization was performed as described by Sha.

Since the PDQ-03 full core model employed two different RCC models and different mesh descriptions, a PDQ-04 full core calculation was performed with the three-square-ring homogenization and a uniformly repeated mesh throughout. The PDQ-03 model was found to over predict the rod worth by about 2% of the total rod worth, which was considered adequate for stuck rod analyses.

For each complete rod pattern, a quarter core and a full core PDQ-03 were run. Comparing rod worths, the full core model always predicted the worth within the range of $\pm 4\%$ of the total worth. Subsequent stuck (ejected) rod worth calculations always were taken to be the more pessimistic rod worth, i.e., if the quarter core calculation resulted in a smaller complete pattern worth, the accompanying

stuck rod pattern was normalized to this value. However, the respective radial nuclear hot channel factors ($F_{\Delta H}^N$) were left unchanged.

The unrodded pattern, the shutdown rod pattern, and the shutdown pattern with the center rod stuck (the only cases where the full and quarter core models could be compared) revealed that the full core model consistently under predicts the magnitude of $F_{\Delta H}^N$ by about 12%. Two-thirds of this error is accounted for by the fact that the ring homogenization scheme completely eliminates the RCC water holes. Since the peaking factor in a fuel cell next to an RCC hole is eight percent higher than the average cell in the assembly, there is actually very little error in the hot channel factor due to this homogenization scheme.

The diffusion constants for the ring regions (containing the RCC water holes or rods) were obtained using the simple volume weighting technique suggested by Sha.

4. Differential Rod Worths

The differential rod worths for hot zero and hot full power were also calculated using slab geometry in the FAB code. Again, the rods were simulated by a smeared uniform poison in the axial regions of interest, i.e., if the rods were twenty percent withdrawn, the top eighty percent of the axial FAB core description contained a smeared, uniform poison. The amount of poison was determined by first running two axial FAB's with the rods fully inserted. Each FAB calculation had a separate Σ_{p2} . These calculations also contained a t_1 (ratio of fast Σ_p to thermal Σ_p) obtained from an RCC unit assembly PDQ-03 ($t_1 \approx 0.096$). Each Σ_p calculation yielded a corresponding multiplication factor which could then be plotted vs Σ_{p2} . The Σ_{p2} (and hence Σ_{p1}) could then be interpolated to give the desired bank worth since the Σ_{p2} vs k_{eff} curve was previously found to be linear.

Once the appropriate rod poisons were found, the differential rod worth curves were generated by calculating a differential worth for each ten percent increment in core height. The bumps used to obtain

the differential worths were two percent of the core height. Therefore, the procedure was to move the rods in nine percent of the core height (28.8 steps); calculate an eigenvalue; move the rods in two percent of the core height (6.4 steps) and calculate another eigenvalue. From these data, a point on the differential rod worth could be found.

The hot full power cases had equilibrium xenon calculated for each ten percent step and two percent bump. The hot zero power cases contain no xenon and have the hot zero power temperatures.

5. Moderator Temperature Coefficients and Differential Boron Worths

The method for calculating moderator temperature coefficients was the same as that used for the differential boron worths, only the independent variable (moderator temperature or boron concentration) was changed.

The method consisted of running several LEOPARDS over the range of interest of the independent variable. The resulting two-group diffusion constants were put into a radial, one-dimensional FAB calculation to obtain the dependent variable, namely reactivity. Finally, a family of curves was generated by changing a third parameter (boron concentration in the case of moderator temperature coefficients and moderator temperature in the case of boron worth).

A one-dimensional calculation is necessary because both the moderator temperature coefficient and the differential boron worth are dependent upon the power distribution. The reason is that the radial enrichment variation flattens the radial power distribution with the result that there is more radial leakage than the usual geometrical buckling calculation would indicate. Since there was no enrichment variation axially, the axial geometric buckling was sufficient. The one-dimensional calculation smears the water holes and water slots into the fuel.

As a test of the effect of one-dimensional smearing of water holes and water slots, two-dimensional PDQ-03 calculations were made of the

moderator temperature coefficient at a few points in common with points treated by one-dimensional analysis. The results of the calculations are compared in Table 2.1.

TABLE 2.1
COMPARISON OF ONE-DIMENSIONAL AND TWO-DIMENSIONAL
CALCULATIONS OF MODERATOR TEMPERATURE COEFFICIENT

Moderator Temperature (°F)	Moderator Coefficient	
	One-Dimensional FAB (pcm/°F)	Two-Dimensional PDQ-03 (pcm/°F)
70	3.1	3.0
200	5.5	5.2
550	5.4	5.2
500	2.5	3.1
Conditions: No Doppler Effect All RCC Units Out 2500 ppm Boron		

The variation of moderator temperature coefficient with control bank insertion was obtained by running axial FAB'S for various control bank positions with corresponding just critical boron concentrations. The change in reactivity for a given change in moderator temperature was obtained in the usual way, namely, by varying the moderator temperature by ten degrees and calculating the cigenvalues.

The calculation was also "verified" by using PDQ-03 to calculate the end points, i.e., the temperature coefficients for the cases with the rods fully inserted and fully withdrawn.

This calculation was performed for the hot full power and hot no power conditions and extrapolated to obtain the cold predictions.

6. Power Coefficients

The power coefficient curve was calculated with Sha's correlation³ and the zero-dimensional LEOPARD code. The power coefficients were calculated at each ten percent increment of power from zero power to 120 percent of power with eigenvalue calculations performed at the five percent power points on each side of the power of interest. The eigenvalue calculations were performed with LEOPARD. The correlation of power and temperature used to predict the appropriate temperatures for LEOPARD were calculated using Sha's model.

For the five percent increments, the moderator temperature was held constant at the temperature corresponding to the power level of interest (that power at which the power coefficient was being calculated). However, the moderator temperature was changed at each ten percent step in power to get the spectrum effect with power level. The temperature was maintained for the actual coefficient calculation in order to prevent the moderator temperature coefficient from appearing in the power coefficient (because of the programmed average moderator temperature with power level).

B. Methods of Empirical Correction

It has been recognized that the analytical methods in use in the period during which the San Onofre reactor was designed were not in all cases correct in an absolute sense. Physics programs executed during the startup of previous nuclear plants had yielded experimental data sufficient not only to identify areas of discrepancy between predictions and actual observations but also to provide indications of the magnitudes and directions of consistent empirical correction terms to be applied to analytical results. It is the purpose of this section to set forth the basis of certain empirical bias factors which were incorporated with the analytical studies described in the preceding section in generating "best expected" values of the significant nuclear design characteristics of the San Onofre core.

To this end, a brief review will be made of predicted and measured values of several basic nuclear design parameters in the three Westinghouse 120 assembly cores, Centrale E. Fermi/Trino, Italy (SELNI), Societe D'Energy Nucleaire Franco-Belgedes Ardennes/Chooz, France (SENA), and Indian Point Unit 1, Core B/Buchanan, New York (IPP-1 (B)), which preceded the San Onofre Nuclear Generating Station in service. Quantities of particular interest are the "all rods out" critical boron concentrations and the integral control rod worths. The critical boron concentrations are derived strictly by chemical analysis of reactor coolant samples secured under prescribed conditions; the integral rod worths are derived from evaluations of changes in core reactivity related to changes in control rod group position. An important and sensitive secondary parameter is boron worth, deduced from a combination of changes in boron concentration and integral control rod worths. All quantities examined are in the zero power, beginning of life condition. Predicted values are taken from the appropriate design reports with revisions where applicable; measured values are drawn from the startup reports and from field reports prepared by on-site physics personnel.

1. Basic Design Characteristics of the Reference Cores

The three cores are generally similar in physical design as may be seen by comparing the several entries of Table 2.2. SELNI and SENA are particularly closely related. Indian Point is smaller in cross sectional area with a slightly higher enrichment and a greater control rod density than the other two and operates at lower temperature and pressure.

The relative similarity in physical design is reflected in similarity in nuclear design characteristics. Analytically predicted values of the significant nuclear parameters under consideration here, all under zero power, beginning of life conditions, are compared in Table 2.3.

The differences between control rod worths expected in Indian Point and those predicted for SELNI/SENA are due to basic differences

TABLE 2.2

PHYSICAL DESIGN CHARACTERISTICS

	IPP-1 (B)	SELNI	SENA
Core Dimensions			
Equivalent Diameter	6.5 ft.	8.2 ft.	8.2 ft.
Length	8.46 ft.	8.66 ft.	8.8 ft.
H ₂ O/UO ₂ Ratio			
Core	1.62	1.66	1.70
Cell	1.45	1.53	1.53
Enrichments			
Region 1	2.86 (7)	2.73 (7)	2.95 (7)
Region 2	3.26 (8)	3.12 (8)	3.35 (8)
Region 3	4.08 (10)	3.90 (10)	3.75 (9)
Average	(3.40)	(3.25)	(3.35)
Number of Control Rods			
Total Bank	21	28	30
Control Group	5	10	10
Moderator Average Temperature	480°F	544°F	547°F
Pressure	1550 psi	2015 psi	2015 psi

TABLE 2.3

SIGNIFICANT ANALYTICAL NUCLEAR DESIGN CHARACTERISTICS

	IPP-1 (B)	SELNI	SENA
Installed Reactivity (Hot, No Poison, No Control)	1.18	1.17	1.18
Control Rod Worths			
Total Bank Worth			
Hot	14.1%	7.9%	8.4%
Cold	11.0%	6.2%	5.9%
Control Group Worth			
Hot	1.9%	2.6%	2.8%
Cold	Not Available	Not Available	Not Available
Boron Worths			
Hot	-7.4 (6.9)* pcm/ppm	-7.4 pcm/ppm	-7.0 pcm/ppm
Cold	-8.7 pcm/ppm	-9.0 pcm/ppm	-9.2 pcm/ppm
	(*Value adjusted to 545°F, 2015 psi, conditions for comparison)		
Critical Boron Concentrations			
All Rods Out			
Hot	1950 ppm	1910 ppm	2110 ppm
Cold	1975 ppm	Not Available	2026 ppm

in number, configuration and density of control rods in the respective cores. Differences in boron worth are attributable to differences in the water-to-fuel ratios and in the operating conditions of the cores.

2. Results of Experimental Measurements on the Reference Cores

In keeping with normal practice, experimental measurements of a number of nuclear parameters were made at the startup of the successive reactors to verify analytical predictions. Neither the startup programs, prescribing the measurements made, nor the methods of measurement were the same in all three cases. However, sufficient similarity in scope and in basic method existed to allow comparison of the results both among themselves and with respect to analytical prediction. Tables 2.4 and 2.5 summarize the results of a number of comparable measurements of parameters together with corresponding analytically predicted values (taken from Table 2.3) where these exist. Included for purposes of comparison are tabulations of the ratios of values in the hot and cold conditions, respectively, of both predicted and measured control rod worths.

Upon examination of the several values in Tables 2.4 and 2.5 several points are evident. In particular:

- (a) Observed critical boron concentrations are consistently lower than the predicted values by 50 to 150 ppm.
- (b) Measured boron worths show marked scatter around the predicted values. The degree of scatter is apparent in the comparison of the ratios of cold-to-hot boron worths obtained by measurement and analysis, respectively. Parenthetically, the predicted ratios are seen to be in consistently good agreement with the basic water density ratios as is expected on basic theoretical grounds.
- (c) Measured values of control rod worth tend to be lower than predicted. Again, the ratios of measured hot and cold total bank worths are in poor agreement with the ratios of predicted values. In the particular case of SENA, the relatively good agreement of

TABLE 2.4

CRITICAL BORON CONCENTRATIONS, BORON WORTHS,
AND MODERATOR TEMPERATURE COEFFICIENTS

	<u>INDIAN POINT</u>	<u>SELNI</u>	<u>SENA</u>
<u>Cold Zero Power</u>			
Critical Boron (ppm)	1949	1910	2040
Calculated	1949	1910	2040
Measured	1897	1800	1885
Difference	52	110	155
Boron Worth (pcm/ppm)			
Calculated	-8.8	-9.2	-9.0
Measured	-8.0	-10.0	-9.0
<u>Hot Zero Power</u>			
Critical Boron (ppm)			
Calculated	1976	1910	2110
Measured	1893	1840	1972
Difference	83	70	138
Boron Worth (pcm/ppm)			
Calculated	-7.5	-7.4	-7.0
Measured	-7.9	-7.2	-8.3
<u>Cold/Hot Ratio</u>			
Calculated	1.17	1.24	1.29
Measured	1.01	1.39	1.08
Water Density Ratio	1.19	1.30	1.30
<u>Temperature Coefficient (pcm/°F)</u>			
Predicted*	-0.7	-2.0	-1.7
Measured	+1.0	-1.0	-1.3
Predicted**	+0.8	-2.0	-1.0
<u>Method of Measurement</u>	Dynamic (Reactivity) Computer	Static (Asymptotic) Periods	Dynamic (Reactivity) Computer

*At zero power nominal core average temperature, control rods out, measured critical boron.

**At zero power nominal core average temperature, control rods out, predicted critical boron.

TABLE 2.5
COMPARISON OF MEASURED AND PREDICTED
VALUES OF CONTROL ROD WORTHS

	IPP-1 (B)	SELNI	SENA
Total Bank Worth			
Hot Zero Power			
Calculated	14.1%	7.9%	8.4%
Measured	12.5%	7.2%	8.1%
Cold Zero Power			
Calculated	11.0%	6.2%	5.9%
Measured	8.6%	6.2%	5.6%
Hot/Cold Ratio			
Calculated	1.28	1.27	1.42
Measured	1.46	1.33	1.45
Control Group Worth			
Hot Zero Power			
Calculated	1.9%	2.6%	2.8%
Measured	2.0%	2.4%	2.7%
Cold Zero Power			
Calculated	Not Available	Not Available	Not Available
Measured	Not Available	2.1%	1.9%
Hot/Cold Ratio			
Calculated	Not Available	Not Available	Not Available
Measured	Not Available	1.12	1.42
Method of Measurement	Dynamic (Reactivity) Computer	Static (Asymptotic) Periods	Dynamic (Reactivity) Computer

the ratios of predicted and measured values may be more apparent than real in that the ratio of predicted values incorporates a suspect cold value. Both ratios are inconsistent with comparable ratios from the other reactors and with basic physical considerations.

