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Startup Test Activity Reduction Program

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Startup Test Activity Reduction Program

CEOG Task 1173

March 2003

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ABSTRACT

The purpose of this report is to propose changes to PWR reload startup testing that reduce testing activities in accordance with the following objectives:

1. To ensure the core can be operated as designed
2. To perform startup testing using normal plant operating practices
3. To reduce startup testing time

The proposed program provides an alternative means of ensuring the core can be operated as designed. The changes eliminate startup tests that require the use of the reactivity computer for core designs that are well characterized by experience. For a typical PWR this would eliminate the measurement of control rod worth and the isothermal temperature coefficient from low power physics testing. Tests and requirements are added that provide assurance the core can be operated as designed. The proposed program is referred to as the Startup Test Activity Reduction (STAR) program.

A Generic Program is identified that is representative of current PWR testing. Problems are defined in the report as core configurations that are not explicitly accounted for in the safety analysis. Problems that can be detected or initiated by startup testing are identified. An evaluation is performed to determine if the ability of the STAR Program to prevent operation with problems is essentially the same as, or better than, the Generic Program. The evaluation demonstrates the following:

- The uncertainties of parameters that are measured in the Generic Program, but not measured in the STAR Program, are bounded by the safety analysis when using the STAR Program.

- The ability of the STAR Program to prevent operation with as-built core problems is essentially the same as, or better than, the Generic Program.

- The STAR Program decreases the likelihood of operation outside the safety analysis due to test performance problems.

Based on the evaluation results, it is concluded that the STAR Program is acceptable for reload startup testing of PWRs.

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LIST OF ACRONYMS

AOA	Axial offset anomaly caused by the deposition of crud with boron on fuel
ANSI	American National Standards Institute
ARO	All rods out, i.e. unrodded operation
BE	Best estimate (used to describe predicted parameters that have been adjusted for measurement bias)
B-10	A boron isotope with a large neutron absorption cross section
BOC	Beginning of cycle operation
CASMO/SIMULATE	Core design method developed by Studsvik
CASMO/(PRISM or XTG)	Core design method used by Advanced Nuclear Fuel / Siemens Nuclear Power
CBC	Critical boron concentration
ΔCBC	Change in critical boron concentration
CE	Combustion Engineering
CEA	Control element assembly (CE terminology for control rod) or any PWR control rod (as used in this report)
CEOG	CE Owners Group
DIT/ROCS	Core design method developed by Combustion Engineering
ECP	Estimated critical position
EOC	End of cycle operation
HFP	Hot full power
HZP	Hot zero power
IASCC	Irradiated assisted stress corrosion cracking
IBW	Inverse boron worth
ITC	Isothermal temperature coefficient
LCO	Limiting condition for operation
LPPT	Low Power Physics Tests
MOC	Middle of cycle operation that includes most of the operating cycle
MTC	Moderator temperature coefficient
NSSS	Nuclear steam supply system
PHOENIX/ANC	Core design method developed by Westinghouse
PWR	Pressurized water reactor
QA	Quality assurance
RCCA	Rod Control Cluster Assembly (Westinghouse terminology for control rod)
RCS	Reactor coolant system
RE	Reactor Engineering
SDM	Shutdown margin
STAR	Startup Test Activity Reduction

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1.0 INTRODUCTION

1.1 PURPOSE

The purpose of this report is to propose changes to PWR reload startup testing that reduce testing activities in accordance with the following objectives:

1. To ensure the core can be operated as designed
2. To perform startup testing using normal plant operating practices
3. To reduce startup testing time

The changes eliminate startup tests that require the use of the reactivity computer for core designs that are well characterized by experience. Core designs that significantly depart from the experience base would require more extensive testing using a reactivity computer. For a typical PWR, the ITC and CEA worth measurements at HZP would be eliminated. The startup test changes also add tests and applicability requirements to ensure the core can be operated as designed. Technical Specification changes would be required to implement the changes to MTC surveillance tests. The proposed program is referred to as the Startup Test Activity Reduction (STAR) program.

The purpose of startup testing is to ensure the core can be operated as designed. This includes ensuring the operating characteristics of the core are consistent with design predictions. This report describes evaluations performed to determine the impact of the startup test program changes on the ability to achieve this purpose. In some instances, compensatory measures are added. These added requirements are in the form of both tests and STAR Applicability Requirements that must be satisfied to use the STAR Program.

Testing using normal plant operating practices can be accomplished by eliminating tests at HZP that require using the reactivity computer. Tests using the reactivity computer are the tests that typically require unique operating practices[1]. The elimination of these tests removes the need for unique plant configurations, some of which require special test exceptions to operate outside safety analysis assumptions. For example, CEA worth tests typically require CEA configurations outside safety analysis assumptions. In some instances plants need to bypass systems important to safety, such as an excore safety channel and associated trip functions, in order to provide the required test configuration. In addition, the frequent interaction between operations and test personnel on requirements for plant operating maneuvers to support testing with the reactivity computer would be avoided. Elimination of these unique operating practices decreases the potential for test performance errors that can result in operational problems. The potential for operational problems in current programs is judged to be small and is not by itself sufficient reason for eliminating tests. However, the elimination of potential operating problems does provide a benefit that should be considered in combination with other benefits. The performance of tests described in Table 1-1 and the use of the reactivity computer should remain as an option for obtaining information that may benefit plant operation.

A significant reduction in testing time can be realized by eliminating tests at HZP that require using the reactivity computer. Most of the lost generation associated with typical startup testing would be avoided. Required support from Operations, Chemistry, and Reactor Engineering would be reduced. In addition, some anomalous results associated with test performance errors would be eliminated.

When tests are eliminated from the startup test program, it is important that the startup test program continue to provide assurance the core can be operated as designed. The objective of the evaluations in this report is to demonstrate the STAR Program provides this assurance.

[1] The proposed program does not require that all unique operating practices be eliminated For example, the use of unique CEA configurations during the ITC test at power by some plants may continue.

1.2 BACKGROUND

Startup tests, including both low power physics tests and power ascension tests, are performed following refueling shutdowns or other significant reactor core alterations to ensure that the core can be operated as designed. This is accomplished by tests configured to determine if the operating characteristics of the core are consistent with core design methods and safety analysis assumptions. Table 1-1 provides a description of the tests discussed in this report. Table 1-2 provides the corresponding test purposes. The applicable sections of the 1997 ANSI standard for reload physics testing from Reference 1 are identified in these tables for tests that are included in the ANSI standard.

The purpose of startup testing is to ensure the core can be operated as designed by detecting design prediction and as-built core problems[2]. The basis for performing startup testing originated in the early years of nuclear power. The comparison of key measured physics parameters to design predictions was an important means of providing confidence that core design methods adequately predicted new core designs. The results of many years of startup testing have demonstrated good overall agreement between measurement and prediction. Furthermore, improvements in core design methods resulted in substantial increases in the accuracy of design predictions. In many instances the causes of recent anomalous test results have been problems with performing tests rather than problems with design predictions or the as-built core. The incidence of recent problems with either design predictions or the as-built core has been low.

The costs associated with startup tests are substantial. Lost generation is in the range of 8 to 24 EFPHs, which represents a cost in the range of $200,000 to $1,000,000. In addition, support from Operations, Chemistry, and Reactor Engineering, as well as analysis support, is required for this testing. Additional costs are occasionally incurred as a consequence of anomalous test results related to test performance problems. The high cost of startup testing, and the demonstrated reliability of design predictions for well established core designs using modern PWR methods such as DIT/ROCS, PHOENIX/ANC, and CASMO/SIMULATE, resulted in a review of startup testing by the CEOG. The objective of the review was to determine if changes to startup testing could be made that would reduce startup testing time while maintaining the ability to ensure the core can be operated as designed. In addition to reducing startup testing time, the incentive for this review was a desire to reduce operational problems by performing startup testing using normal plant operating practices. Normal operating practices are the use of plant equipment and processes that are characteristic of normal operation instead of unique operating practices used only during testing.

This report describes and evaluates the startup test program that resulted from the CEOG review. The evaluation assesses the impacts of the changes to current programs. These impacts involve the detection of design prediction and as-built core problems, and the initiation of test performance problems. Problems with the design predictions are related to the accuracy with which parameters important to the safety analysis are predicted. Problems with the as-built core are related to (a) errors in calculating parameters important to the safety analysis as well as (b) fuel and CEA fabrication errors, (c) core reassembly errors, and (d) fuel, CEA, or RCS abnormalities. Problems with test performance are related to errors associated with test equipment, processes, or results. This report evaluates the impact of the changes on each of these problems and develops conclusions on the acceptability of the STAR Program.