In the particular case of total bank worths, it should be recognized that, because of the shutdown capability criterion, it is difficult to measure the differential worth of the total bank. In fact, very seldom can the total bank be inserted far enough to reach the apex of the differential rod worth curve. As a result, it is generally necessary to estimate the shape of the differential curve over the lower half of rod insertion. This estimate is usually not just an extrapolation since the last half of the differential curve for a large worth bank is not monotonic. The results of the comparison of experimental and calculated total bank worths should be considered with this in mind.

By way of comparison, note should be taken of the close agreement between the calculational and experimental rod worths for the control groups where it is possible to travel the entire length of rod insertion without violating the shutdown capability criterion. In these cases, the entire differential worth function can be measured; hence, no estimate of curve shape or of the value of the integral rod worth need be made.

3. Evidence of Consistent Bias Factors

One very definite conclusion can be drawn here. That is the consistent discrepancy between predicted and measured critical boron concentration is indicative of a basic fault in analysis. This fault has, of course, been previously noted and has been the subject of detailed investigation. The preponderance of evidence indicates that the fault manifests itself most directly in terms of a consistent analytical underestimation of boron worth.

The evidence tabulated in Table 2.4 leads to the conclusion that the analytical prediction of boron worth is, indeed, in error. However, the error can be significantly reduced by the introduction of an empirical bias factor operating on the boron worth and, hence, also on the critical boron concentration.

Reference to Table 2.4 yields the following bias factors:

	<u>SELNI</u>	<u>SENA</u>	<u>IPP-1 (B)</u>
Hot	1.038	1.070	1.044
Cold	1.061	1.082	1.027

with an average value of the order of 1.05, where the bias factor operates directly on boron worths and inversely on just critical boron concentrations.

The effect of the error in calculated boron worth on the moderator temperature coefficient is the prediction of a value for the temperature coefficient which is correct in terms of macroscopic core absorption cross section but biased in terms of calculated equivalent boron concentration. Predicted values of moderator temperature coefficient quoted for given control rod configurations where boron concentration is controlled in the calculation to a unity eigenvalue have been found to agree closely with measured values in the same control rod configuration when boron concentration is again necessarily controlled to criticality. The differences between calculated critical boron concentrations and measured critical concentration reflect the previously identified five percent bias.

Also, data from operating Westinghouse PWR's indicates that the bias factor may vary with moderator temperature and probably will disappear when the control rods are inserted.

III. FINAL DESIGN PREDICTIONS

The methods of analytical calculation and empirical adjustment outlined in the preceding sections were pursued through initial survey studies, establishment of a reference core design and finally development of a set of "best" predicted values of those nuclear design characteristics which could be verified by experimental measurement at plant startup.

Summarized below are the significant predicted values of relative power distributions, control worths, and kinetics coefficients which were generated in anticipation of the zero power and power escalation test program.

1. Power Distributions

Figures 3.1 through 3.3 show the predicted x-y power distributions, normalized to an average value of unity for all fuel assemblies in an octant of the core. Since the core has one-eighth core symmetry both in fuel loading and in control rod configuration, the tabulated values reflect to all fuel assemblies in the core in all cases shown. Figures 3.4 and 3.5 present the predicted behavior of the axial hot channel factor and total hot channel factor, respectively, as functions of Control Group 2 position in the hot, zero power and full power operating conditions.

2. Control Worths

Table 3.1 lists the integral RCC group worths anticipated for the normal operational control group patterns, for the total bank and for the worst RCC unit ejected, stuck and dropped from appropriate configurations.

For the particular case of RCC unit ejection at power where Control Group 2 would necessarily be partially withdrawn, an approximate determination was made of the reactivity insertion associated with ejection of the center RCC unit (RCC H-8) as a function of Control Group 2 position. The resultant predicted function is shown in Figure 3.6. Only the directly calculated analytical predictions are given since no basis exists for introducing empirical bias factors in control rod worth predictions.

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Unit 1, Cycle 1 Startup

Figure 3.1 Relative Assembly-wise Power Densities in the x-y Plane
With No RCC Units Inserted at Startup

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Unit 1, Cycle 1 Startup

Figure 3.2 Relative Assembly-wise Power Densities in the x-y Plane
With Control Group 2 Inserted at Startup

San Onofre Nuclear Generating Station
Unit 1, Cycle 1 Startup

Figure 3.3 Relative Assembly-wise Power Densities in the x-y Plane
With Control Groups 1 and 2 Inserted at Startup

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Unit 1, Cycle 1 Startup

Figure 3.4 Axial Nuclear Hot Channel Factor for the Hot Full Power and Hot Zero Power Conditions

San Onofre Nuclear Generating Station
Unit 1, Cycle 1 Startup

Figure 3.5 Total Nuclear Hot Channel Factor From Zero to Hot Full Power
Control Group Follows the Upper Maneuvering Band Limit

TABLE 3.1

SUMMARY OF PREDICTED INTEGRAL RCC GROUP
AND UNIT WORTHS UNDER VARIOUS CONDITIONS

Configuration	Cold, Zero Power Condition	Hot, Zero Power Condition	Full Power Condition
Control Group 2 Moving in Sequence	2080 pcm	2540 pcm	2540 pcm
Control Group 1 Moving in Sequence	1630 pcm	2000 pcm	2000 pcm
Shutdown Groups 1 and 2 Moving in Sequence	2090 pcm	2560 pcm	2560 pcm
All RCC Units Moving as a Bank	5800 pcm	7100 pcm	7100 pcm
Worst Ejected RCC Unit From Control Group 2 (RCC H-8)	< 0.4%	< 0.4%	< 0.4%
Worst Ejected RCC Unit From Control Groups 1 and 2 (RCC M-12)	< 0.5%	< 0.6%	< 0.6%
Worst Stuck RCC Unit With All Other RCC Units In (RCC H-8)	< 0.9%	< 1.0%	< 1.0%
Worst Single Dropped RCC Unit With All Other RCC Units Out (RCC H-8)		< 0.25%	< 0.25%

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Unit 1, Cycle 1 Startup

Figure 3.6 Maximum Reactivity Insertion Associated With Ejection
of RCC Unit H-8 From Control Group 2 At Power

Differential RCC Control Group worth predictions were made parametrically over a range of values of interest both in the hot, zero power and in the full power conditions. The results are shown in Figures 3.7 and 3.8, respectively.

Predicted values of critical boron concentrations in two-dimensional RCC Group configurations (so called "endpoint" boron concentrations) and differential boron worths under various conditions suffer from previously discussed analytical errors and necessitated empirical adjustment in preparing "best expected" values for comparison with measurement. The analytically predicted and "best expected" values of the endpoint boron concentrations and the differential boron worths are compiled together with the bias factors used in Table 3.2 for the cold and hot, zero power conditions in which corresponding measurements were programmed.

3. Kinetics Coefficients

Predictions of isothermal (moderator plus fuel) temperature coefficients were made by controlled calculations, maintaining the calculated eigenvalue very close to unity, to approximate actual operating control configurations in the cold and hot, zero power conditions. The results of the calculations are presented in Figure 3.9 as functions of "best expected" boron concentrations for convenience in making direct comparisons with measured values. Corresponding "best expected" endpoint boron concentrations of RCC Control Group configurations of interest are identified as alternate reference points at which to compare predicted and measured values of the isothermal temperature coefficient in the event that the anticipated empirical bias factors on boron worth prove incorrect. A formulation for estimating values of isothermal temperature coefficients under typical conditions during power operation is compiled in Figure 3.10 in terms of the detailed hot, zero power predicted values.

Figure 3.11 shows the predicted differential power ("Doppler") coefficient function under power operation conditions at startup. Again, no empirical bias factor is called for; accordingly only the results of direct analytical prediction need be presented.

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Unit 1, Cycle 1 Startup

Figure 3.7 Parametric Differential RCC Group Worth Functions
in the Hot, Zero Power Condition at Startup

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Unit 1, Cycle 1 Startup

Figure 3.8 Parametric Differential RCC Group Worth Functions
in the Full Power Condition at Startup

TABLE 3.2

PREDICTED CRITICAL BORON CONCENTRATIONS
AND BORON WORTHS AT STARTUP

	Cold, Zero Power Condition			Hot, Zero Power Condition		
	Analytically Predicted	"Best" Expected	Empirical Bias	Analytically Predicted	"Best" Expected	Empirical Bias
RCC GROUP CONFIGURATIONS	ENDPOINT BORON CONCENTRATIONS					
All RCC Units Out	2380 ppm	2240 ppm	$(1.05)^{-1}$	2620 ppm	2520 ppm	$(1.05)^{-1}$
Control Group 2 In	2150 ppm	2030 ppm	$(1.05)^{-1}$	2250 ppm	2160 ppm	$(1.05)^{-1}$
Control Groups 1 and 2 In	1980 ppm	1862 ppm	$(1.05)^{-1}$	1954 ppm	1880 ppm	$(1.05)^{-1}$
	DIFFERENTIAL BORON WORTHS					
Control Group 2 Moving	$-8.96 \frac{\text{pcm}}{\text{ppm}}$	$-9.41 \frac{\text{pcm}}{\text{ppm}}$	(1.05)	$-6.82 \frac{\text{pcm}}{\text{ppm}}$	$-7.2 \frac{\text{pcm}}{\text{ppm}}$	(1.05)
Control Group 1 Moving	$-8.99 \frac{\text{pcm}}{\text{ppm}}$	$-9.44 \frac{\text{pcm}}{\text{ppm}}$	(1.05)	$-6.85 \frac{\text{pcm}}{\text{ppm}}$	$-7.2 \frac{\text{pcm}}{\text{ppm}}$	(1.05)

San Onofre Nuclear Generating Station
Unit 1, Cycle 1 Startup

Figure 3.9 Predicted Differential Isothermal Moderator Temperature Coefficients for Hot and Cold Zero Power Conditions

San Onofre Nuclear Generating Station Unit 1 Cycle 1

$$\left(\frac{\partial \rho}{\partial T} \right)_{\text{Actual}} (h) \left(\frac{\text{pcm}}{^\circ\text{F}} \right) = \left(\frac{\partial \rho}{\partial T} \right)_{\text{Zero Power}} (h) \left(\frac{\text{pcm}}{^\circ\text{F}} \right)$$

$$\begin{aligned} & \text{Boron Correction} & + \left\{ -0.8 \times \frac{[B(h) - B(h)]_{\text{Zero Power Actual}} (\text{ppm})}{100} \right\} \left(\frac{\text{pcm}}{^\circ\text{F}} \right) \\ & \text{Power Level Correction} & + \left\{ -1.3 \times \frac{Q(\%)}{100} \right\} \left(\frac{\text{pcm}}{^\circ\text{F}} \right) \\ & \text{Xenon Worth Correction} & + \left\{ 1.4 \times \frac{\Delta \rho_{Xe} (\text{pcm})}{-2700} \right\} \left(\frac{\text{pcm}}{^\circ\text{F}} \right) \\ & \text{Depletion Correction} & + \left\{ -1.4 \times \frac{\text{Burnup (MWD/MTU)}}{1000} \right\} \left(\frac{\text{pcm}}{^\circ\text{F}} \right) \\ & \text{Moderator Temperature Correction} & + \Delta \left(\frac{\partial \rho}{\partial T} \right) \left(\frac{\text{pcm}}{^\circ\text{F}} \right) \end{aligned}$$

Figure 3.10 Predicted Differential Isothermal Temperature Coefficient Formulation for Typical Operating Conditions

San Onofre Nuclear Generating Station
Unit 1, Cycle 1, Startup

Figure 3.11 Predicted Differential Power Coefficient Function
for Power Generation Conditions at Startup

IV. EXPERIMENTAL METHODS

A. Basic Methods of Measurement

1. Reactivity Oriented Measurements

Two basic methods of measurement of reactor physics parameters were pursued concurrently. In one, the reactivity equivalents of changes in core conditions were evaluated directly by analyzing the flux level responses to the changes in core conditions in terms of the neutron kinetics point model to yield changes in core reactivity. In the other, continuous records of RCC unit positions, moderator average temperature and pressure and reactor coolant temperature rise across the reactor vessel were maintained, while at prescribed points in time samples of reactor coolant, from which boron concentration was to be subsequently determined by chemical analysis, were secured. Correlation of RCC unit position, boron concentration, moderator average temperature and pressure and, where applicable, relative power level yielded a series of core condition statepoints from which the reactivity equivalents of the several parameters could be established indirectly in terms of the reactivity equivalent of a single parameter (either boron concentration or RCC unit position).

2. Power Distribution Measurements

Indications of relative power distribution within the core were obtained by measuring the Mn^{54} activation induced in manganese-bearing steel pellets (Aeroballs) during periods of controlled exposure in fixed axial thimbles in selected fuel assemblies. Activation at measured points in the core was related to local power generation in the vicinity of the points by means of analytically derived relative power distribution factors.

Provision was also made for monitoring gross power generation and distribution by means of thermocouples positioned in the reactor coolant outlet streams above selected fuel assemblies. It was intended

that through this system channel enthalpy rises could be deduced from thermocouple voltages with appropriate isothermal calibration constants and analytically derived factors relating coolant outlet temperature to local power production. Unfortunately, the electrical penetrations initially installed to carry thermocouple signals through the containment vessel walls proved hypersensitive to changes in ambient temperature so that reliable data from the thermocouple system were not available during the period of startup operations covered in this report.

B. Experimental Instrumentation

Experimental data were gathered from both normal plant instrumentation and from special equipment intended primarily for physics test purposes. The conventional plant instrumentation is straightforward and needs no further comment. Two nonstandard components of the experimental instrumentation system merit some elaboration.

1. Reactivity Computer

Core reactivity was continuously monitored during testing by means of an on-line analogue computer which analyzed changes in flux level (detector response in actuality) with time in terms of the point model neutron kinetics equations to yield transient reactivity and reactivity change. Since extensive previous studies had shown that the derivative term in the neutron kinetics equation could be neglected without introducing detectable error in the reactivity solution, the expressions relating reactivity, flux and delayed neutron precursor concentrations were cast in the forms

$$\rho = \beta - \frac{1}{n} \sum \lambda_i C_i^*$$

and

$$\frac{d}{dt} C_i^* = n \beta_i - \lambda_i C_i^*$$

where

$$C_i^* = C_i e^{\lambda_i t}$$

and all other symbols have their accustomed meanings. The computer was programmed to consider six groups of delayed neutrons in analyzing flux response; Table 4.1 lists the basic delayed neutron parameters appropriate to San Onofre Unit 1 Cycle 1 startup.

Although in principle the response of any nuclear channel on scale at a given time could have been used in evaluating changes in system reactivity, only the Westinghouse experimental channel incorporating a special long ion chamber was used to avoid possible interference with the regular plant nuclear instrumentation.

2. Datalogger System

A series of maps relating to power distribution in the core were secured with the aid of the incore instrumentation system. Readout of aeroball activity in successive thimbles was accomplished by passing the irradiated balls on a rotating table before a well columnated crystal and photomultiplier tube. Analogue signals proportional to activity were recorded on a strip chart recorder and were sampled, digitized and recorded in digital form on punched paper tape through the data logging system. Had the fault by the vessel penetrations been rectified in time, incore thermocouple signals reflecting temperature and enthalpy distributions would also have been monitored, sampled, digitized and punched in digital form on paper tape. The punched data were subsequently transmitted via the teletype system to the Westinghouse Atomic Power Division facilities in Pittsburgh for detailed, machine-aided analysis of the aeroball information.

The experimental instrumentation system and the flow of information are shown schematically in Figure 4.1. Details of the incore and excore sensor locations relevant to the core configuration are sketched in Figure 4.2.

C. Calibration

Much of the data used in developing experimental values of the reactor design parameters reported here was taken from displays of normal plant

SAN ONOFRE NUCLEAR GENERATING STATION UNIT 1 CYCLE 1

TABLE 4.1

DELAYED NEUTRON PARAMETERS AT STARTUP

Group	Delayed Neutron Fraction β^i	Effective Delayed Neutron Fraction β_{eff}^i	Decay Constant, λ^i
1	0.000219	0.00021	0.0124 seconds ⁻¹
2	0.001478	0.00143	0.0305
3	0.001371	0.00133	0.111
4	0.002851	0.00276	0.301
5	0.000972	0.00094	1.13
6	0.000343	0.00033	3.00
Σ	0.00723	0.0070	
Delayed Neutron Importance $\bar{I} = 0.97$			

The above parameters vary by less than two percent for the following conditions:

β Set	Temperature	Pressure	Boron Concentration	RCC Group Configuration		
				Shutdown 1 and 2	Control 1	Control 2
	Zero Power With No Xenon					
I	150°F	350 psia	2380 ppm	Out	Out	Out
II	535°F	2100 psia	2690 ppm	Out	Out	Out
III	535°F	2100 psia	2025 ppm	Out	In	In
	Full Power With Equilibrium Xenon					
IV	578°F	2100 psia	2000 ppm	Out	Out	Out

San Onofre Nuclear Generating Station
Unit 1 Cycle 1

FIGURE 4.2 Incore and Excure Sensor Locations at Startup

instrumentation signals. The validity of such information is a matter of concern to plant operators and staff personnel; hence, calibration and verification of the plant instrumentation response and readout are carried out on a regular basis and reported through appropriate channels.

Comparable calibration and verification of the response and readout of the special experimental instrumentation supplied for plant startup normally falls outside of regular plant functions. Since the results of the experimental measurements depend directly on the validity of the input data, the results of the most significant calibrations of the special instrumentation response to and readout of input signals are summarized below as partial indications of the overall reliability of the experimental results.

1. Reactivity Indications

Early in the zero power phase of the startup program, a series of controlled flux level transients were initiated by RCC group withdrawal and insertion to verify that values of reactivity obtained from the reactivity computer were consistent with those obtained by hand analysis of neutron detector responses to the transients. For convenience in analysis, only the special case of step reactivity insertion was considered. Evaluation of core reactivity by on-line analogue analysis was made with signals from two special nuclear channels, the long ion chamber channel used throughout the program and a separate movable ion chamber channel. Comparative values of reactivity were obtained by analysis in terms of asymptotic startup rates derived from plant startup rate meters and from detector response traces directly and by graphical analysis of the detector response traces in terms of flux ratios. The results of the several modes of reactivity evaluation are compared for the successive transients in Table 4.2.

2. Nuclear Detector Linearity

As the power escalation part of the program progressed, the previously assumed linearity of the Westinghouse long ion chamber response to change in reactor flux level and the concurrent independence of

TABLE 4.2

COMPARISON OF THE RESULTS OF EXPERIMENTAL DETERMINATIONS OF CORE REACTIVITY AT STARTUP

Run No.	Direction of Reactivity Step	Reactivity Computer Results		Asymptotic Startup Rates		Startup Rate Meter		Flux Ratio Analysis	
		W LIC	W MIC	NE 1203 B	NE 1204 B	NE 1203 B	NE 1204 B	NE 1203 B	NE 1204 B
1	Positive	+60 pcm	+60 pcm	+60 pcm	+63 pcm			+57 pcm	+61 pcm
1	Negative	-45 pcm	-45 pcm					-45 pcm	-45 pcm
2	Positive	+85 pcm	+85 pcm	+82 pcm	+83 pcm			+81 pcm	+82 pcm
3	Positive	+100 pcm		+98 pcm	+103 pcm			+101 pcm	+100 pcm
4	Positive	+97 pcm	+96 pcm	+97 pcm	+92 pcm	+98 pcm	+82 pcm	+97 pcm	+98 pcm
4	Negative		-106 pcm					-97 pcm	
5	Positive	+105 pcm	+103 pcm			+112 pcm	+98 pcm	+102 pcm	+108 pcm
5	Negative		-210 pcm					-212 pcm	-222 pcm
6	Positive	+95 pcm	+93 pcm	+91 pcm	+95 pcm	+98 pcm	+83 pcm	+97 pcm	+97 pcm
6	Negative		-195 pcm					-245 pcm	-200 pcm
7	Positive	+103 pcm	+100 pcm	+98 pcm	+100 pcm	+105 pcm	+91 pcm	+102 pcm	+105 pcm
7	Negative	-195 pcm						-200 pcm	-220 pcm

detector response to RCC control group axial position were verified. At successive power levels and over a range of allowed RCC control group positions, indications of reactor flux level derived from the long ion chamber were correlated with indications of reactor power level derived from the plant average ΔT signal which represents the average of the vessel reactor coolant temperature increases for the three separate loops. The results of correlating indicated long ion chamber response with indicated reactor power level are shown in Figure 4.3. It may be noted in passing that the scatter in the data is attributed in large part to uncertainties in reading the transient ΔT signal with a multipoint recorder.

3. Power Level Indication

The plant average ΔT signal which is the electrical average of the decrease in reactor coolant temperature across each of the three steam generators was taken to be the best continuous indication of reactor power level. The individual ΔT signals had been shown to be linear with temperature difference within close approximation during initial checkout and calibration of plant instrumentation. Reactor coolant flow is nearly constant, varying only with cold leg temperature over a relatively narrow range. Hence, the multipoint recorder readout of the plant average ΔT signal could be, and was, considered to be proportional within acceptable error to reactor thermal power level. The recorder readout was calibrated directly against reactor power level at zero power and at approximately 65% of full power, the latter point based on detailed calorimetric balances, and linearity over the full range was assumed.

4. Incore Flux Monitoring

Extensive experience at earlier installations had established the validity of relating Mn-54 activity induced in aeroballs to local flux and power levels in the region of the core in which the aeroballs were activated. In principle, a direct absolute relation of activation to flux level could be derived; in practice a relative relation is entirely adequate for power distribution measurement provided the

FIGURE 4.3 W Long Ion Chamber Response to Changing Power Level at Startup

relation is strictly linear. Calibration checks were made during the initial stages of the power escalation program to verify that linearity between the actual Mn-54 decay signal and the datalogger output display of that signal was preserved. Specific tests allowed determination of the maximum voltage to be applied to the photomultiplier tube and identification of the maximum signal strength which could be processed by the crystal/photomultiplier/datalogger system without detectable loss of information due to nonlinearity or saturation of the equipment.

5. Core Outlet Temperature Monitoring

A paralleling program of calibration checks was initiated to verify the validity of the thermocouple/datalogger system as a means of sensing absolute and relative temperature distributions in reactor coolant outlet flow from the core. The calibration checks were designed to test the linearity of datalogger response to temperature level at core outlet and to permit internal calibration of the individual thermocouples against a reference value under isothermal conditions. The tests showed, in fact, that ambient temperature variations operating on nominally (but not actually) identical metal contacts in the containment vessel shell penetrations lead to the generation of spurious voltage signals. These spurious signals, which followed a definite diurnal cycle, proved to be of sufficient strength to destroy the significance of the basic thermocouple data. More recent improvements in design and fabrication methods have resulted in the installation of compatible electrical penetrations in the incore thermocouple system with a significant enhancement in the reliability and reduction in uncertainty in the thermocouple data. However, these improvements were not available in time to benefit the startup program during power escalation to 395 MWe.

V. EXPERIMENTAL RESULTS

In the following pages the results of experimental measurements of significant nuclear design parameters are compiled. In keeping with usual practice, results are presented at two levels of analysis. The actual data points derived from discrete measurements of the several quantities of interest are first shown in graphical or tabular forms to provide a permanent summary of the detailed data and to allow, where appropriate, visual indication of the density and degree of internal consistency of the data. Subsequently, the results are re-expressed in smoothed or "best experimental value" form for convenience in use in operations and in developing comparisons between analytically predicted and experimentally measured values.

A. Direct Results of Measurements

1. Zero Power Test Program

Nuclear design parameters explicitly examined during the zero power phase of plant startup were:

- a) differential RCC group worths as functions of group position for the two control groups in the hot and cold conditions;
- b) equivalent critical boron concentrations in the "two-dimensional" RCC configurations (each RCC group either fully in or fully out) in the hot and cold conditions;
- c) differential isothermal temperature coefficients as functions of RCC control group positions in the hot and cold conditions;
(where an isothermal temperature coefficient is defined to be the combined moderator and fuel coefficient measured under isothermal conditions);
- d) combined reactivity defect associated with changes in moderator temperature and pressure during reactor coolant system heatup and cooldown between hot and cold conditions;

- e) relative aeroball activations in normal two-dimensional RCC group configurations and in configurations simulating RCC unit ejection in the hot condition;
- f) differential single RCC unit worths as functions of unit position in configurations simulating RCC ejection in the hot condition;
- g) differences in core shutdown in configurations simulating RCC sticking during reactor trip in the hot condition.

From the results of direct measurement, there were deduced values of:

- h) differential boron worths as functions of boron concentration in the hot and cold conditions.

Finally, as correlative tests of data validity, there were generated values of:

- i) changes in critical boron concentrations as functions of changes in core reactivity due to RCC group insertion in the hot and cold conditions;
- j) differential RCC group worths of functions of group position and moderator average temperature during reactor coolant heatup and cooldown.

In all cases moderator temperatures and pressures in the "hot" and "cold" zero power conditions are defined to be 535°F, 2100 psia and 150°F, 350 psia, respectively.

Differential RCC Control Group Worths

Figures 5.1 and 5.2 show the results of direct experimental determination of the differential worths of Control Groups 1 and 2 in the hot and cold conditions, respectively. The plotted points show, in general, the individual results of several independent determinations of worth, although no attempt has been made on the plots to distinguish data secured in different stages of the program. As experimental operations proceeded, it was observed that on occasion a sub-group of RCC units within the moving control group could execute one step more or less than intended and that on succeeding

Figure 5.1

Measured Differential Worths of Control Groups
in the Hot, Zero Power Condition at Startup

San Onofre Nuclear Generating Station Unit 1 Cycle 1

Figure 5.2

Measured Differential Worths of Control Groups
 in the Cold, Zero Power Condition at Startup

movement of the group, the errant sub-group would return to its normal relative position within the group. Since indicated changes in reactivity responded to actual RCC motion but were analyzed in terms of nominal RCC movement, the effect of these temporary mismatches in RCC sub-group stepping was the generation of pairs of alternately high and low values of differential worth. Whenever identified, such pairs of data were combined and were re-evaluated as single determinations of worth.

It is noteworthy that the inconel grids which form part of the mechanical structure of the fuel assemblies cause marked local flux depressions resulting in readily identified perturbations in differential group worth as the tips of the RCC units pass through the regions dominated by the grids.