[2] In this report "problems" refer to core operating configurations that are not explicitly accounted for in the safety analysis. The word "problem" was selected to be consistent with the terminology used in Reference 1, the 1997 ANSI standard for reload physics testing.

Table 1-1 Startup Test Descriptions

TEST	POWER	ANSI[1]	DESCRIPTION
CEA Drop Time	Shutdown	NA	Determination of CEA drop time from measured trends of CEA position vs. time during CEA drops
CEA Drop Characteristics	Shutdown	NA	Verification of CEA coupling from analysis of measured rod drop test characteristics such as trends of drop time by location, slowing in the dashpot, and normal rebound
CEA Flux Change	HZP	NA	Verification of CEA coupling from measurements of reactivity or startup rate changes during CEA movement
CBC	HZP	6.2	Determination of CBC from chemical analysis of RCS samples
IBW	HZP	6.3	Determination of IBW from measurements of changes in reactivity and CBC
CEA Worth	HZP	6.4	Determination of CEA worth from measured change in reactivity during CEA motion
ITC	HZP	6.5	Determination of the ITC from measurements of changes in reactivity and moderator temperature when fuel and moderator temperature changes are isothermal
MTC Surveillance	HZP	NA	Determination of the MTC for various operating conditions from the measured ITC, the predicted Fuel Temperature Coefficient, and the predicted MTC
MTC Alternate Surveillance	HZP	NA	Determination of the MTC for various operating conditions by adjusting the predicted MTC using the measured CBC
SDM Surveillance	HZP	NA	Determination of the SDM using parameters measured as part of startup testing at HZP
CEA Flux Symmetry	HZP	6.6	Determination of the degree of azimuthal asymmetry in the neutron flux from measurements of the variation in CEA Worth from symmetric CEAs
Incore Flux Symmetry	Low	6.6	Determination of the degree of azimuthal asymmetry in the neutron flux from measurements of the variation in incore detector signals from symmetric incore detectors
Incore Power Distribution	Intermediate	6.7	Determination of the relative power distribution from the measurement of incore detector signals. Tests are typically performed at intermediate power levels in the 40-80% range.
ITC	Intermediate to HFP	NA	Determination of the ITC from measurements of changes in reactivity and moderator temperature when fuel and moderator temperature changes are isothermal
MTC Surveillance	Intermediate to HFP	NA	Determination of the MTC for various operating conditions from the measured ITC, the predicted Fuel Temperature Coefficient, and the predicted MTC
Incore Power Distribution	HFP	6.8	Determination of the relative power distribution from the measurement of incore detector signals. Tests are typically performed at power levels greater than 90%
ΔCBC HZP-HFP	HFP	6.9	Determination of the change in measured CBC between HZP and HFP from chemical analysis of RCS samples

[1] ANSI numbers refer to sections in the 1997 ANSI standard for reload startup physics tests, Reference 1, that describe the test. Tests designated "NA" are outside the scope of the ANSI standard.

Table 1-2 Startup Test Purposes

TEST	POWER	ANSI[1]	PURPOSE
CEA Drop Time	Shutdown	NA	To determine if CEA drop times are within Technical Specification limits and verify proper reassembly of the reactor vessel and internal components
CEA Drop Characteristics	Shutdown	NA	To determine if CEAs are coupled
CEA Flux Change	HZP	NA	To determine if CEAs are coupled
CBC	HZP	6.2	To Determine if the measured and predicted total core reactivity are consistent
IBW	HZP	6.3	To determine if the measured IBW is consistent with the predicted value
CEA Worth	HZP	6.4	To determine if the worth of selected rod groups is consistent with predictions
ITC	HZP	6.5	To determine if the measured ITC is consistent with the predicted value
MTC Surveillance	HZP	NA	To determine if the calculated MTC derived using the measured ITC is within Technical Specification limits
MTC Alternate Surveillance	HZP	NA	To determine if the calculated MTC derived using the measured CBC is within Technical Specification limits for various operating conditions
SDM Surveillance	HZP	NA	To determine if the calculated shutdown margin derived using measured test values is within Technical Specification limits
CEA Flux Symmetry	HZP	6.6	To determine if the measured azimuthal flux symmetry is consistent
Incore Flux Symmetry	Low	6.6	To determine if the measured azimuthal flux symmetry is consistent
Incore Power Distribution	Intermediate	6.7	To determine if the measured and predicted core power distributions are consistent
ITC	Intermediate to HFP	NA	To determine if the measured ITC is consistent with the predicted value
MTC Surveillance	Intermediate to HFP	NA	To determine if the calculated MTC derived using the measured ITC is within Technical Specification limits for various operating conditions
Incore Power Distribution	HFP	6.8	To determine if the measured and predicted core power distributions are consistent
ΔCBC HZP-HFP	HFP	6.9	To determine if the reactivity difference between zero and full power conditions is consistent with design predictions

[1] ANSI numbers refer to sections in the 1997 ANSI standard for reload startup physics tests, reference 1, that describe the test. Tests designated "NA" are outside the scope of the ANSI standard.

2.0 SUMMARY

A review of startup test programs at Participating Plants[3] is performed. A Generic Program is identified that is representative of current PWR testing. The STAR Program involves the following changes to the Generic Program:

- Elimination of the CEA Worth, ITC, and MTC Surveillance tests at HZP
- Addition of an MTC Alternate Surveillance test at HZP and an ITC and ΔCBC HZP-HFP test at power
- Addition of Applicability Requirements for core design, fuel fabrication, refueling, startup testing, and CEA lifetime

The Generic Program is considered to be a representative set of acceptable tests for PWRs. An evaluation is performed to determine if the impact of the changes to the Generic Program on safety analysis conformance is acceptable. Potential problems involving core configurations that are not explicitly accounted for in the safety analysis are identified for evaluation. Evaluation criteria are used that ensure the ability to prevent operation with these problems is not significantly degraded. Different evaluation criteria and processes are used for each of the following problem categories:

- Design prediction problems related to the accuracy of core design methods
- As-built core problems related to core anomalies or errors in core design, fabrication, or reassembly
- Test performance problems related to errors using test equipment, processes, or results

Problems are identified and evaluations are performed that demonstrate the following:

- An evaluation of four design prediction problems demonstrates that uncertainties in parameters that are measured in the Generic Program, but not measured in the STAR Program, are bounded by the safety analysis when using the STAR Program. This conclusion is supported by demonstrating the ability to benchmark a wide range of core designs and core design methods using a large database of recent startup test results.
- An evaluation of nineteen as-built core problems demonstrates that the ability of the STAR Program to prevent operation with problems is essentially the same as, or better than, the Generic Program. This conclusion is supported by a review of industry experience using NRC and INPO databases for past industry events.
- An evaluation of three test performance problems demonstrates that there is a decrease in the likelihood of operation outside the safety analysis as a result of eliminating unique operating practices associated with tests that require the reactivity computer.

In addition, an evaluation demonstrates there are no unique design features in the Participating Plants that require deviations from the STAR Program. Based on these evaluations, it was concluded that implementation of the STAR Program in the Participating Plants is acceptable. Further, it was concluded that implementation of the STAR Program in the non-participating PWRs is acceptable provided there are no unique design features that require additional startup testing.

[3] The Participating plants are the subset of CE Plants that are participants in CEOG Task 1173, Startup Test Time Reduction. These plants are ANO 2, Waterford 3, Millstone 2, SONGS 2 &3, Calvert Cliffs 1 & 2, St. Lucie 1 & 2 and Ft. Calhoun.

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3.0 REVIEW OF STARTUP TEST PROGRAMS

This section provides a review of startup test programs at Participating Plants. The results of this review are used to define a Generic Program that is representative of current PWR startup testing. The proposed STAR Program is presented and differences between the Generic Program and STAR Program are identified for subsequent evaluation. The following summarizes the startup test program changes for a typical Participating Plant:

- Elimination of the CEA Worth, ITC and MTC Surveillance tests at HZP
- Addition of an MTC Alternate Surveillance test at HZP and ITC and ΔCBC HZP-HFP tests at power
- Addition of Applicability Requirements

Sections 3.1 and 3.2 below describe the Generic and STAR Programs respectively, while Section 3.3 describes the changes between the two.