Critical Boron Concentrations

Listed in Table 5.1 are the equivalent critical boron concentrations corresponding to the several two-dimensional RCC configurations established during the zero power program of measurement. The tabulated values of boron concentration are each the average of at least three successive determinations of the equivalent critical concentration and represent the algebraic sum of observed boron concentration and the boron equivalents of deviations of core conditions from the prescribed values. In no case did the result of the individual determinations yield values of equivalent critical boron concentration differing among themselves by more than 1 part per million; hence, a single value is quoted. The errors assigned are estimates of the absolute errors to be associated with the determination of boron concentration by chemical analysis.

Isothermal Temperature Coefficient

Concurrent values of change in core reactivity and change in moderator average temperature were secured during successive moderator temperature increases and decreases in several RCC group configurations in both the hot and the cold isothermal conditions. Correlated values of accumulated changes in reactivity and in temperature, each with respect to arbitrary reference

TABLE 5.1

MEASURED CRITICAL BORON CONCENTRATIONS IN THE HOT AND COLD
ZERO POWER CONDITION AT STARTUP

San Onofre Nuclear Generating Station Unit 1 Cycle 1

RCC CONFIGURATION				Equivalent Critical Boron Concentration	
Shutdown Group		Control Group		Hot (530°D) Condition (ppm)	Cold (150°F) Condition (ppm)
1 (steps)	2 (steps)	1 (steps)	2 (steps)		
320	320	320	320	2524 ± 2	2250 ± 2
320	320	320	0	2197 ± 2	2050 ± 2
320	320	0	0	1944 ± 2	1898 ± 2

values, are plotted in Figures 5.3 and 5.4 for the hot and cold conditions, respectively. Data taken on successive swings in temperature under otherwise nominally identical conditions are distinguished by symbols, and empirical linear "best" fits are passed among the data points representing the several swings. The slopes of the respective linear fits which correspond to the measured isothermal temperature coefficients for the individual swings, are tabulated on the Figures.

The differences in slope measured under a given set of conditions are attributed in part to basic experimental uncertainty and in part to the effects of small changes in reactor coolant boron concentration resulting from transfer of water between the reactor coolant system and the pressurizer as fluid density changes. The errors assigned to the average values of the coefficient represent estimates of one standard deviation based on the observed differences in slope.

Reactor Coolant Cooldown/Heatup Reactivity Defect

The changes in core reactivity associated with the combined effects of changes in fuel and moderator temperature and moderator pressure during reactor cooldown and heatup, respectively, under normal operating conditions at start of life are shown as functions of moderator temperature in Figure 5.5. The two determinations, which were carried out under virtually identical conditions of boron concentration and RCC unit configuration, are seen to yield closely similar values for the total reactivity defect between the cold and hot conditions although the trajectories of reactivity versus temperature are noticeably different due to the respective pressure-temperature relations during cooldown and heatup.

Aeroball Activation Measurements in Two-Dimensional RCC Group Configurations

Measurements of the relative distributions of Mn-54 activity induced in aeroballs exposed in incore instrumentation thimbles during operations at a few percent of full power were made both to test the validity of analytical predictions of power sharing under particularly "clean" conditions and to

Figure 5.3

Differential Isothermal Temperature Coefficient Determination
in the Hot, Zero Power Condition at Startup

San Onofre Nuclear Generating Station Unit 1 Cycle 1

Figure 5.4 Differential Isothermal Temperature Coefficient Determination in the Cold, Zero Power Condition at Startup

Figure 5.5 Temperature/Pressure/Reactivity Relations During Reactor Coolant Cooldown and Heatup at Startup.

provide reference data against which to compare similar data obtained in abnormal RCC unit configurations. Aeroball irradiations were made under hot, nominally zero power conditions with all RCC units fully withdrawn and with one and both control groups inserted. The resultant activations were analyzed in terms of both relative axial (z) and relative radial (x-y) power distributions. A typical axial activation scan, chosen at random from the traces obtained, but representative of all such traces in the various configurations is shown in Figure 5.6. The relative radial activations (the ratios of the activation in a given thimble, integrated over the length of the thimble, to the average of all such integrals in a given configuration) are listed for the three configurations considered in Figures 5.7, 5.8, and 5.9.

It is interesting to note that the local depressions in neutron flux level attributed to the effect of increased neutron absorption in the inconel grids in the fuel assemblies are again quite marked.

Measurements in Abnormal RCC Unit Configurations

Experimental determinations were made of the effects of the hypothesized ejection of the calculated most significant RCC unit in configurations involving Control Group 2 alone fully inserted and Control Groups 1 and 2 fully inserted together in the hot, zero power condition. Pseudo-ejection of an RCC unit was accomplished by gradual withdrawal of the selected RCC unit during slow boration of the reactor coolant. The resultant differential RCC unit worth data in the two configurations of interest are plotted as functions of RCC unit position during withdrawal in Figure 5.10. The associated changes in power distribution in the core were monitored by comparing aeroball activations induced in the incore instrumentation thimbles in the two configurations prior to and after pseudo-ejection of the respective RCC units. The ratio of aeroball saturation activities after pseudo-ejection to the corresponding activities before ejection are superimposed on appropriate core cross section maps in Figures 5.11 and 5.12.

Figure 5.6 Typical Plot of Axial Distribution of Aeroball Activation
Obtained in Thimble G-6 With all RCC Units Fully
Withdrawn in the Hot, Zero Power Condition at Startup

San Onofre Nuclear Generating Station Unit 1 Cycle 1

Shutdown Group 1 at 320 steps
 Shutdown Group 2 at 320 steps
 Control Group 1 at 320 steps
 Control Group 2 at 320 steps

Figure 5.7 Relative Saturation Activations in Aeroball Thimbles With all RCC Units Fully Withdrawn From the Core In the Hot, Zero Power Condition at Startup

San Onofre Nuclear Generating Station Unit 1 Cycle 1

Shutdown Group 1 at 320 steps
 Shutdown Group 2 at 320 steps
 Control Group 1 at 320 steps
 Control Group 2 at 0 steps

Figure 5.8 Relative Saturation Activations in Aeroball Thimbles With Control Group 2 Inserted in the Hot, Zero Power Condition at Startup

Shutdown Group 1 at 320 steps
 Shutdown Group 2 at 320 steps
 Control Group 1 at 0 steps
 Control Group 2 at 0 steps

Assembly Containing
 Inserted RCC Unit

Figure 5.9 Relative Saturation Activations in Aeroball Thimbles With Control Groups 1 and 2 Inserted in the Hot, Zero Power Condition

San Onofre Nuclear Generating Station Unit 1 Cycle 1

Figure 5.10 Measured Differential Worths of Pseudo-Ejected RCC Units
In the Hot, Zero Power Condition at Startup

Figure 5.11 Effect on Relative Aeroball Activation of Pseudo-Ejection of RCC H-8 From Control Group 2 Under Hot, Zero Power Conditions at Startup

San Onofre Nuclear Generating Station Unit Cycle 1

Activation with M-12 Withdrawn
Activation with M-12 In Group

Total Core Reactivity Change
in the Pseudo-Ejection of RCC
M-12 is + 0.47 %

○ Aeroball Thimble Location

□ Inserted RCC

■ Pseudo-Ejected RCC

Figure 5.12 Effect on Relative Aeroball Activation of Pseudo-Ejection of RCC M-12 From Control Groups 1 and 2 Under Hot, Zero Power Conditions at Startup

An approximate determination was made also of the integral reactivity worth of the most significant RCC unit when fully withdrawn from the core with all other RCC units fully inserted. The value of the worth so determined simulates the loss of core shutdown capability in the event that a fully withdrawn RCC unit fails to fall when a reactor trip occurs. Since the particular configuration of immediate interest cannot be established in the critical condition without jeopardizing reactor safety, an allowed configuration was established and the worth of the "stuck" RCC unit was assessed by comparing the magnitudes of negative reactivity inserted by reactor trip with the selected unit included in the bank and with the single RCC unit fully withdrawn and temporarily isolated from the bank. The results of the shutdown measurements are listed in Table 5.2.

The absolute experimental values of the negative reactivity insertion for both reactor trips are seen to be significantly larger than the expected -2.3% and -1.3% values, respectively. The source of the discrepancy between the values of apparent shutdown measured at reactor trip and the expected value is believed to be associated with the marked changes in axial flux distribution which occurs when the RCC units pass from a banked configuration low in the core to the fully inserted configuration, particularly as seen by detectors located at the lower quarter plane. Since the RCC unit configurations, with the exception of the center unit, were virtually the same before tripping in both cases, the difference between the indicated shutdown values in the two cases is believed to approximate the worth of the isolated unit.

Indirect Differential Boron Worths

Indirect experimental values of differential boron worth were determined by computing the ratios of integral change in core reactivity due to RCC group movement to the corresponding change in critical boron concentration. The changes in reactivity were evaluated by integrating the differential control group worth functions for the two control groups in the hot and cold conditions. The changes in boron concentration were taken to be

TABLE 5.2

RESULTS OF RCC TRIP TESTS FOR SHUTDOWN MARGIN EVALUATION AT STARTUP

San Onofre Nuclear Generating Station Unit 1 Cycle 1

RCC CONFIGURATION					Core Reactivity (%)
Shutdown Group		Control Group		RCC H-8 (steps)	
1 (steps)	2 (steps)	1 (steps)	2 (steps)		
61	60	60	61	In Group	-4.35
0	0	0	0	In Group	
59	59	59	59	320	-3.40
0	0	0	0	320	

Worth of RCC H-8 if fully withdrawn from the all-RCC's-In configuration at startup is 0.95%.

Minimum shutdown boron concentration in the hot, zero power condition at startup is 1910 ppm (yielding 1% shutdown with RCC H-8 fully withdrawn).

the differences in the equivalent critical boron concentrations associated with the corresponding two-dimensional RCC configurations. The reference data and the resultant differential boron worth values are listed in Table 5.3. The errors assigned to critical boron concentrations are taken from Table 5.1; the errors associated with changes in core reactivity due to RCC group insertion are assumed nominal 3% values.

Boron/Reactivity Correlations

A sensitive test of the validity of experimental data secured during the zero power portion of reactor startup is the correlation of reactor coolant boron concentration values secured as functions of RCC control group positions under prescribed core conditions with the integral reactivity equivalents of the respective control group positions. In principal, plots of measured boron concentration versus measured reactivity defect due to control group insertion should yield virtually linear relations, the slopes of which are alternate indirect values of differential boron worth. If the changes in boron concentration are large, a slight degree of curvature in the fitted relations is expected due to the effect of boron concentration on boron worth.

Correlative plots of boron concentration versus reactivity defect constructed with data secured during reactor coolant boration and dilution operations in the hot and cold conditions, respectively, are presented in Figures 5.13 and 5.14. The scatter in the data about the empirical "best" fits shown as solid lines, in general, representative of the combined measurement uncertainties in reactor coolant boron concentration and in physical core condition data normally associated with dynamic statepoints.

Temperature Dependent Differential RCC Control Group Worths

A second check on basic validity of experimental data secured at zero power is the comparison of differential RCC control group worth values secured under transient reactor coolant heatup/cooldown conditions with data obtained during reactor coolant boration or dilution at constant

TABLE 5.3

INDIRECTLY MEASURED DIFFERENTIAL BORON WORTH
IN THE HOT AND COLD ZERO POWER CONDITION AT STARTUP

San Onofre Nuclear Generating Station Unit 1 Cycle 1

Shutdown Group		Control Group		Critical Boron Concentration	Change in Core Reactivity	Change in Boron Concentration	Indirectly Measured Boron Worth
1	2	1	2				
steps	steps	steps	steps	ppm	pcm	ppm	pcm/ppm
535°F, 2100 psia							
320	320	320	320	2524 ± 2	-2504 ± 75	-327 ± 3	-7.7 ± 0.25
320	320	320	↓ 0	2197 ± 2	-2001 ± 60	-253 ± 3	-7.9 ± 0.3
320	320	↓ 0	0	1944 ± 2			
150°F, 350 psia							
320	320	320	320	2250 ± 2	-1999 ± 60	-200 ± 3	-10.0 ± 0.3
320	320	320	↓ 0	2050 ± 2	-1484 ± 45	-152 ± 3	± 0.35
320	320	↓ 0	0	1898 ± 2

Figure 5.13 Correlation of Measured Boron Concentration With Measured Integral RCC Control Group Worth in the Hot, Zero Power Condition at Startup

Figure 5.14 Correlation of Measured Boron Concentration With Measured Integral RCC Control Group Worth in the Cold, Zero Power Condition at Startup

temperature. It is expected that the values obtained show a gradual transition from the typical hot zero power worths to typical cold worths as cooldown progresses and conversely during heatup. Measured values of differential RCC group worth obtained during heatup and cooldown are superimposed on empirical "best" fits to the hot and cold constant temperature data in Figure 5.15. Typical values of reactor coolant temperature are correlated with control group position to give an indication as to which of the empirical fits should be more nearly applicable at any given point.

2. Power Escalation Test Program

Nuclear design parameters explicitly examined during the power escalation phase of plant startup were:

- a) combined power and programmed temperature coefficients as functions of reactor power level up to 88% (395 MWe) of full power;
- b) differential RCC control group worths as functions of group position under various conditions of power level and xenon concentration and distribution for Control Group 2;
- c) differential isothermal temperature coefficient under a prescribed set of conditions of control configuration, power level and xenon equilibrium;
- d) radial and axial power distributions as functions of control group position and reactor power level up to 88% (395 MWe) of full power;
- e) integral single RCC unit worths as functions of the controlling group position in configurations simulating single RCC unit ejection from several normal operating configurations;
- f) integral single RCC unit worths as functions of RCC unit location in the radial sense in the reactor in configurations simulating single RCC unit trip from a normal operating configuration;
- g) relative aeroball activations in configurations simulating RCC unit ejection and RCC unit trip from normal operating configurations.