For the purpose of this report, startup tests are tests that involve measured parameters and have specific test criteria that are used to verify that the core can be operated as designed following reloads. Table 1-1 provides a description of the startup tests discussed in this report and Table 1-2 provides the corresponding test purposes. In some instances parameters may be measured but this does not constitute a test of the parameter if the measurement is not compared to test criteria. For example, the ITC is typically measured at both HZP and HFP but may only be compared to test criteria at HZP. In this instance there is only an ITC test at HZP. However the ITC measurements at HZP and HFP are typically used to calculate a MTC at both HZP and HFP that is compared to test criteria derived from Technical Specifications. In this instance there is a MTC Surveillance test at both HZP and HFP.

3.1 CURRENT STARTUP TEST PROGRAMS

This section provides the results of a review of current startup testing at Participating Plants. This information is used to identify a Generic Program that is representative of current PWR startup testing.

3.1.1 Review of Startup Test Programs for Participating Plants

A review of the current design and licensing bases for startup tests was conducted for Participating Plants. The Participating Plants were requested to identify specific documents and corresponding sections that contain design and licensing bases related to startup tests. These documents were then reviewed to determine the design and licensing bases relating to startup tests for Participating Plants. The results of this review are summarized in Table 3-1 by Licensee. A recommended program from the ANSI standard for startup testing, Reference 1, is also included in Table 3-1 for comparison.

3.1.2 Generic Program

A Generic Program was determined by selecting a subset of tests that are typically performed by the Participating Plants. The following list summarizes the tests that comprise the Generic Program:

TEST	POWER	RODS
CEA Drop Time	Shutdown	Moved
CEA Drop Characteristics	Shutdown	Moved
CBC	HZP	ARO
CEA Worth	HZP	Rodded
ITC	HZP	ARO
MTC Surveillance	HZP	ARO
Incore Flux Symmetry	Low	ARO
Incore Power Distribution	Intermediate	ARO
MTC Surveillance	Intermediate to HFP	ARO
Incore Power Distribution	HFP	ARO

Table 3-2 provides descriptions of the tests in the Generic Program. Table 3-1 compares the Generic Program to the programs for individual licensees and a set of tests recommended by ANSI. In addition to the detailed reviews performed on a representative set of plants, knowledge of testing requirements for CE Plants by individuals experienced in startup testing and a review of all CE UFSARs for startup test requirements provide confidence in the Generic Program selections.

Although all the plants reviewed perform a SDM surveillance[4] at HZP following a refueling, no standard requirements for the use of startup test measurements in the surveillance was found, and the licensees employed different practices as illustrated in the following summary:

- Two plants did not use startup test measurements
- One plant used measured CBC
- Three plants required an acceptable CEA worth measurement as a prerequisite
- Two plants adjusted the predicted CEA worth if the measured value is low by 10% or more.
- Two plants adjusted the predicted CEA worth using the measured CEA worth.

Based on the above, it was concluded that there was no generic practice regarding the use of startup test measurements in shutdown margin surveillances. Furthermore, the requirement for a SDM surveillance prior to exceeding 5% power after fuel loading has been removed from the CE Standard Technical Specifications, Reference 3, and the remaining SDM surveillances require the verification of CBC in shutdown conditions. Because there was no generic practice or general requirement for SDM surveillances at HZP, the SDM surveillance test was not included as part of the Generic Program.

Although the Generic Program is based on a review of Participating Plants, it is generally applicable to all PWRs. This is because the startup testing requirements and practices at Westinghouse and B&W Plants are similar to CE Plants. Furthermore, the similarity of all PWR designs suggests that a startup program appropriate for any subset would be appropriate for other PWRs. This is supported by a single ANSI standard for startup testing that is applicable to all PWRs. Exceptions would be PWRs with unique design features that could cause significant increases in problems that could be detected by additional startup testing.

[4] SDM surveillances that do not use startup test measurements are not considered startup tests.

3.2 STAR PROGRAM

This section describes a reload startup test program for PWRs configured to accomplish the following objectives:

1. To ensure the core can be operated as designed
2. To perform startup testing using normal plant operating practices
3. To reduce startup testing time

The proposed program is referred to as the Startup Test Activity Reduction (STAR) program.

3.2.1 STAR Program Tests

The STAR Program eliminates tests that require the use of the reactivity computer. Tests are added to ensure that the core can be operated as designed. Table 3-3 provides a description of the tests required by the STAR Program[5]. For a typical PWR the STAR Program would involve elimination of the ITC and CEA worth measurements at HZP. This removes the need for unique operating practices while significantly reducing the testing time. Unique plant operating practices are the use of plant equipment and processes that are not characteristic of normal operation. Startup testing using only normal operating practices reduces the potential for operational problems associated with the testing. The following list summarizes the tests that comprise the STAR startup program:

TEST	POWER	RODS
CEA Drop Time	Shutdown	Moved
CEA Drop Characteristics	Shutdown	Moved
CBC	HZP	ARO
MTC Alternate Surveillance[6]	HZP	ARO
Incore Flux Symmetry	Low	ARO
Incore Power Distribution	Intermediate	ARO
ITC	Intermediate to HFP	ARO
MTC Surveillance	Intermediate to HFP	ARO
Incore Power Distribution	HFP	ARO
ΔCBC HZP-HFP	HFP	ARO

The MTC Alternate Surveillance test at HZP is a new test that is added by the program. It replaces the MTC Surveillance test at HZP with a test that does not require the use of the reactivity computer. Instead of calculating the MTC from the measured ITC, the MTC is calculated from the measured CBC. The predicted MTC is adjusted for the measured CBC using the predicted relationship between MTC and CBC and compared with Technical Specification limits.

3.2.2 STAR Program Applicability Requirements

The STAR Applicability Requirements are conditions that must be satisfied to use the STAR Program. The STAR Applicability Requirements are provided in Table 3-4 and provide compensatory measures that ensure the core can be operated as designed when used in conjunction with the proposed tests. The STAR Applicability Requirements involve the following areas:

- Core Design
- Fuel Fabrication

[5] This is the minimum acceptable set of tests using the STAR program. Additional tests may be performed when using the STAR program including tests that require the use of the reactivity computer.
[6] Not required if an MTC Surveillance test is performed at HZP.

- Refueling
- Startup Testing
- CEA Lifetime

3.3 CHANGES TO GENERIC PROGRAM

The changes to the Generic Program tests are identified in Table 3-5 so the impact of the STAR Program can be evaluated. Table 3-5 also provides a description of these tests involved in the change. The following list summarizes the changes to the Generic Program tests:

TEST	POWER	RODS	CHANGE
CEA Worth	HZP	Rodded	Eliminated
ITC	HZP	ARO	Eliminated
MTC Surveillance	HZP	ARO	Eliminated
MTC Alternate Surveillance	HZP	ARO	Added
ITC	Intermediate to HFP	ARO	Added [7]
ΔCBC HZP-HFP	HFP	ARO	Added

These test changes, along with the STAR Applicability Requirements in Table 3-4, comprise the changes to the Generic Program. The changes to the Generic Program eliminate the use of the reactivity computer and the requirement for unique operating practices. Two measurements, CEA worth and ITC, are eliminated at HZP, which result in the elimination of the CEA Worth, ITC, and MTC tests at HZP. The ITC measurement is still performed at power and thus remains as part of the overall startup test program. The STAR Program adds a CBC measurement at HFP in order to perform the ΔCBC HZP-HFP test at HFP.

The measurements are eliminated only when they are no longer required to benchmark core design methods. The benchmarking of core design methods is the determination of the uncertainties associated with the methods based on deviations between measured and predicted values for parameters. These uncertainties are applicable for a range of core designs similar to those involved in the benchmarking process. Subsequent measurements for core designs outside this range can be used to extend the range of applicable core designs by demonstrating consistency with previously established uncertainties. Once sufficient data has been obtained to establish the accuracy of core design methods for a range of core designs it is no longer necessary to perform measurements for this range of core designs. Although these measurements are no longer required to benchmark core designs, they may be useful in detecting as-built core problems. The following provides the compensatory measures in the STAR Program that ensure the core can be operated as designed with these measurements eliminated:

- The added MTC Alternate Surveillance at HZP provides a method of detecting MTC noncompliance with Technical Specification requirements at HZP without the reactivity computer. This replaces the MTC Surveillance test at HZP and provides an alternate means of MTC surveillance until the MTC Surveillance test is performed at power.