From the results of the direct measurements, there were deduced values of:

Figure 5.15 Measured Temperature Dependent Differential Worth of Control Group 2 at Startup

- h) reactivity coefficients of power level only as functions of reactor power level up to 88% of full power.

Combined and Separated Power Coefficients

Concurrent values of change in reactor power level and programmed temperature and of the resultant change in core reactivity were secured during power level increases and decreases as the power escalation program progressed. Correlated values of accumulated changes in reactivity and in power level each with respect to an arbitrary reference value were plotted, and the slope of the correlating function, representing the differential combined power and temperature coefficient, was determined for each set of data. The contribution of the reactivity effect of change in power level alone (the so-called "Doppler" coefficient) was estimated by subtracting from the values of the combined coefficient the calculated contribution due to the programmed change in moderator average temperature.

The resultant values of the combined power and programmed temperature coefficients and of the power coefficient only are compiled together with the conditions of measurement in Table 5.4. The values of the power only or "Doppler" coefficients are plotted versus power level in Figure 5.16.

Differential RCC Control Group Worths

The results of the direct determinations of differential worth of Control Group 2, obtained during periods of power escalation and stable power generation up to 88% (395 MWe) of full power, are summarized in Figure 5.17. Since the magnitude of control group differential worth is a function of power level and xenon concentration and distribution as well as of group position, data points are differentiated by symbols according to measurement steps and the relevant conditions under which the several measurements were made are noted on the figure. It is noteworthy that the scatter in measured differential worth values over small regions of control group position is

TABLE 5.4

MEASURED VALUES OF POWER COEFFICIENTS OF REACTIVITY AT STARTUP

San Onofre Nuclear Generating Station Unit 1 Cycle 1

Average Reactor Power Level (% of 1346 MWt)	Power and Temperature Coefficient (pcm/%)	"Doppler" Coefficient (pcm/%)	CONDITIONS OF MEASUREMENT	
			Controlling Group Position (steps)	Xenon Condition
4.5%	-27.4*	-27.4	-	~ None
7 %	-24.0*	-24.0	~ 120	None
10 %	-23.0*	-23.0	-	~ None
15 %	-26.4*	-26.4	~ 130	30 Hours Decay From ~ 260 Step Equilibrium
18 %	-18.8*	-18.8	~ 150	None
29 %	-15.1	-16.8	~ 180	~ Equilibrium at 160 Steps
31 %	-13.9	-16.5	~ 190	None
35 %	-	-15.1	~ 215	Weak Transient From ~ 170 Step Equilibrium
51 %	-17.5	-17.8	~ 155	~ 16 Hours Decay + ~ 4 Hours Recovery From ~ 270 Step Equilibrium
55 %		-13.3	~ 270	~ Equilibrium at 270 Steps
66 %		-12.0	~ 240	~ Equilibrium at 265 Steps
68 %	-10.0	-10.7	~ 235	Weak Transient From ~ 215 Step Equilibrium
77 %	- 9.4	- 9.9	~ 255	Weak Transient From ~ 215 Step Equilibrium
85%	- 8.9	- 9.4	~ 280	36 Hours Decay + 6 Hours Recovery From 240 Step Equilibrium

*Moderator average temperature was nominally constant during measurement.

San Onofre Nuclear Generating Station Unit 1 Cycle 1

Figure 5.16 Measured Values of the "Doppler" Coefficient at Startup

San Onofre Nuclear Generating Station Unit 1 Cycle 1

Symbol	Power Level (% of 1346 MWt)	CONDITIONS OF MEASUREMENT	
		Xenon Condition	Direction of Group Motion
Δ	~ 15	30 Hours Decay From ~ 260 Step Equilibrium	Withdrawal
$\blacktriangle$	~ 50	12 Hours Decay and 12 Hours Recovery From ~ 270 Step Equilibrium	Withdrawal and Insertion
∇	~ 50	16 Hours Decay and 4 Hours Recovery From ~ 270 Step Equilibrium	Withdrawal and Insertion
$\bullet$	~ 30	~ Equilibrium at ~ 160 Steps	Withdrawal
$\circ$	~ 40	~ Equilibrium at ~ 165 Steps	Withdrawal
$\times$	~ 35	Weak Transient From ~ 170 Step Equilibrium	Withdrawal
$\blacksquare$	~ 70	Weak Transient From ~ 215 Step Equilibrium	Withdrawal
$+$	~ 65	~ Equilibrium at ~ 265 Steps	Withdrawal
$\square$	~ 75	Weak Transient From ~ 215 Step Equilibrium	Withdrawal
$\circ$	~ 80	36 Hours Decay and 6 Hours Recovery From ~ 240 Step Equilibrium	Withdrawal

Figure 5.17 Measured Values of the Differential Worth of Control Group 2 During Power Operation at Startup

markedly greater in these measurements than in comparable measurements taken under zero power conditions, reflecting the additional experimental uncertainties introduced by the effects of Doppler and moderator temperature feedback during power operation.

Isothermal Temperature Coefficient

Measurement of the isothermal temperature coefficient proved particularly difficult during operation at power due to the strong coupling of moderator temperature with reactor power level effected by the steam generators and the secondary steam system. Relatively modest changes in temperature and core reactivity lead quickly to compensating changes in power level with significantly larger reactivity effects due to the relative magnitudes of the respective reactivity coefficients. Further, changes in power level trigger prolonged transients in xenon concentration with resultant reactivity changes comparable in magnitude to the temperature induced reactivity changes. As a result, despite a number of attempts, only one valid measurement of isothermal temperature coefficient was obtained during power operation.

The measurement procedure paralleled that followed at zero power but incorporated additional steps to provide an estimate of the strengths and directions of xenon induced reactivity transients during periods of actual measurement. Data points representing correlated values of change of moderator average temperature and change of reactivity associated with change in temperature are shown in Figure 5.18. The slope of the several empirical linear fits to the several sets of data is taken to be the value of the isothermal coefficient and is tabulated together with the relevant conditions of measurement in the figure.

Aeroball Activation Measurements

Measurements of the relative distributions of Mn-54 activity induced in aeroballs exposed in incore instrumentation thimbles during power generation operations were made at several combinations of power level and control group position both to verify that the radial and

San Onofre Nuclear Generating Station Unit 1 Cycle 1

Figure 5.18 Differential Isothermal Temperature Coefficient Determination During Power Operation at Startup

axial power distributions were acceptable and to provide reference data against which to compare similar data obtained in abnormal RCC unit configurations.

Relative radial activations (the ratios of the activation in a given geometry and in a given thimble, integrated over the length of that geometry, to the average of all such integrals in a given configuration) are listed for the configurations investigated in Figures 5.19 through 5.24. In each case the number shown in the upper portion of a given location relates to the rodded region (or geometry) of the core; the number in the lower position relates to the unrodded region.

Relative axial activations (the ratios of the peak activation in a given thimble to the average activation in that thimble) are also listed in parentheses in Figures 5.19 through 5.24.

Measurements in Abnormal RCC Unit Configurations RCC Unit Ejection Studies

Experimental determinations were made of the effects of the hypothesized ejection of the calculated most significant RCC unit from Control Group 2 in typical operating configurations during power generation. Pseudo-ejection of the selected RCC unit (RCC H-8) from the initial highest worth configuration was accomplished by gradual withdrawal of the RCC unit during slow boration of the reactor coolant. Pseudo-ejection of the same RCC unit from lower worth configurations with Control Group 2 further withdrawn was done by observing the change in reactor power level and programmed moderator average temperature resulting from inserting RCC H-8 from fully out to the group bank position. The changes in core reactivity associated with ejection of the RCC unit were evaluated in terms of the compensating changes in boron concentration or power level, as appropriate, and the respective previously determined reactivity coefficients. The resultant values are plotted in Figure 5.25 as functions of control group position together with the corresponding value obtained by pseudo-ejection of RCC H-8 from the fully inserted Control Group at zero power.

San Onofre Nuclear Generating Station Unit 1 Cycle 1

Figure 5.19 Relative Saturation Activities in Aeroball Thimbles at ~10% of Full Power with Control Group 2 Withdrawn 147 steps at Startup

San Onofre Nuclear Generating Station Unit 1 Cycle 1

Shutdown Group 1 at 320 steps
 Shutdown Group 2 at 320 steps
 Control Group 1 at 320 steps
 Control Group 2 at 140 steps
 Power Level ~37 % of 1346 MWt
 ~ 24 hours of Xenon recovery after
 20% load increase.

Figure 5.20 Relative Saturation Activities in Aeroball Thimbles at ~37% of Full Power with Control Group 2 Withdrawn 140 steps at Startup

San Onofre Nuclear Generating Station Unit 1 Cycle 1

Shutdown Group 1 at 320 steps
 Shutdown Group 2 at 320 steps
 Control Group 1 at 320 steps
 Control Group 2 at 140 steps
 Power Level ~37 % of 1346 MWt
 ~36 hours of Xenon recovery after
 20% load increase.

Figure 5.21 Relative Saturation Activities in Aeroball Thimbles at ~37% of Full Power with Control Group 2 Withdrawn 140 steps at Startup

San Onofre Nuclear Generating Station Unit 1 Cycle 1

Shutdown Group 1 at 320 steps
 Shutdown Group 2 at 320 steps
 Control Group 1 at 320 steps
 Control Group 2 at 188 steps
 Power Level 60 % of 1346 MWt
 ~ 12 hours of Xenon buildup with
 RCC compensation.

Figure 5.22 Relative Saturation Activities in Aeroball Thimbles at 60 % of Full Power with Control Group 2 Withdrawn 188 steps at Startup

San Onofre Nuclear Generating Station Unit 1 Cycle 1

Shutdown Group 1 at 320 steps
 Shutdown Group 2 at 320 steps
 Control Group 1 at 320 steps
 Control Group 2 at 265 steps
 Power Level 60 % of 1346 MWt
 XENON in equilibrium at ~ 260 steps.

Figure 5.23 Relative Saturation Activities in Aeroball Thimbles at 60 % of Full Power with Control Group 2 Withdrawn 265 steps at Startup

San Onofre Nuclear Generating Station Unit 1 Cycle 1

Shutdown Group 1 at 320 steps
 Shutdown Group 2 at 320 steps
 Control Group 1 at 320 steps
 Control Group 2 at 280 steps
 Power Level 88 % of 1346 MWt

Xenon in ~ equilibrium at ~ 270 steps

Figure 5.24 Relative Saturation Activities in Aeroball Thimbles at 88 % of Full Power with Control Group 2 Withdrawn 280 steps at Startup

San Onofre Nuclear Generating Station Unit 1 Cycle 1

Figure 5.25 Measured Reactivity Insertions Associated With the Pseudo-Ejection of RCC H-8 at Startup

The associated effects on power distribution in the core were monitored by observing the changes in relative saturation activities induced in aeroballs irradiated under pseudo-ejected RCC unit conditions compared to activities measured under normal conditions. For these purposes only the activations in the initially rodded region of the core need be considered. In Figure 5.26 the ratios of abnormal activation to normal activation are tabulated for pseudo-ejection of the center RCC unit from successively higher bank positions.

RCC Unit Trip Studies

Experimental determinations were also made of the effects of the hypothesized trip of various single RCC units from a typical operating configuration. Static trip of the center RCC unit was accomplished by gradual insertion of the unit (RCC H-8) during slow dilution of the reactor coolant boron concentration. Trip of other individual units was synthesized by exchange in position between the selected individual unit and the center unit. The reactivity worth of the center unit was evaluated in terms of equivalent change in boron concentration; the worths of the other units examined were obtained in terms of the first. The resultant values are summarized in Figure 5.27.

The effects on power distribution were monitored in detail in the extreme cases of the center RCC unit and two peripheral units by comparing aeroball activations induced in the incore thimbles in the abnormal configurations with those in a comparable reference configuration. The resultant ratios of relative saturated activities are compiled in Figures 5.28 through 5.30.

B. Best Experimental Values

1. Zero Power Test Program

The direct results of experimental measurement, as presented in the preceding sections, are inconvenient to use either for guidance in reactor operation or for comparison with the results of analytical

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Figure 5.26 Effect on Relative Aeroball Activation in the Rodded Region of the Core Due to Pseudo-Ejection of RCC H-8 Under Typical Power Operation Conditions at Startup

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Shutdown Group 1 at 320 steps
 Shutdown Group 2 at 320 steps
 Control Group 1 at 320 steps
 Control Group 2 at 260 steps
 Power Level 50% of 1346 MWt

Figure 5.27 Summary of the Measured Reactivity Worths of Statically Tripped Single RCC Units During Power Operations at Startup

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Figure 5.28: Effect on Relative Aeroball Activation in the Unrodded Region of the Core Due to Static Trip (Insertion) of RCC H-8 Under Typical Power Operation Conditions at Startup.

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Figure 5.29 Effect on Relative Aeroball Activation in the Unrodded Region of the Core Due to Static Trip (Insertion) of RCC M-12 under Typical Power Operation Conditions at Startup.

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Figure 5.30 Effect on Relative Aeroball Activation in the Unrodded Region of the Core Due to Static Trip (Insertion) of RCC P-8 Under Typical Power Operation Conditions at Startup

prediction. In the cases of measurements oriented toward the determination of differential and integral changes in reactivity, "best" fits to the respective sets of data were constructed and plots of the fitting functions were prepared in the forms of Figures 5.31 through 5.37, which follow. These fits are all empirical in nature. Where the data are sparse, absolute magnitudes are consistent with measurement but trends are based on analytical prediction; where adequate data exist, both magnitudes and trends are deduced directly from experimental results.