- The added ITC test at power replaces the ITC test at HZP and thus provides continued verification of ITC predictions by core design methods.

[7] This does not add a measurement to the startup program because the ITC measurement is already performed for the MTC Surveillance at intermediate to HFP. However, this does add a test because the measured ITC is compared to test criteria in the STAR Program.

- The added ΔCBC HZP-HFP test at HFP provides additional verification of core design methods and problem detection capability without any requirement for plant maneuvers.

The MTC surveillance changes in the STAR Program may require Technical Specification changes. The following changes to typical Technical Specification requirements for CE Plants are necessary to implement the STAR Program:

- For operating cycles that meet the STAR Applicability Requirements, the MTC that is required prior to entering MODE 1 may be obtained from either the measured isothermal temperature coefficient or the predicted MTC adjusted for the measured CBC.
- If the MTC that is required prior to entering MODE 1 was obtained from the predicted MTC adjusted for the measured CBC, then the MTC obtained at BOC in MODE 1 must be verified to be within the upper limit.
- For CE Plants that have eliminated the 2/3 cycle MTC surveillance in accordance with CEOG Task 1009 and Reference 2, the determination whether the 2/3 cycle MTC surveillance is required is based on the MTC obtained at BOC in MODE 1. Currently, both the BOC MTC obtained prior to MODE 1 and in MODE 1 are used to make this determination.

Table 3-1 Startup Test Programs for Participating Plants

TEST[1]	POWER	RODS	PARTICIPATING PLANT									
			ANO 2	Waterford 3	Calvert Cliffs 1 & 2	Millstone 2	SONGS 2 & 3	St Lucie 1	St Lucie 2	Ft Calhoun	Generic Program	1997 ANSI Standard
CEA Drop Time	Shutdown	Moved	X	X	X	X	X	X	X	X	X	
CEA Drop Characteristics	Shutdown	Moved	X		X	X	X	X	X		X	
CEA Flux Change	HZP	Moved	X	X			X					
CBC	HZP	ARO	X	X	X	X	X	X	X	X	X	X
CBC	HZP	Rodded	X		X	X						
IBW	HZP	Rodded						X	X			X
CEA Worth	HZP	Moved	X	X	X	X	X	X	X	X	X	X
ITC	HZP	ARO	X	X	X	X	X	X	X	X	X	X
MTC Surveillance	HZP	ARO	X	X	X	X	X	X	X	X	X	
MTC Alternate Surveillance	HZP	ARO										
SDM Surveillance	HZP	ARO	X	X		X	X	X	X	X		
CEA Flux Symmetry	HZP	Moved			X							
Incore Flux Symmetry	Low	ARO	X	X	X	X	X	X	X	X	X	X[2]
Incore Power Distribution	Intermediate	ARO	X	X	X	X	X	X	X	X	X	X
ITC	Intermediate to HFP	ARO	X	X		X		X	X			
MTC Surveillance	Intermediate to HFP	ARO	X	X	X	X		X	X	X	X	
Incore Power Distribution	HFP	ARO	X	X	X	X	X	X	X	X	X	X
ΔCBC HZP-HFP	HFP	ARO										X

[1] Table 1-1 provides descriptions, and Table 1-2 provides the purposes, of the tests discussed in this report.
[2] The CEA Flux Symmetry test is an alternate to the Incore Flux Symmetry test in the ANSI Standard.

Table 3-2 Generic Program Tests

TEST[1]	POWER	DESCRIPTION
CEA Drop Time	Shutdown	Determination of CEA drop time from measured trends of CEA position vs. time during CEA drops
CEA Drop Characteristics	Shutdown	Verification of CEA coupling from analysis of measured rod drop test characteristics such as trends of drop time by location, slowing in the dashpot, and normal rebound
CBC	HZP	Determination of CBC from chemical analysis
CEA Worth	HZP	Determination of CEA worth from measured change in reactivity during CEA motion
ITC	HZP	Determination of the ITC from measurements of changes in reactivity and moderator temperature when fuel and moderator temperature changes are isothermal
MTC Surveillance	HZP	Determination of the MTC for various operating conditions from the measured ITC, the predicted Fuel Temperature Coefficient, and the predicted MTC
Incore Flux Symmetry	Low	Determination of the degree of azimuthal asymmetry in the neutron flux from measurements of the variation in incore detector signals from symmetric incore detectors
Incore Power Distribution	Intermediate	Determination of the relative power distribution from the measurement of incore detector signals. Tests are typically performed at intermediate power levels in the 40-80% range.
MTC Surveillance	Intermediate to HFP	Determination of the MTC for various operating conditions from the measured ITC, the predicted Fuel Temperature Coefficient, and the predicted MTC
Incore Power Distribution	HFP	Determination of the relative power distribution from the measurement of incore detector signals. Tests are typically performed at power levels greater than 90%

[1] Table 1-2 provides the purposes of the tests discussed in this report.

Table 3-3 STAR Program Tests

TEST[1]	POWER	DESCRIPTION
CEA Drop Time	Shutdown	Determination of CEA drop time from measured trends of CEA position vs. time during CEA drops
CEA Drop Characteristics	Shutdown	Verification of CEA coupling from analysis of measured rod drop test characteristics such as trends of drop time by location, slowing in the dashpot, and normal rebound
CBC	HZP	Determination of CBC from chemical analysis
MTC Alternate Surveillance[2]	HZP	Determination of the MTC for various operating conditions by adjusting the predicted MTC for various operating conditions using the measured CBC
Incore Flux Symmetry	Low	Determination of the degree of azimuthal asymmetry in the neutron flux from measurements of the variation in incore detector signals from symmetric incore detectors
Incore Power Distribution	Intermediate	Determination of the relative power distribution from the measurement of incore detector signals. Tests are typically performed at intermediate power levels in the 40-80% range.
ITC	Intermediate to HFP	Determination of the ITC from measurements of changes in reactivity and moderator temperature when fuel and moderator temperature changes are isothermal
MTC Surveillance	Intermediate to HFP	Determination of the MTC for various operating conditions from the measured ITC, the predicted Fuel Temperature Coefficient, and the predicted MTC
Incore Power Distribution	HFP	Determination of the relative power distribution from the measurement of incore detector signals. Tests are typically performed at power levels greater than 90%
ΔCBC HZP-HFP	HFP	Determination of the change in CBC between HZP and HFP from chemical analysis

[1] Table 1-2 provides purposes of the tests discussed in this report
[2] Not required if a MTC Surveillance test is performed at HZP.

Table 3-4 STAR Program Applicability Requirements

Table 3-4 STAR Program Applicability Requirements

[1] Not required if an MTC Surveillance test is performed at HZP.

Table 3-4 STAR Program Applicability Requirements

Table 3-5 Changes to Generic Program Tests

TEST[1]	POWER	RODS	CHANGES	DESCRIPTION
CEA Worth	HZP	Moved	Eliminated	Determination of CEA worth from measured change in reactivity during CEA motion
ITC	HZP	ARO	Eliminated	Determination of the ITC from measurements of changes in reactivity and moderator temperature when fuel and moderator temperature changes are isothermal
MTC Surveillance	HZP	ARO	Eliminated	Determination of the MTC for various operating conditions from the measured ITC, the predicted Fuel Temperature Coefficient, and the predicted MTC
MTC Alternate Surveillance	HZP	ARO	Added	Determination of the MTC for various operating conditions by adjusting the predicted MTC for various operating conditions using the measured CBC
ITC	Intermediate to HFP	ARO	Added	Determination of the ITC from measurements of changes in reactivity and moderator temperature when fuel and moderator temperature changes are isothermal
ΔCBC HZP-HFP	HFP	ARO	Added	Determination of the change in CBC between HZP and HFP from chemical analysis

[1] Table 1-2 provides the purposes of the tests discussed in this report

4.0 EVALUATION OF IMPACT OF CHANGES TO GENERIC PROGRAM

This section provides an evaluation of the STAR Program for PWRs. The STAR Program consists of startup tests listed in Table 3-3 selected to determine if the core can be operated as designed provided the STAR Applicability Requirements in Table 3-4 are satisfied. The changes to be evaluated include both the changes to the tests and the added STAR Applicability Requirements. The approach is to assess the impact of the changes to the Generic Program identified in Section 3.3. The Generic Program is considered to be a representative set of acceptable tests.

The impact of changes to the Generic Program is considered acceptable if there is no significant adverse impact on safety analysis conformance. In this evaluation core configurations that are not explicitly accounted for in the safety analysis are referred to as "problems." The word "problem" was selected to be consistent with the terminology used in the 1997 ANSI standard for reload physics testing, Reference 1. Startup tests can both detect and initiate problems. The evaluation consists of determining if the change in the ability to prevent problems is acceptable. The impact of the change on each problem is evaluated separately. This was found to be desirable because each problem identified for evaluation has many unique aspects that need to be considered in conjunction with all the changes to the Generic Program. The problems are divided into the following three general categories:

- Design Prediction problems related to the accuracy of core design methods
- As-Built Core problems related to core anomalies or errors in core design, fabrication, or reassembly
- Test Performance problems related to errors using test equipment, processes, or results

The detection of design prediction and as-built core problems by startup tests can impact safety analysis conformance through corrective actions that ensure operation within the safety analysis. The initiation of problems during the performance of startup tests can cause operation outside the safety analysis.

Four design prediction problems are identified for evaluation. The design prediction problems are based on the parameters measured in the Generic Program. CEA worth, CBC, ITC, and Power Distribution are the parameters measured in the Generic Program and thus inaccuracies in these parameters constitute the design prediction problems. Nineteen as-built core problems are identified for evaluation in Appendix C. Appendix C addresses the detection of as-built core problems and identifies as-built core problems that are based in part on a problem identification matrix from the 1997 ANSI standard for reload physics testing, Reference 1. Three test performance problems are identified for evaluation in Appendix D. Appendix D addresses the initiation of test performance problems associated with errors using test equipment, processes, or results.

The impact of the STAR Program is considered acceptable if there is no significant adverse impact on safety analysis conformance. [

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A specific evaluation criterion is developed for each problem category that ensures the general evaluation criterion above is satisfied. The impact of the changes on design prediction, as-built core, and test performance problems are addressed in Sections 4.1, 4.2, and 4.3 respectively. The section for each problem category contains a subsection that describes the (a) criterion used in the evaluation, (b) information required for the evaluation and (c) process used to perform the evaluation. Following each description there is a section that provides an individual evaluation for each problem.

The impact of current deviations in startup testing by Participating Plants from the Generic Program is evaluated in Appendix F. In all cases it was determined that eliminating the deviations and using only the tests in the STAR Program is acceptable for Participating Plants. In addition, it is concluded that a CEA Flux Change test based on either measured reactivity changes or startup rates is an acceptable alternative to the CEA Drop Characteristics test for detecting CEA uncoupling.

4.1 IMPACT OF CHANGES ON DESIGN PREDICTION PROBLEMS

This section evaluates the impact of the changes to the Generic Program on design prediction problems. In this evaluation, design prediction problems are deviations between the predictions from core design methods and the operating characteristics of the core, and reflect the accuracy of the predictions[8]. The selection of design prediction problems for evaluation was based on the parameters that are measured for comparison to predictions in the Generic Program. Table 1-2 provides the startup test purposes from which this information was obtained. The following are the design prediction problems that are identified for evaluation:

- CEA Worth Inaccuracy
- CBC Inaccuracy
- ITC Inaccuracy
- Power Distribution Inaccuracy

4.1.1 Design Prediction Problem Evaluation Description

This section describes the following:

- Criterion used in the evaluation
- Information required for the evaluation
- Process used to perform the evaluation

4.1.1.1 Design Prediction Problem Evaluation Criterion

The design prediction criterion is based on ensuring that the core design methods provide predicted parameters that are conservative when used in the safety analysis. The following evaluation criterion is used to ensure the general evaluation criterion in Section 4.0 is satisfied:

- [

]

[8] Errors in the application of the core design methods are not addressed in this section but are addressed in Section 4 2 on as-built core problems. Section 4 1 addresses only the accuracy of the core design methods

4.1.1.2 Design Prediction Problem Evaluation Information

In order to determine if the evaluation criterion is satisfied it is necessary to determine if the uncertainty remains bounded by the safety analysis assumptions when using the STAR Program. The following information is required for this evaluation:

Appendix B provides a review of startup tests used as the source of information for the analysis of deviations between measurements and predictions. Best estimate predictions are the predictions from core design methods corrected for the bias between past measurements and predictions. A large database was used to characterize the differences between measurements and best estimate predictions for CEA worth and ITC in Appendix B. Included are measurement results for CEA worth and ITC from multiple cycles for Participating Plants as well as some nonparticipating CE Plants. This data consists of deviations between measurements and best estimate predictions and covers a wide range of core designs that include significant variations in fuel management, fuel enrichment, poison type, poison loading, and exposure. The data also reflect changes that have occurred as core designs have evolved with time. In addition, the data include a range of modern PWR core design methods including DIT/ROCS, PHOENIX/ANC, and CASMO/SIMULATE.

Table B-10 summarizes the results of analyses in Appendix B that examined the data described above. The data was demonstrated to be consistent with a normal distribution with a mean near zero. The results in Appendix B provide information related to the following:

[

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[

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4.1.1.3 Design Prediction Problem Evaluation Process

Figure 4-1 presents a flowchart of the evaluation process for design prediction problems. The process evaluates the impact of the changes on the design prediction problems identified in Section 4.1 by determining if the change is acceptable using the design prediction criterion. The evaluation process for design prediction problems consists of the following three steps:

4.1.2 Design Prediction Problem Evaluation

This section provides an individual evaluation for each of the four design prediction problems potentially impacted by startup testing. The corresponding evaluations using the process described in Section 4.1.1 are provided in Appendix E. Significant results from these evaluations are summarized below. A summary of the impacts on the ability to ensure uncertainties are bounded by the safety analysis is provided in Table 4-1. A summary of all the impacts associated with the changes to the Generic Program is provided in Table 5-1.

4.1.2.1 Impact on CEA Worth Inaccuracy

CEA worth inaccuracy is the deviation between the CEA worth predicted by core design methods and the CEA worth actually present in the core. CEA worth inaccuracy is characterized by an uncertainty that is based on deviations between core design predictions and startup test measurements at HZP. Typically best estimate predictions are used in which the predicted result from the core design method is adjusted for the bias between past measurements and predictions. Appendix B presents the results of a review of past startup tests. Based on this review of a large number of recent CEA Worth test results, the biases and uncertainties from previous benchmarking continue to be applicable. A CEA worth inaccuracy problem is CEA worth that is not bounded by the safety analysis. Appendix A provides the results of a review of industry problems and Table A-7 summarizes the results for design prediction problems. The review identified three instances of CEA worth inaccuracy problems. One instance involved a failure to account for the decay of Pu in irradiated fuel that had been discharged for four cycles. Accounting for decay in fuel has subsequently been incorporated in modern core design methods. This problem is detectable by the Incore Power Distribution tests that are retained in the STAR Program. One instance involved a core design method that did not have an appropriate bias for the core being designed and one instance involved a change in fuel management to a low leakage core design that had not been benchmarked. The STAR Program Applicability Requirements require that benchmarking be performed and that the tests are eliminated only for core designs that are similar to those used to benchmark the predictions.

The impact of the change on CEA worth inaccuracy is determined to be acceptable based on the evaluation performed in Section E.2.1.1 of Appendix E. [

]

4.1.2.2 Impact on CBC Inaccuracy

CBC inaccuracy is the deviation between the CBC predicted by core design methods and the CBC actually present in the core. CBC inaccuracy is characterized by an uncertainty that is based on deviations between core design predictions and startup test measurements. Typically best estimate predictions are used that adjust the predicted result from the core design method for the bias between past measurements and predictions.
A CBC inaccuracy problem is a CBC that is not bounded by the safety analysis. Appendix A provides the results of a review of industry problems and Table A-7 summarizes the results for design prediction problems. The review identified one instance of a CBC inaccuracy problem that was detected by a CBC surveillance during MOC operation. The CBC inaccuracy was not detected by the startup CBC test at HZP because the effect of the problem on CBC was within the test criteria at BOC. The problem was detected by a surveillance later in the cycle when the effect was larger. The STAR Program does not impact any CBC surveillances.

The impact of the change on CBC inaccuracy is determined to be acceptable based on the evaluation performed in Section E.2.1.2 of Appendix E. [

]

[9] The startup test data analyzed consist of deviations between measured and predicted CEA worth.