For those measurements dealing primarily with power distribution the measured values of activity induced in aeroballs at discrete points in the core were smoothed and combined with analytical predictions of local neutron flux distributions in the x-y plane to yield gross two and three dimensional power distributions. In the particular cases considered here, the axial components of the distributions are uniform and follow closely the typical distribution shown in Figure 5.6. The relative power sharing in the x-y plane in the three RCC group configurations is tabulated on a normalized assembly average basis in Figures 5.38, 5.39 and 5.40.

2. Power Escalation Test Program

As in the case of the zero power portion of the program, the direct results of measurement, as presented above, are not immediately useable either for operational purposes or for verification of design methods. However, contrary to the zero power case unique values, particularly of reactivity worths of kinetics coefficients and control group worths, are not readily defined due to the influences of the magnitudes and distributions of fuel resonance absorption effects, of xenon and other less rapidly varying fission product isotopic concentrations and of moderator density. Accordingly, the plots of the "best" fits of Control Group 2 differential worth and of differential power coefficient presented in Figures 5.41 and 5.42, respectively, are meaningful only in the sense of "near equilibrium" values with

Figure 5.31 Differential and Integral Worths of Control Group 2
In the Hot, Zero Power Condition at Startup

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Figure 5.32 Differential and Integral Worths of Control Group 1 in the Hot, Zero Power Condition at Startup

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Figure 5.33 Differential and Integral Worths of Control Group 2 in the Cold, Zero Power Condition at Startup

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Figure 5.34 Differential and Integral Worths of Control Group 1 in the Cold, Zero Power Condition at Startup.

Figure 5.35 Correlation of RCC Control Group Position With
Critical Boron Concentration at Startup

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Figure 5.36 Differential Boron Worth at Startup

Figure 5.37 Isothermal Temperature Coefficients
During Zero Power Operations at Startup

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Shutdown Group 1 at 320 Steps
 Shutdown Group 2 at 320 Steps
 Control Group 1 at 320 Steps
 Control Group 2 at 320 Steps

Figure 5.38 Relative Assemblywise Power Sharing With All RCC Units Fully Withdrawn in the Hot, Zero Power Condition at Startup

Figure 5.39 Relative Assemblywise Power Sharing With Control Group 2
 Inserted in the Hot, Zero Power Condition at Startup

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Shutdown Group 1 at 320 steps
 Shutdown Group 2 at 320 steps
 Control Group 1 at 0 steps
 Control Group 2 at 0 steps

 Assembly Containing
 Inserted RCU Unit

Figure 5.40 Relative Assemblywise Power Sharing With Control Groups 1 and 2 Inserted in the Hot, Zero Power Condition at Startup

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Figure 5.41 "Near Equilibrium" Differential and Integral Reactivity Worths of Control Group 2 During Power Operations at Startup

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Figure 5.42 "Near Equilibrium" Differential and Integral Reactivity Changes Due to Operation at Power at Startup

the recognition that the actual values reflecting core behavior at any given time may differ significantly under particular transient conditions from those shown here.

The results of the zero power physics test program yielded sufficient data to justify redefining the zero power isothermal temperature coefficient function in terms of "best experimental" rather than analytically/empirically predicted values. This redefinition carries over, of course, as the basis of the formulation for estimating the value of the isothermal coefficient during power operation. However, the single experimental value obtained for the coefficient at power does not provide a sufficient basis for generating a revised set of "best experimental" correction terms for the formulation. Accordingly, Figure 5.43, summarizing the "best experimental" version of the formulation differs from Figure 3.8 only in the values assigned for the hot, zero power reference points.

The measured values of activity induced in aeroballs in the monitoring thimbles in the core during measurements of power distribution were separated into axial regions ("geometries") corresponding to the rodged and unrodged portions of the core and then independently smoothed and combined with appropriate analytically predicted values of local power distributions in the x-y plane to yield three dimensional power distributions. These three dimensional distributions were then examined on an individual fuel rod basis to establish the magnitudes and locations of the maximum nuclear hot channel factors. Subsequent examination on a core region basis yielded indications of gross power sharing ratios under various operating conditions. The hot channel factor data are summarized in Table 5.5; the gross power sharing ratios are compiled in Table 5.6. Separate edits of the three dimensional power distributions yielded two dimensional distributions for each fuel assembly. Figure 5.44 shows the relative power sharing in the x-y plane on a normalized assembly average basis for the unrodged region of the core under conditions representative of normal full power operations.

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$$\begin{aligned} \frac{\partial \rho}{\partial T} \text{ Actual (h)} \left(\frac{\text{pcm}}{^\circ\text{F}} \right) &= \frac{\partial \rho}{\partial T} \text{ Zero Power (h)} \left(\frac{\text{pcm}}{^\circ\text{F}} \right) \\ &+ \left\{ -0.8 \times \frac{[B \text{ (h)} - B \text{ (h) Zero Power}]}{100} \right\} \left(\frac{\text{pcm}}{^\circ\text{F}} \right) \\ &+ \left\{ -1.3 \times \frac{Q(\%)}{100} \right\} \left(\frac{\text{pcm}}{^\circ\text{F}} \right) \\ &+ \left\{ 1.4 \times \frac{\Delta \rho_{\text{Xe}} (\text{pcm})}{-2700} \right\} \left(\frac{\text{pcm}}{^\circ\text{F}} \right) \\ &+ \left\{ -1.4 \times \frac{\text{Burnup (MWD/MTU)}}{1000} \right\} \left(\frac{\text{pcm}}{^\circ\text{F}} \right) \\ &+ \Delta \frac{\partial \rho}{\partial T} \left(\frac{\text{pcm}}{^\circ\text{F}} \right) \end{aligned}$$

Boron Correction

Power Level Correction

Xenon Worth Correction

Depletion Correction

Moderator Temperature Correction

Figure 5.43 Measured/Predicted Differential Isothermal Temperature Coefficient Formulation for Typical Operating Conditions

TABLE 5.5

MEASURED NUCLEAR HOT CHANNEL FACTORS IN NORMAL CONTROL CONFIGURATIONS DURING POWER OPERATIONS AT STARTUP

Power Level (% of 1346 MWt)	Control Group Position (steps)	Xenon Condition	Radial Hot Channel Factor			Total Hot Channel Factor			Axial Avg. Hot Channel Factor
			Core Maximum (both geometries)		M-8 Maximum (unrodded)	Core Maximum		M-8 Maximum	
			(location)	$F_{\Delta H}^N$ *	$F_{\Delta H}^N$ *	(location)	F_Q^N *	F_Q^N *	F_Z^N
2%	320	None	H-8	1.56	1.54	K-10	2.33	2.29	1.46
10%	147	Transient	K-8	1.56	1.58	M-8	3.08	3.08	1.96
37%	140	Transient	K-12	1.56	1.54	D-8	2.93	2.92	1.87
37%	140	Equilibrium	K-12	1.54	1.56	D-8	2.86	2.86	1.83
60%	188	Transient	H-11	1.52	1.56	D-8	2.66	2.65	1.70
58%	265	Equilibrium	M-8	1.53	1.53	M-8	2.21	2.21	1.40
88%	280	Transient	D-8	1.54	1.57	D-8	2.20	2.19	1.40

*Including +8% allowance for instrument inaccuracy.

TABLE 5.6

MEASURED GROSS POWER SHARING RATIOS IN NORMAL
CONTROL CONFIGURATIONS DURING POWER OPERATIONS AT STARTUP

Power Level (% of 1346 Mw)	Control Group Position (steps)	Xenon Condition	Fraction of Power Below Control Group	Relative Quadrant Power (Unrodded Region)				<u>Inside*</u> <u>Outside</u> Activation Ratio
				NE	SE	NW	SW	
2%	320	None	1.00	0.997	0.987	1.010	1.006	3.91
10%	147	Transient	0.83	0.995	0.987	1.010	1.009	3.67
37%	140	Transient	0.67	0.989	1.011	1.003	0.997	3.60
37%	140	Equilibrium	0.65	0.994	1.013	0.994	0.999	3.60
60%	188	Transient	0.85	0.988	1.023	0.996	0.993	3.31
58%	265	Equilibrium	0.96	0.977	1.036	0.998	0.998	3.09
88%	280	Transient	0.96	0.982	1.035	0.992	0.992	3.01

*Ratio of combined activation in thimbles H-9, G-6 and L-8 to that in thimbles A-8, G-15 and R-8.

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Shut down Group 1 at 320 steps
 Shut down Group 2 at 320 steps
 Control Group 1 at 320 steps

Power Level (%) Control Group 2 Position (steps)

Figure 5.44 Relative Assemblywise Power Sharing in the Unrodded Region of the Core Under Conditions Representative of Full Power Operation at Startup

"Best experimental" values of the worth of the worst RCC unit when ejected from Control Group 2 positions over the allowed range for power operation are plotted in Figure 5.45 corresponding "best experimental" values of the integral worths of individual RCC units when fully inserted from the fully withdrawn position under typical power operating conditions are tabulated and plotted in Figure 5.46. The "best estimated" values of the hot channel factors and of the gross power sharing ratios in the abnormal RCC configurations are compiled in Tables 5.7 and 5.8, respectively.

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Figure 5.45 "Best Estimated" Values of Maximum Reactivity Insertions Associated with Ejection of an RCC Unit from Control Group 2 During Startup Power Operations

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Figure 5.46 "Best Experimental" Integral Worths of Single RCC Units when Tripped from the Fully Withdrawn Position in Normal Power Operating Control Configurations at Startup

TABLE 5.7

"BEST ESTIMATED" NUCLEAR HOT CHANNEL FACTORS IN ABNORMAL CONTROL
CONFIGURATIONS DURING STARTUP POWER OPERATIONS

Power Level (% of 1346 MWt)	Control Group Position (steps)	Abnormal RCC Unit Configuration		Radial Hot Channel Factor		Total Hot Channel Factor	
		Core Location	Position (steps)	Core Maximum		Core Maximum	
				Location	$F_{\Delta H}^N *$	Location	F_Q^{N*}
RCC UNIT EJECTION STUDIES				H-10 G-10 D-8	1.54 1.54 1.56	D-8 D-8 D-8	2.91 2.65 2.34
50%	158	H-8	320				
50%	210	H-8	320				
50%	280	H-8	320				
RCC UNIT TRIP STUDIES				M-8 D-8 E-11	1.56 1.68 1.73	M-8 D-8 E-11	2.29 2.48 2.47
50%	260	H-8	0				
50%	260	M-12	0				
50%	260	P-8	0				

*Includes 8% for instrument accuracy.

TABLE 5.8

"BEST ESTIMATED" GROSS POWER SHARING RATIOS IN ABNORMAL CONTROL CONFIGURATIONS DURING STARTUP POWER OPERATIONS

Power Level (% of 1346 MWt)	Control Group Position (steps)	Abnormal RCC Unit Configuration		Relative Quadrant Power				Local Power Ratio
		Core Location	Position (steps)	NE	SE	NW	SW	
RCC UNIT EJECTION STUDIES								
50%	158	H-8	320	0.989	1.027	0.990	0.994	1.02
50%	210	H-8	320	0.985	1.034	0.989	0.992	1.04
50%	230	H-8	320	0.987	1.029	0.987	0.997	1.03
RCC UNIT TRIP STUDIES								
50%	260	H-8	0	0.980	1.008	1.007	1.005	0.91
50%	260	M-12	0	1.066	1.036	1.033	0.864	0.63
50%	260	P-8	0	1.022	1.089	0.932	0.957	0.71

^{/1} Local Avg. | Perturbed / Local Avg. | Reference

where "Local Average" is the average relative assembly-wise power in the affected assembly and its seven or eight nearest neighbors.

VI. COMPARISON OF PREDICTED AND MEASURED VALUES OF NUCLEAR DESIGN PARAMETERS

A matter of fundamental interest in the San Onofre reactor startup program was the comparison of analytically predicted and experimentally measured reactor characteristics. While the scope of the startup test program was broader than the investigation of nuclear design parameters, for the purposes of this discussion, the more limited area of consideration pursued in the preceding chapters is maintained here as well.

In the following sections of this chapter comparison is made between the "final predicted" values of significant nuclear design parameters developed and presented in Chapter III and the results of direct measurement of the same parameters as summarized in Chapter V. Successive sections are concerned with power distribution, control worth and kinetics coefficient comparisons. Discussion of the significance of agreement or deviation between prediction and measurement are reserved for Chapter VII which follows.

1. Power Distributions

In making direct comparison of analytically predicted and experimentally measured indications of core power distributions, three parameters are particularly significant:

- a) the average absolute discrepancy between predicted and measured values of relative saturation activity for all aeroball channels mapped at a given time, where

$$\text{avg. absolute error} = \frac{1}{N} \sum_{i=1}^N \left| \frac{\text{Measured}|_i - \text{Predicted}|_i}{\text{Predicted}|_i} \right| \times 100\% ;$$

- b) the deviation between measured and predicted values of the Inside/Outside activation ratio, where the numerator and denominator are each the sum of aeroball saturation activities in selected thimbles in the inner and outer regions of the core, respectively;

- c) the relative power sharing among the four nominally symmetric quadrants of the core, where the quadrant tilt factors are obtained by weighting the saturation activity of each aeroball channel in a given quadrant in terms of the number and enrichment of fuel assemblies in its proximity, summing the weighted activities in each quadrant and normalizing the sums to a core average of 4.0. It should be noted that the core loading scheme gives complete quadrant symmetry in number and enrichment of fuel assemblies so that quadrant tilt factors of 1.0 are expected.

The results of making these several comparisons between predicted and measured core power distribution parameters are summarized in Table 6.1. Upon examination of the tabulated values, several observations may be made:

- a) Under nominally zero power conditions, prediction and measurement are in close agreement. On the average the observed relative aeroball activations differ from the predicted values by only two to four percent in the two configurations of prime interest, while the ratios of activations in the central region of the core to those on the periphery are in virtually perfect agreement and the quadrant tilt factors are identical within measurement error.
- b) As power level and core xenon content increase, the average difference between predicted activation and measured activation also increases, particularly in the dominant unrodded region of the core. However, the deviation apparently tends to stabilize as the conditions of measurement approach those typical of normal power operation.
- c) The increasing disagreement between prediction and measurement is definitely not random, as evidenced by the steadily increasing deviation of the measured Inside/Outside activation ratio from the analytical ratio.
- d) It can be seen that quadrant tilt also changes with power level in a non-random fashion with the SE quadrant consistently high at elevated powers. However, the deviation from expected behavior is clearly

TABLE 6.1

COMPARISON OF PREDICTED AND DIRECTLY MEASURED POWER DISTRIBUTION CHARACTERISTICS

Power Level (% of 1346 MWt)	Control Configuration			Xenon Conditions	Avg. Absolute ¹ Error		Inside ¹ Outside Ratio ²	Relative Quadrant Power			
	Shutdown Groups (steps)	Control Groups			Rodded Region	Unrodded Region	Unrodded Region Only				
		1 (steps)	2 (steps)					NE	SE	NW	SW
~2%	320	40	0	None	2.3	-	-	1.000	0.993	0.994	1.015
~2%	320	320	17	None	4.2	-	-	1.001	0.993	0.994	1.012
~2%	320	320	320	None	-	1.9	1.0	0.997	0.987	1.010	1.006
10%	320	320	147	Transient	4.3	3.2	0.94	0.995	0.987	1.010	1.009
37%	320	320	140	Transient	3.3	3.7	0.92	0.989	1.011	1.003	0.997
37%	320	320	140	Equilibrium	4.7	4.2	0.92	0.994	1.013	0.994	0.999
60%	320	320	188	Transient	4.3	5.6	0.85	0.988	1.023	0.996	0.993
58%	320	320	265	Equilibrium	5.5	7.4	0.79	0.977	1.036	0.998	0.998
88%	320	320	280	Transient	8.0	8.9	0.77	0.982	1.035	0.992	0.992

1. $\frac{\text{Measured} - \text{Predicted}}{\text{Predicted}} \times 100\%$

2. $\frac{\text{Activations in H9 + G6 + L8}}{\text{Activations in A8 + G15 + R8}} \div \text{Predicted Ratio}$

leveled off above 60% of full power. The pervading implication of these observations is that as power level and xenon concentration increase, the core experiences a gradual decrease in the fraction of total power generated in its center and a corresponding increase in its outer region. Re-examination of Table 5.5 shows that the location of the core maximum radial hot channel factor, $F_{\Delta H}^N$, gradually migrates from the center assembly (H-8) at relatively low power levels to assembly M-8 in the intermediate region, four assemblies removed from the center, as the reactor approaches typical conditions of power generation. Conversely, the radial hot channel factor for the hottest fuel rod in assembly M-8 remains virtually constant throughout power escalation.

In Figure 6.1 the deviations between predicted and measured relative saturation activities in the individual aeroball thimbles are plotted as functions of distance of the respective thimbles from the core center for measurements made under zero power conditions and conditions typical of power operation. The same comparisons of the differences between prediction and measurement under differing power conditions are presented in terms of differences in the actual assemblywise power distributions in the core in Figure 6.2.

At this point several relevant remarks are in order. First, it must be noted that the analytical values used in making comparisons between measured and predicted power distribution parameters were the same in all cases. In particular, the "hot, full power" analytical power distribution calculations described in Section 3.1 were used as the reference predicted values regardless of the conditions of measurement. Hence, a gradual change in the deviation of measured values from prediction over some region of power level is to be expected.

Second, and again as noted in Sections 2.1 and 3.1, the "hot, full power" calculations of power distribution included no effects due to xenon and assumed spatially uniform effects due to fuel temperature and moderator density. Hence, the better agreement between prediction and measurement

Figure 6.1 Differences Between Predicted and Measured Saturated Aeroball Activations Under Zero Power and Typical Power Operation Conditions

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Figure 6.2 Differences Between Predicted and Measured Assemblywise Relative Power Distributions under Zero Power and Typical Power Operation Conditions

under spatially uniform zero power conditions than under power operating conditions, where xenon, fuel temperature and moderator density all contribute spatially varying effects, is less surprising, although the response of core power distribution to spatially varying effects is somewhat stronger than anticipated.

It is further to be noted at this juncture that core power distribution is not a direct criterion of safe reactor operation at power. More significant considerations in terms of core safety are the magnitudes of the core maximum hot channel factors. In Figures 6.3 and 6.4, comparison is made between predicted and measured values of the maximum axial (F_Z^N) and total (F_Q^N) nuclear hot channel factors under various conditions of power level and controlling RCC group withdrawal. It is evident upon examination of these figures that the magnitudes of the limiting hot channel factors are closely approximated by analytical prediction under a variety of operating conditions and configurations.

2. Control Worths

Since the methods of calculating integral RCC unit and group worths, differential group worths and differential boron worths differ among themselves, comparison will be made between prediction and measurement separately for each of the three sets of parameters. Table 6.2 summarizes the "final predicted" and "best experimental" values of significant RCC group and unit integral worths drawn from Table 3.1 and from the appropriate figures of Chapter V. Comparison of the predicted and measured integral worth of the worst single ejected RCC unit as a function of control group position in typical power operating configurations is made in Figure 6.5.

In Figure 6.6 predicted and measured differential worth functions for the control groups are compared in the hot, zero power condition. In making the comparisons, values of the predicted function were obtained by interpolation among the parametric predictions of Figure 3.7 in terms of predicted integral worth values taken from Table 3.1. The experimental values are the result of direct measurement, reported in as-measured forms

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Figure 6.3 Comparison of Predicted and Measured Axial Nuclear Hot Channel Factors

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Figure 6.4 Comparison of Predicted and Measured
Total Nuclear Hot Channel Factors

TABLE 6.2

COMPARISON OF FINAL PREDICTED AND BEST EXPERIMENTAL VALUES OF RCC GROUP
AND UNIT INTEGRAL WORTHS UNDER VARIOUS CONDITIONS AT STARTUP

Configuration	Cold, Zero Power Testing Condition		Hot, Zero Power Testing Condition		Full Power Operating Condition	
	Final Prediction (pcm)	Best Measurement (pcm)	Final Prediction (pcm)	Best Measurement (pcm)	Final Prediction (pcm)	Best Measurement (pcm)
Control Group 2 (In Normal Sequence)	2080	1999	2540	2504	2540	-
Control Group 1 (In Normal Sequence)	1630	1484	2000	2001	2000	-
Shutdown Groups 1 and 2 (In Normal Sequence)	2090	-	2560	-	2560	-
All RCC Units (Banked)	5800	-	7100	-	7100	-
Worst Ejected RCC Unit (With Control Group 2 Fully Inserted)	< 400	-	< 400	320	< 400	-
Worst Ejected RCC Unit (With Control Groups 1 and 2 Fully Inserted)	< 500	-	< 600	470	< 500	-
Worst Stuck RCC Unit (With All Other RCC Units Fully Inserted)	< 900	-	< 1000	950	< 1000	-
Worst Single Tripped RCC Unit (With All Other RCC Units Fully Withdrawn)	-	-	-	-	< 250	185

Figure 6.5 Comparison of Predicted and Measured Functions of the Maximum Reactivity Insertion Associated with Ejection of the Worst RCC Unit from Control Group 2 During Typical Power Operation at Startup

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Figure 6.6 Comparison of Final Predicted and Best Experimental RCC Control Group Differential Worth Functions in the Hot Zero Power Condition at Startup

in Figure 5.1 and in smoothed form on Figures 5.31 and 5.32. Apart from the localized effects of fuel assembly grids, the agreement is seen to be rather striking.

The corresponding comparison of predicted and measured differential worth functions under typical power operation conditions is shown in Figure 6.7. The predicted function was constructed by interpolation in Figure 3.8; the smoothed measured function is that of Figure 5.41. Clear evidence of consistently higher worth measured than predicted is evident. Here let it be noted that while the analytical calculations properly accounted for the (one-dimensional) spatial effects of xenon and Doppler in terms of the "near equilibrium" conditions of comparison, no account was taken of axial moderator density variation in developing the predicted differential worth function.

Critical boron concentrations and differential boron worths obtained by calculation and experiment are summarized in Table 6.3. The experimental values are taken from Table 5.3 which reflects in part data in Table 5.1. Predicted values were obtained by direct calculation according to the methods described in Section 2.5 and were empirically adjusted to account for previously identified calculational faults as outlined in Section 2.9. Also listed are alternate predicted values of differential boron worth obtained indirectly in terms of critical boron concentrations and integral rod worth given in Tables 3.2 and 3.1, respectively. It is worth noting in passing that the measured values restated from Table 5.3 are more nearly analogous in form to the indirect, rather than the direct, predicted values of Table 6.3. In general, measurement shows consistently higher boron differential worth and critical boron concentration than does prediction. The agreement between indirectly predicted and indirectly measured differential boron worth in the cold, zero power condition is probably fortuitous in that it is based on low values (compared to measurement) of both integral RCC group worths and change in critical boron concentration.

Figure 6.7 Comparison of Final Predicted and Best Experimental Control Group 2 Differential Worth Functions in Nearly Equilibrium Power Operating Conditions at Startup.

TABLE 6.3

COMPARISON OF PREDICTED AND MEASURED VALUES OF CRITICAL
BORON CONCENTRATION AND BORON WORTH AT ZERO POWER AT STARTUP

RCC Group Configuration				Critical Boron Concentrations		Differential Boron Worths		
Shutdown Groups		Control Groups		Final Predicted Value	Best Experimental Value	Final Predicted Value		Best Experimental Value
1.	2	1	2			Direct	Indirect	
steps	steps	steps	steps	ppm	ppm	pcm/ppm	pcm/ppm	pcm/ppm
COLD ZERO POWER TESTING CONDITIONS								
320	320	320	320	2240	2250			
Control Group 2 Moving						-9.4	-9.9	-9.9
320	320	320	0	2030	2050			
Control Group 1 Moving						-9.4	-9.7	-10.0
320	320	0	0	1862	1898			
HOT ZERO POWER TESTING CONDITIONS								
320	320	320	320	2520	2524			
Control Group 2 Moving						-7.2	-7.1	-7.8
320	320	320	0	2160	2197			
Control Group 1 Moving						-7.2	-7.1	-7.9
320	320	0	0	1880	1944			

Examination of the patterns of agreement and deviation between predicted and measured values of critical boron concentrations and differential boron worth show that the use of a single constant empirical bias factor operating independently of temperature and RCC unit configuration on boron worth and boron concentration is not appropriate. Unfortunately, if the comparisons of prediction and measurement reported here are not invoked, little basis exists from previous startup programs to justify construction of varying bias factors.

3. Kinetics Coefficients

Comparison of predicted and measured values of the isothermal (combined moderator and fuel) temperature coefficient is most effectively made in two-dimensional RCC group configurations where each RCC group is either fully in or fully out. Referencing values of temperature coefficient directly to critical RCC group configuration eliminates any effects due to the calculational bias which operates inversely on boron concentration and directly on boron worth. The two-dimensional RCC group configurations allow the additional facility of alternate schemes of calculation. In Table 6.4 the final predicted and best experimental values of the isothermal temperature coefficient, taken from Figures 3.9 and 5.37, are compared in the cold and hot condition. The agreement is seen to be consistently good although there exists a definite bias whereby measured values are more positive than predicted.

Comparisons of values of isothermal temperature coefficient under typical operating conditions are less conveniently made due to the relative large number of parameters affecting the value of the coefficient. Under these circumstances a meaningful comparison is best made by generating analytical values corresponding to the conditions of measurement by introduction of the appropriate parameters in the formulation presented in Figure 3.10. The results of the measurement summarized in Figure 5.18 provide a valid reference against which to test the analytical formulation. In Table 6.5 the conditions of measurement are compiled, the

TABLE 6.4

COMPARISON OF PREDICTED AND MEASURED VALUES OF ISOTHERMAL
TEMPERATURE COEFFICIENT AT ZERO POWER AT STARTUP

RCC Group Configuration				Cold Zero Power Condition		Hot Zero Power Condition	
Shutdown Groups		Control Groups		Final Predicted Value	Best Experimental Value	Final Predicted Value	Best Experimental Value
1	2	1	2				
steps	steps	steps	steps	pcm/°F	pcm/°F	pcm/°F	pcm/°F
320	320	320	320	+3.4	+3.4	+7.2	+7.4
320	320	320	0	+2.2	+2.4	+2.0	+2.3
320	320	0	0	+1.3	+1.6	-2.1	-1.7

TABLE 6.5

COMPARISON OF PREDICTED AND MEASURED VALUES OF ISOTHERMAL TEMPERATURE
COEFFICIENT DURING TYPICAL POWER OPERATING CONDITIONS AT STARTUP

Calculation of Predicted Value

Conditions: Control Group 2 Position	225 steps
Boron Concentration	1969 ppm
Reactor Power Level	60%
Xenon	Equilibrium (~ 2270 pcm)
Burnup	~ 350 MWD/MTU
Moderator Average Temperature	549°F

Evaluation of Formulation

Reference Value (535°F, 225 steps, zero power, startup)	+6.4 pcm/°F
Boron Correction $(-0.8 \times (2493 - 1969)/100)$	-4.2 pcm/°F
Power Level Correction $(-1.3 \times 60/100)$	-0.8 pcm/°F
Xenon Correction $(+1.4 \times -2270/-2700)$	+1.2 pcm/°F
Burnup Correction $(-1.4 \times 350/1000)$	-0.5 pcm/°F
Moderator Temperature Correction (549°F, 1969 ppm)	-0.4 pcm/°F
Predicted Value for Conditions of Measurement	+1.7 pcm/°F
<u>Best Experimental Value</u>	
Best experimentally measured value	+2.1 pcm/°F

evaluation of the predicted magnitude of the coefficient is carried out and the corresponding measured value is presented for comparison. Again, the agreement is good; the positive bias of measurement compared to prediction persists. It should be noted that the uncertainty associated with the quoted measured value is relatively large and the significance of the comparison is therefore open to some question.