[

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4.1.2.3 Impact on ITC Inaccuracy

ITC inaccuracy is the deviation between the ITC predicted by core design methods and the ITC actually present in the core. ITC inaccuracy is characterized by an uncertainty that is based on deviations between core design predictions and startup test measurements. Typically best estimate predictions are used that adjust the predicted result from the core design method for the bias between past measurements and predictions. Appendix B presents the results of a review of past startup tests. Based on this review of a large number of recent ITC test results, the bias and uncertainty from previous benchmarking continue to be applicable. An ITC inaccuracy problem is an ITC that is not bounded by the safety analysis. Appendix A provides the results of a review of industry problems and Table A-7 summarizes the results for design prediction problems. The review did not identify any instances of ITC inaccuracy problems. Although instances of measurements in excess of MTC Technical Specification limits were identified in Section 4.2.2.5, the review did not identify any instances of ITC inaccuracy problems in the prediction of MTC or ITC.

The impact of the change on ITC inaccuracy is determined to be acceptable based on the evaluation performed in Section E.2.1.3 of Appendix E. [

]

4.1.2.4 Impact on Power Distribution Inaccuracy

Power distribution inaccuracy is the deviation between the power distribution predicted by core design methods and the power distribution actually present in the core. Power distribution inaccuracy is characterized by an uncertainty that is based on deviations between core design predictions and startup test measurements at power. A power distribution inaccuracy problem is a power distribution that is not bounded by the safety analysis. Appendix A provides the results of a review of industry problems and Table A-7 summarizes the results for design prediction problems. The review did not identify any instances of power distribution inaccuracy problems.

The impact of the change on power distribution inaccuracy is determined to be acceptable based on the evaluation performed in Section E.2.1.4 of Appendix E. [

]

4.2 IMPACT OF CHANGES ON AS-BUILT CORE PROBLEMS

This section evaluates the impact of the changes to the Generic Program on as-built core problems. In this evaluation, as-built core problems are deviations from the intended core design. As-built core problems are a result of either errors in the core design process or physical characteristics of the core that differ from the core design. The identification of as-built core problems is in part based on the kinds of problems and their symptoms that have been identified in the past and documented by ANSI, and in part on a review of industry problems coupled with engineering judgment. The following are the as-built core problems that are identified for evaluation:

- CEA Worth Error
- CBC Error
- ITC Error
- Power Distribution Error
- MTC Noncompliance
- SDM Noncompliance
- Fuel Fabrication Error
- Fuel Misloading
- Fuel Distortion
- Fuel Poison Loss
- Fuel Crudding
- CEA Fabrication Error
- CEA Misloading
- CEA Uncoupling
- CEA Distortion
- CEA Absorber Loss
- CEA Finger Loss
- RCS Anomaly
- RCS B-10 Depletion

4.2.1 Description of As-Built Core Problem Evaluation

This section describes the following elements of the as-built core problem evaluation:

- Criterion used in the evaluation
- Information required for the evaluation
- Process used to perform the evaluation

4.2.1.1 As-Built Core Problem Evaluation Criterion

The as-built core criterion is based on ensuring that increases in as-built core problems are small. Additional as-built core problems could result from changes in the effectiveness of detecting as-built core problems in the STAR Program. The following evaluation criterion is used in the evaluation of as-built core problems to ensure the general evaluation criterion in Section 4.0 is satisfied:

- [

]

4.2.1.2 As-Built Core Problem Evaluation Information

In order to determine if the evaluation criterion is satisfied it is necessary to determine the effectiveness of the various methods of detecting as-built core problems. The following detection related information is required for this evaluation:

- Effectiveness of startup tests in detecting as-built core problems
- Effectiveness of pre-operational activities in detecting as-built core problems
- Effectiveness of STAR Applicability Requirements in detecting as-built core problems

Appendix C addresses the detection of as-built core problems. Tables C-5, C-6, and C-7 provide the effectiveness in detecting as-built core problems for startup tests, pre-operational activities, and STAR Applicability Requirements respectively. These tables use a three level rating system to represent the effectiveness of the various methods in detecting as-built core problems. Detection methods are rated as "Good," "Fair," or "Poor. This information is used in the evaluations to determine changes in the ability to detect as-built core problems between the Generic and STAR Programs.

4.2.1.3 As-Built Core Problem Evaluation Process

Figure 4-2 presents a flowchart of the evaluation process for as-built core problems. The process evaluates the impact of the changes on the as-built core problems identified in Section 4.2 by determining if the change is acceptable using the as-built core criterion. The evaluation process for as-built core problems consists of the following four steps:

[

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[

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4.2.2 As-Built Core Problem Evaluation

This section provides an individual evaluation for each of the nineteen as-built core problems potentially impacted by startup testing. The corresponding evaluations using the process described in Section 4.2.1 are provided in Appendix E. Significant results from these evaluations are summarized below. A summary of the impacts on the overall effectiveness in detecting as-built core problems is provided in Table 4-2. A summary of all the impacts associated with the changes to the Generic Program is provided in Table 5-1.

4.2.2.1 Impact on CEA Worth Error Detection

CEA worth error detection is the detection of CEA worth predictions that result from errors in the application of PWR core design methods. Appendix A provides the results of a review of industry problems and Table A-8 summarizes the results for as-built core problems. The review did not identify any instances of CEA worth errors in the PWR design process.

The impact of the change on the detection of CEA worth errors is determined to be acceptable based on the evaluation performed in Section E.2.2.1 of Appendix E. [

]

4.2.2.2 Impact on CBC Error Detection

CBC error detection is the detection of CBC predictions that result from errors in the application of PWR core design methods. Appendix A provides the results of a review of industry problems and Table A-8 summarizes the results for as-built core problems. The review did not identify any instances of a CBC error in the PWR design process.

The impact of the change on the detection of CBC errors is determined to be acceptable based on the evaluation performed in Section E.2.2.2 of Appendix E. [

]

4.2.2.3 Impact on ITC Error Detection

ITC error detection is the detection of ITC predictions that result from errors in the application of PWR core design methods. Appendix A provides the results of a review of industry problems and Table A-8 summarizes the results for as-built core problems. The review did not identify any instances of ITC errors in the PWR design process. Although instances of measurements in excess of MTC Technical Specification limits were identified in Section 4.2.2.5, no errors in the prediction of MTC or ITC were identified.

The impact of the change on the detection of ITC errors is determined to be acceptable based on the evaluation performed in Section E.2.2.3 of Appendix E. [ ]

[

]

4.2.2.4 Impact on Power Distribution Error Detection

Power distribution error detection is the detection of power distribution predictions that result from errors in the application of PWR core design methods. Appendix A provides the results of a review of industry problems and Table A-8 summarizes the results for as-built core problems. The review identified one instance of a power distribution error that was detected by the Incore Flux Symmetry test at power. This test is included in the STAR Program.

The impact of the change on the detection of power distribution errors is determined to be acceptable based on the evaluation performed in Section E.2.2.4 of Appendix E. [

]

4.2.2.5 Impact on MTC Noncompliance Detection

MTC noncompliance detection is the detection of MTC values that are outside Technical Specification limits. MTC surveillances are required at HZP and power in the Generic Program. Appendix A provides the results of a review of industry problems and Table A-8 summarizes the results for as-built core problems. The review identified thirty instances of MTC measurements at HZP in excess of Technical Specification limits. Typically operation with a MTC in excess of the MTC limit during HZP startup testing is not a violation of Technical Specifications because of special test exceptions. In these instances appropriate corrective actions were implemented and no instances of Technical Specification violations were identified. The review of these potential problems did not identify any instances of inaccuracies or errors in the prediction of MTC or ITC and the differences between predicted and measured values were consistent with typical MTC uncertainties. These instances, instead, appear to be the result of core designs with predicted MTCs that were close to the Technical Specification limits. These potential problems were detected by the MTC Surveillance test at HZP. An MTC Alternate Surveillance test at HZP is used in the STAR Program. [

] The STAR Program Core Design Applicability Requirements ensure that the tests are eliminated only for core designs that are similar to those used to benchmark the predictions. The MTC Alternate Surveillance test at HZP used in the STAR Program adjusts the predicted MTC at HZP to account for the measured CBC and thus provides a best estimate MTC prediction. Thus, the MTC Alternate Surveillance test at HZP is capable of detecting these potential problems with the same accuracy as the MTC Surveillance test at HZP. This results in the same ability to prevent similar MTC noncompliance problems when using the STAR Program.

The impact of the change on the detection of MTC noncompliance is determined to be acceptable based on the evaluation performed in Section E.2.2.5 of Appendix E. [

]

[

]

In addition, the added MTC Alternate Surveillance at HZP provides a method of detecting MTC noncompliance at HZP. This test replaces the MTC Surveillance test at HZP and provides an alternate means of MTC surveillance prior to performing the MTC Surveillance test at power. [

]

Some CE Plants have eliminated a MOC MTC Surveillance test contingent on the results of the BOC MTC Surveillance tests at HZP and power in accordance with Reference 2. For these plants reliance on the MTC Surveillance test at power to make this determination is sufficient. [

]

4.2.2.6 Impact on SDM Noncompliance Detection

SDM noncompliance detection is the detection of SDM values that are outside Technical Specification limits. Appendix A provides the results of a review of industry problems and Table A-8 summarizes the results for as-built core problems. The review identified one instance of SDM noncompliance involving shutdown CBC that was detected by core design QA. The STAR Program does not impact core design QA or CBC surveillances.

The impact of the change on the detection of SDM Noncompliance is determined to be acceptable based on the evaluation performed in Section E.2.2.6 of Appendix E. [

]

The verification of SDM using measured startup test parameters is not a test in the Generic Program in Section 3.1.2. Although verification of SDM may be a Technical Specification requirement[10] at HZP following a refueling for some plants, there is no typical use of measured startup test parameters, and no requirement for the use of measured startup test parameters. The various practices for using measured startup test parameters were

[10] This is not a requirement in the CE Standard Technical Specifications, Reference 3.

considered to be deviations from the Generic Program. The impact of the changes on these deviations is evaluated in Appendix F that addresses deviations from the Generic Program by Participating Plants. The detection of errors in the individual parameters affecting SDM in the STAR Program and the core design QA process are judged to be more effective than the SDM surveillance typically performed at HZP following a refueling.

4.2.2.7 Impact on Fuel Fabrication Error Detection

Fuel fabrication error detection is the detection of as-built fuel characteristics that are different from the intended design. Potentially affected as-built fuel characteristics include enrichment, poison loading, fuel pellet placement and size, fuel rod placement, and poison rod placement. Appendix A provides the results of a review of industry PWR problems and Table A-8 summarizes the results for as-built core problems. This review identified fourteen instances of fuel fabrication errors. Eight were detected by fuel fabrication QA prior to fuel shipment, three were detected by fuel receipt inspection at the utility and three were detected by the Incore Power Distribution test at power. The STAR Program does not impact fuel fabrication QA, fuel receipt inspection or the Incore Power Distribution test at power. [

]

The impact of the change on the detection of fuel fabrication errors is determined to be acceptable based on the evaluation performed in Section E.2.2.7 of Appendix E. [

]

In addition, the likelihood of fabrication errors has decreased since the current startup tests were established because of improvements that have been instituted in the manufacturing process. For example, gamma scanning of loaded fuel rods to check enrichment, and the use of bar codes instead of serial numbers for rod tracking, has reduced the likelihood of fuel fabrication errors. Fabrication errors have generally been detected in the fabrication shop, which indicates effective quality control practices are being employed. It is also noted that the primary means of detecting credible fuel fabrication errors in the core are similar to those of credible fuel misloadings. Fuel misloadings are discussed in Section 4.2.2.8 and analyses have been performed to demonstrate the acceptability of undetectable fuel misloadings. None of the detection methods credited in these analyses are eliminated in the STAR Program.

4.2.2.8 Impact on Fuel Misloading Detection

Fuel misloading detection is the detection of errors in the placement of fuel in the core during core loading. This could involve the placement of fuel in an incorrect location or orientation. Appendix A provides the results of a review of industry PWR problems and Table A-8 summarizes the results for as-built core problems. This review identified five instances of fuel misloading errors. One was detected by core design QA, two were detected by the Incore Flux Symmetry test at power, and two were detected by the Incore Power Distribution test at power. The STAR Program does not impact core design QA or the Incore Flux Symmetry and Incore power Distribution tests at power.

The impact of the change on the detection of fuel misloading is determined to be acceptable based on the evaluation performed in Section E.2.2.8 of Appendix E. [

]

[

]

4.2.2.9 Impact on Fuel Distortion Detection

Fuel distortion detection is the detection of changes in fuel assembly geometry that result in core operating characteristics different from design assumptions. Excessive fuel assembly distortions can be the result of operation in the reactor or the result of damage incurred during fuel handling. Reactor operation can result in fuel distortions such as bowing. Appendix A provides the results of a review of industry problems and Table A-8 summarizes the results for as-built core problems. The review identified eight instances of fuel distortion. Three were detected by the CEA Drop Time test while shutdown, two were detected by CEA manipulations, one was detected by CEA inspection and two were detected by CEA trips. The STAR Program includes the CEA Drop Time test and does not impact any of the other methods of detection. Fuel damage from fuel handling has usually been detected and characterized by visual inspections prior to operation. The visual inspections have usually been initiated in response to fuel handling events or visual observations of apparent anomalies.

The impact of the change on the detection of fuel distortion is determined to be acceptable based on the evaluation performed in Section E.2.2.9 of Appendix E. [

]

4.2.2.10 Impact on Fuel Poison Loss Detection

Fuel poison loss detection is the detection of burnable poison degradation that results in the loss of neutron absorber material. Appendix A provides the results of a review of industry problems and Table A-8 summarizes the results for as-built core problems. The review did not identify any instances of fuel poison loss.

The impact of the change on the detection of fuel poison loss is determined to be acceptable based on the evaluation performed in Section E.2.2.10 of Appendix E. [

]

4.2.2.11 Impact on Fuel Crudding Detection

Fuel crudding detection is the detection of deposits of material from the coolant on the outside of fuel rods. Significant fuel crudding can result in fuel failure due to temperature and corrosion effects. Appendix A provides the results of a review of industry PWR problems and Table A-8 summarizes the results for as-built core problems. The review identified five instances of fuel crudding that were detected by the Incore Power Distribution test at power. This test is included in the STAR Program.

The impact of the change on the detection of fuel crudding is determined to be acceptable based on the evaluation performed in Section E.2.2.11 of Appendix E. [

]

[

]

4.2.2.12 Impact on CEA Fabrication Error Detection

CEA fabrication error detection is the detection of as-built CEA characteristics that are different from the intended design. Appendix A provides the results of a review of industry problems and Table A-8 summarizes the results for as-built core problems. The review identified one instance of a CEA fabrication error that was detected by CEA fabrication QA. The STAR program does not impact fabrication QA. [

]

The impact of the change on the detection of CEA fabrication errors is determined to be acceptable based on the evaluation performed in Section E.2.2.12 of Appendix E. [

]

4.2.2.13 Impact on CEA Misloading Detection

CEA misloading detection is the detection of errors in the placement of CEAs in the core during core loading. This could involve the placement of CEAs in an incorrect location or orientation. Appendix A provides the results of a review of industry problems and Table A-8 summarizes the results for as-built core problems. The review did not identify any instances of CEA misloading.

The impact of the change on the detection of CEA misloading is determined to be acceptable based on the evaluation performed in Section E.2.2.13 of Appendix E. [

]

4.2.2.14 Impact on CEA Uncoupling Detection

CEA uncoupling detection is the detection of the failure to couple a CEA properly. Appendix A provides the results of a review of industry problems and Table A-8 summarizes the results for as-built core problems. The review identified eight instances of CEA Uncoupling. Four were detected by the CEA Flux Symmetry test at HZP, one was detected by the Incore Flux Symmetry test at power, one was detected by the Incore Power Distribution test at power and two were detected by CEA position indication. The STAR Program includes the Incore Flux Symmetry and Incore Power Distribution tests, and does not impact CEA position indication. The

CEA Flux Symmetry test at HZP is not included in the Generic Program because it has been eliminated from most startup test programs. The Incore Flux Symmetry test at power is an alternative test recommended by ANSI in Reference 1 and is included in the STAR Program. Other effective methods for detecting CEA uncoupling are currently being used that are not impacted by the STAR Program. These include verifications performed during the CEA coupling process and the CEA Drop Time Characteristics test.