Greater confidence can be placed in the comparison of predicted and measured values of the differential power coefficient, due in large part to the relatively strong effect on core reactivity of changes in power level. The final predicted and best experiment differential worth functions of power level under typical nearly equilibrium conditions of power operation are compared in Figure 6.8. Apart from a small difference in shape, the predicted and measured functions are in close agreement over the entire range of power operation.

Figure 6.8 Comparison of Final Predicted and Best Experimental Power Coefficient Functions in Nearly Equilibrium Power Operating Conditions at Startup

VII. EVALUATION OF RESULTS AND CONCLUSIONS

The discussions thus far have progressed through review of analytical and experimental methods, computation of "best predicted" and "best measured" values of significant nuclear design parameters and, finally, in the preceding chapter, comparison of the results of analytical and empirical predictions with the results of corresponding measurements. In this concluding chapter remarks are made concerning the significance of the degree of agreement observed between prediction and measurement and conclusions are drawn as to the validity of the design methods involved.

1. Power Distributions and Hot Channel Factors

The data presented and the observations made in Section 6.1 show rather clearly that a discrepancy exists between predicted and measured core power distribution characteristics. No clear evidence has as yet been developed to show that the fault lies in measurement, although the possibility of systematic error in aeroball activation readout has not been completely eliminated. On the other hand, several differences in the conditions of prediction compared to those of measurement are known to exist. In particular, no provision was made in the analytical description supplied to the PDQ-03 code to account for the effects of Xe-135 absorption or for the spatial distribution of Doppler absorption and moderator density. Hence, initial post-startup analytical efforts were devoted to obtaining calculated core power distributions with spatially varying xenon concentration, Doppler absorption and moderator density explicitly included in the analytical core description. Preliminary results indicate that a large part, but not all, of the discrepancy evident in the comparison of "best predicted" and "best measured" power distribution characteristics in Chapter 6 is indeed to be found in the difference in conditions of prediction and measurement. A residual discrepancy, whereby predicted values are still high by about five percent in the central region of the core and correspondingly low in peripheral assemblies, persists. Current investigations are being directed toward the question of PDQ convergence on spatial flux distributions with coarse and fine meshes.

In contrast to the discrepancy between predicted and measured power distribution characteristics, the results summarized in Figures 6.3 and 6.4 indicate that the magnitudes of predicted and measured nuclear hot channel factors are in close agreement. The agreement is, in part, fortuitous in that the predicted radial location of the hottest channel is seen to be in error at the higher power levels. However, reference to Table 5.5 shows that the central region of the core is virtually flat in terms of the magnitudes of the radial hot channel factors in individual fuel assemblies. Specifically, the difference in magnitude between the hottest channel in the core and the hottest channel in assembly M-8 is no more than three percent regardless of power level or RCC control configuration. Accordingly, position of the hottest channel may be expected to be highly sensitive to differences between predicted and measured power distributions while magnitude of the hot channel factor would remain largely unaffected by change in power distribution. The implication is that, apart from the effects of the still persisting difference between predicted and measured power distribution characteristics discussed above, the analytical procedures for anticipating the magnitudes of radial and axial nuclear hot channel factors are valid.

2. RCC Group and Unit Worths

Review of Table 6.2 and Figures 6.5, 6.6 and 6.7 indicates that the basic nuclear design procedures which were used in predicting integral and differential RCC unit and group worths are indeed adequate for the purpose. Particularly close agreement between predicted and measured values was obtained in the hot zero power condition. In the relatively less important cold zero power condition, predicted values tended to be high by five to ten percent, while in typical power operating conditions, the predicted values proved low by perhaps ten percent over the region examined experimentally. The measured worths of individual RCC units in abnormal configurations were all seen to be lower than the predicted "upper limit" values assumed in the Safeguards Analysis.

It is suspected, but not yet demonstrated, that the difference between predicted and measured values of differential group worth under typical operating conditions may lie in the failure to treat explicitly in the analytical description of the core the axial variation in moderator density which exists during power operation. If so, the difference observed here may be symptomatic of the same disagreement between predicted and measured power distributions as was discussed above.

As an interesting aside, recent studies have shown that the observed perturbations in differential worth functions, attributed to the effects of local flux depression in the vicinity of the inconel fuel assembly grids, can indeed be reproduced analytically provided the neutron absorption properties of the grids are included explicitly in the analytical descriptions of the core.

3. Critical Boron Concentrations and Boron Worths

The use of an empirical bias factor to adjust analytically predicted critical boron concentrations to "best predicted" values appears to have been again appropriate in the design procedures in use just prior to San Onofre startup. Careful examination of Table 6.3 shows, however, that the bias factor should be dependent on RCC control group insertion and perhaps on moderator temperature as well.

Introduction of the empirical bias factor was not sufficient to give acceptable agreement between predicted and measured values of differential boron worth, as shown in Table 6.3. The sources of difficulty in predicting differential boron worth are several. A prime source of potential error lies in having to resort to empirical correction factors to account for a basic fault in the treatment of dissolved boron in the core moderator. Secondary sources of discrepancy can be found in exclusion of the effect of control rods on competitive absorption and on neutron leakage from the differential worth calculations.

The analytical fault in the treatment of dissolved absorbers has been the subject of continuing investigation, spurred in part by results reported

here. In recent months two improvements in analytical procedures have eliminated the need for an empirical bias factor. First, a special Dancoff correction has been introduced into the LEOPARD Code to account for the different self-shielding characteristics of non-asymptotic fuel cells, i.e., those cells which border on either an RCC water hole or a "between-assembly" water slot. Second, the PDQ-07 code which allows two-dimensional calculations with a comparatively large number of mesh points has become available. With the enhanced capability of PDQ-07, as compared to PDQ-03, a significantly more detailed description of the core in the x-y plane can be treated. In particular, the PDQ-07 core description makes use of nine mesh points in a 3x3 array to represent the boundary and center point of an equivalent RCC water hole while the corresponding PDQ-03 description allowed only a 2x2 array representing the corners of the equivalent water hole. Hence, PDQ-07 can treat explicitly thermal flux peaking in the water holes and in "between-assembly" water slots and thereby preserve the correct thermal absorption reaction rate in dissolved boron. The PDQ-03 approach is necessarily implicit (via the empirical bias factor). A comparison of predicted and measured values of the all RCC units out endpoint boron concentration and isothermal temperature coefficient, analogous to the comparisons made in Table 6.3 and Table 6.4 but based on direct analytical results derived from PDQ-07 calculations with the special Dancoff correction, is made in Table 7.1.

4. Isothermal Temperature Coefficients

The comparisons of predicted and measured values of isothermal temperature coefficients presented in Tables 6.4 and 6.5 indicate that analytical procedures in use prior to the startup of the San Onofre reactor can be expected to give valid predictions of the magnitude of isothermal coefficients provided the conditions of comparison are expressed in terms of just critical moderator absorber cross sections, not dissolved boron concentration, for specified control configurations. Relation of isothermal temperature coefficient magnitude to moderator boron concentration is prone to the basic analytical fault in treating boron which was discussed above.

TABLE 7.1

COMPARISON OF MEASURED VALUES OF A FEW SIGNIFICANT NUCLEAR DESIGN PARAMETERS
WITH ANALYTICAL VALUES OBTAINED BY RECENTLY IMPROVED CALCULATIONAL TECHNIQUES

Parameter	Cold Zero Power		Hot Zero Power	
	Predicted	Measured	Predicted	Measured
Endpoint Boron Concentration With All RCC Units Out	2265 ppm	2250 ppm	2532 ppm	2524 ppm
Isothermal Temperature Coefficient At All RCC Units Out Boron Concentration	+3.4 pcm/°F	+3.4 pcm/°F	+7.5 pcm/°F	+7.4 pcm/°F

Note: No empirical bias factors are applied.

Recently improved computational tools eliminate the analytical fault and allow comparison of predicted and measured values of the isothermal coefficient directly in terms of just critical boron concentrations in appropriate RCC group configurations as was shown in Table 7.1.

5. Differential Power ("Doppler") Coefficients

Justification of the analytical techniques used in predicting the differential power coefficient function in typical "near equilibrium" power operating conditions is readily found in the comparison of predicted and measured functions in Figure 6.8. In this case, the relatively good agreement between prediction and measurement is in part to be expected in that the analytical procedure involves the use of a semi-empirical correlation of experimental data derived from earlier reactor test programs. The results reported here further verify the validity of the correlation and increase the confidence that can be placed in future predictions.

6. Summary and Conclusions

The results of the comparison of predicted and measured nuclear design parameters at startup of the San Onofre reactor may be summarized as follows:

- a) Close agreement was obtained between predicted and measured values of:
 - 1) RCC group worth in the zero power condition
 - 2) Isothermal temperature coefficient under all conditions
 - 3) Power coefficient under all relevant conditions
 - 4) Nuclear hot channel factors in power operation conditions
 - 5) Power distributions in nominally zero power conditions.
- b) Less close, but entirely acceptable, agreement was obtained for:
 - 1) RCC group worths in typical power operation conditions
 - 2) RCC unit integral worths in abnormal configurations
 - 3) Endpoint boron concentrations at zero power
- c) In poorer agreement were:

- 1) Differential boron worths at zero power
- 2) Power distributions in typical power operation conditions.

Of the last set of parameters where good agreement was wanting, the analytical fault in the treatment of dissolved absorber (boron) worth has since been identified and resolved. Insight into the nature of the fault or faults in analytical prediction, and perhaps in experimental measurement as well, in evaluating core power distribution parameters has been gained; however, some degree of difference between predicted and measured power distributions persists.

The final conclusions to be drawn are accordingly:

- a) That nuclear design procedures in use at this writing are indeed adequate for, and suited to, the prediction of the values and properties of the parameters which characterize the static behavior of large PWR cores;
- b) That nuclear experimental test procedures and facilities have evolved to a point where meaningful data relating to significant nuclear design parameters can be obtained during the initial startup of a major nuclear power station;
- c) That a coordinated program of prediction, measurement and comparison of prediction with measurement can yield fruitful results in terms both of obtaining operationally useful data relating to the behavior of the particular reactor under study and of verification of analytical and experimental techniques.

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APPENDIX A

INITIAL APPROACH TO CRITICALITY

While the initial approach to criticality is of little significance in terms of experimental verification of analytical predicted core characteristics, it is of importance from the operational point of view. Accordingly, pertinent data are summarized here as a matter of record.

The basic scheme of the approach to criticality combined RCC group withdrawal and reduction of the reactor coolant boron content by dilution. The approach began with the reactor coolant in the hot, shutdown condition with all RCC units fully inserted and the reactor coolant borated to refueling concentration (nominally 2900 ppm, but in fact slightly over 3000 ppm).

It had been verified during previous operations in the startup program that criticality was not imminent with the two shutdown RCC groups withdrawn; hence, these groups were withdrawn without formal inverse count rate ratio (so-called $1/M$) monitoring. Since subsequent withdrawal of the Control Groups had not been attempted previously, the configuration with the Shutdown groups fully withdrawn, the Control Groups fully inserted and the reactor coolant boron concentration at 3008 parts per million was taken as the initial reference datum for subsequent operations. Formal inverse count rate ratio monitoring was carried out as the Control Groups were withdrawn in sequence. The results of the monitoring are shown in Figure A.1. The relatively large scatter in successive data points reflects the experimental uncertainty in determining the magnitudes of comparatively low count rates (less than 100 counts per second) by reading sensitive count rate meters.

At completion of RCC group withdrawal, Control Group 2 was inserted part way to obtain increased leverage on reactor control in terms of differential group worth, the inverse count rate ratio was renormalized to unity and dilution of the borated reactor coolant was begun. Figure A.2 shows the resultant change in inverse count rate ratio as functions of the volume of demineralized water injected into the reactor coolant system and

Figure A.1 Inverse Count Rate Ratio Monitoring During the RCC Withdrawal Phase of the Approach to Initial Criticality

Figure A.2 Inverse Count Rate Ratio Monitoring During the Boron Dilution Phase of the Approach to Initial Criticality

of the measured decrease in reactor coolant boron concentration. It is noteworthy that the initial response of both boron concentration and core multiplication, as indicated by count rate, to primary water injection was very slow. The delayed response is attributed to the fact that the initial volume of primary water, which registered as being injected to the reactor coolant system, served instead to displace borated water in the lines and headers of the chemical and volume control system. As the lines in service were flushed and demineralized water began to reach the reactor coolant system, both observed count rates and measured boron concentration showed anticipated behavior.

At a predetermined point, the dilution rate was reduced, again with a temporarily non-linear response in neutron count rate and measured boron concentration as a new system equilibrium state was established. When extrapolation of the inverse count rate ratio plot indicated that criticality was imminent Control Group 2 was withdrawn 15 steps, corresponding to a positive step of about 60 pcm in reactivity in an attempt to establish unequivocally a self-sustaining chain reaction. The first attempt by group withdrawal proved premature by a few minutes, the second led to a sustained asymptotic startup rate of about 0.5 decade per minute.