The impact of the change on the detection of CEA uncoupling is determined to be acceptable based on the evaluation performed in Section E.2.2.14 of Appendix E. [

]

In addition, the likelihood of operating with an uncoupled CEA has decreased since the current startup tests were established because of improvements that have been instituted. For example, criteria on CEA drop characteristics from CEA Drop tests have enhanced the ability to detect and correct uncoupled CEAs prior to criticality.

4.2.2.15 Impact on CEA Distortion Detection

CEA distortion detection is the detection of changes in CEA geometry that affect the ability of CEAs to move as designed. Of particular concern is the ability of CEAs to trip as designed. CEA degradation such as cracking indicates the presence of strain and is included in CEA distortion. Strain may affect the ability of CEAs to move as designed and may be a precursor to CEA absorber loss. Appendix A provides the results of a review of industry problems and Table A-8 summarizes the results for as-built core problems. The review identified twelve instances of CEA distortion. Ten were detected by CEA inspections, one was detected EOC CEA insertion, and one was detected by CEA manipulation. [

]

The impact of the change on the detection of CEA distortion is determined to be acceptable based on the evaluation performed in Section E.2.2.15 of Appendix E. [

]

4.2.2.16 Impact on CEA Absorber Loss Detection

CEA absorber loss detection is the detection of CEA degradation that results in the loss of neutron absorber material. Degradation of control elements that involve a loss of CEA integrity can result in the subsequent leaching of absorber through defects in the control elements, or the physical transport of intact absorber through large defects. Additionally, mechanical interference of CEAs may also occur due to the CEA distortion. Appendix A provides the results of a review of industry PWR problems and Table A-8 summarizes the results for as-built core problems. The review identified four instances of CEA absorber loss. One was detected by EOC CEA insertion, one was detected by CEA manipulations, and two were detected by CEA inspections. [

]

[

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The impact of the change on the detection of CEA absorber loss is determined to be acceptable based on the evaluation performed in Section E.2.2.16 of Appendix E. [

]

[11] Uncertainties in key parameters that affect the CEA lifetime should be conservatively accounted for CEA lifetime predictions. Parameters such as CEA insertion and CEA dimensions may affect lifetime predictions.

[

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4.2.2.17 Impact on CEA Finger Loss Detection

CEA finger loss detection is the detection of the physical separation of CEA fingers from CEAs. The separated finger subsequently remains in the fuel while the CEA is withdrawn. Appendix A provides the results of a review of industry problems and Table A-8 summarizes the results for as-built core problems. The review identified four instances of CEA finger loss. One was detected by the Incore Power Distribution test at power, one was detected by CEA manipulations and two were detected by CEA inspections. The STAR Program includes the Incore Power Distribution test and does not impact CEA manipulations or CEA inspections for CEA finger loss.

The impact of the change on the detection of CEA finger loss is determined to be acceptable based on the evaluation performed in Section E.2.2.17 of Appendix E. [

]

4.2.2.18 Impact on RCS Anomaly Detection

RCS anomaly detection is the detection of anomalous changes in local RCS parameters such as temperature or flow. Appendix A provides the results of a review of industry PWR problems and Table A-8 summarizes the results for as-built core problems. The review did not identify any instances of RCS anomalies.

The impact of the change on the detection of RCS anomalies is determined to be acceptable based on the evaluation performed in Section E.2.2.18 of Appendix E. [

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[12] Nondestructive examinations of CEAs using techniques such as eddy current or ultrasonic typically do not detect absorber loss directly but instead detect the clad defects that are present when CEA absorber loss occurs

[13] The CEA inspection that detected the cracking was not performed in the outage in which the CEA was discharged. The inspection was performed in the following outage after CEA absorber loss was detected in other CEAs.

4.2.2.19 Impact on RCS B-10 Depletion Detection

RCS B-10 depletion detection is the detection of the proportion of the isotope B-10 in the RCS boron. B-10 has a high cross section for neutron absorption and becomes depleted with prolonged exposure to the neutron flux. This could result in the RCS boron being a less effective neutron absorber than assumed in supporting analyses. The changes in B-10 with exposure are generally understood and are best detected through isotopic analysis of boron samples and the use of boron rundown curves to track changes relative to prediction. Appendix A provides the results of a review of industry problems and Table A-8 summarizes the results for as-built core problems. The review did not identify any instances of B-10 depletion problems. B-10 depletion occurs continuously but does not represent a problem unless it exceeds assumptions about the extent of depletion. Although this was once an issue at some plants, currently B-10 depletion is understood and managed to prevent adverse impacts on safety analysis conformance. B-10 depletion typically accounted for in ECPs and reactivity balances.

The impact of the change on the detection of B-10 depletion is determined to be acceptable based on the evaluation performed in Section E.2.2.19 of Appendix E. [

]

4.3 IMPACT OF CHANGES ON TEST PERFORMANCE PROBLEMS

This section evaluates the impact of the changes to the Generic Program on test performance problems. In this evaluation, test performance problems are test initiated errors that have the potential for significantly impacting the operation of the core. The identification of test performance problems was based on a review of startup test performance activities to determine associated practices that have the potential for causing errors that impact core operation. Unique operating practices involving equipment and processes necessary to support testing may cause errors that impact operation. In addition, normal operating practices involving infrequently performed reactivity maneuvers as part of the test process may also cause errors that impact operation. Finally, errors in test results[14] have the potential of impacting plant operation through the substitution of measured values for predicted values in operating instructions. Operating instructions are any instructions that have the potential to affect plant operation. The following are the test performance problems that are identified for evaluation:

- Test equipment errors
- Test process errors
- Test result errors

4.3.1 Description of Test Performance Problem Evaluation

This section describes the following elements of the test performance problem evaluation:

- Criterion used in the evaluation
- Information required for the evaluation
- Process used to perform the evaluation

[14] In this evaluation, the impact of the inherent uncertainty associated with the test measurement is not considered to be an error in the test result.

4.3.1.1 Test Performance Problem Evaluation Criterion

The test performance criterion is based on ensuring that test performance errors in the STAR Program will not increase the likelihood of causing operation outside the safety analysis. The following evaluation criterion is used in the evaluation of test performance problems to ensure the general evaluation criterion in Section 4.0 is satisfied:

- [

]

4.3.1.2 Test Performance Problem Information

In order to determine if the evaluation criterion is satisfied it is necessary to determine the likelihood of startup tests initiating operation outside the safety analysis. Appendix D addresses the initiation of test performance problems. Table D-1 provides the likelihood of startup tests initiating test performance problems. A three level rating system is used to represent the likelihood of a startup test initiating a problem. The likelihood of a test initiating test performance problems is rated as "greatest," "intermediate" or "smallest" depending on whether unique operating practices, normal operating practices, or operating instructions are involved. Unique operating practices have the greatest likelihood of initiating operation outside the safety analysis. Normal operating practices have only an intermediate likelihood of initiating operation outside the safety analysis. Operating instructions have the smallest likelihood of initiating operation outside the safety analysis. Appendix D provides the background on establishing these ratings. This information is used in the evaluations to determine changes in the likelihood of initiating test performance problems between the Generic and STAR Programs.

4.3.1.3 Test Performance Problem Evaluation Process

Figure 4-3 presents a flowchart of the evaluation process for test performance problems. The process evaluates the impact of the changes on the test performance problems identified in Section 4.3 by determining if the change is acceptable using the test performance criterion. The evaluation process for test performance problems consists of the following four steps:

[

]

[15] If one of the following pairs of tests is already being performed at the same conditions, then adding the second will not require additional measurements because the data from the first can be used for the second:

- CEA Drop Time and CEA Drop Characteristics
- CBC and MTC Alternate Surveillance
- ITC and MTC Surveillance

[

]

A summary of the impacts on the likelihood of operation outside the safety analysis is provided in Table 4-3. A summary of all the impacts associated with the changes to the Generic Program is provided in Table 5-1.

4.3.2 Test Performance Problem Evaluation

This section provides an individual evaluation for each of the three test performance problems potentially impacted by startup testing. The corresponding evaluations using the process described in Section 4.3.1 are provided in Appendix E. Significant results from these evaluations are summarized below. A summary of the impacts on the likelihood of operation outside the safety analysis is provided in Table 4-3. A summary of all the impacts associated with the changes to the Generic Program is provided in Table 5-1.

4.3.2.1 Impact on Test Equipment Errors

Test equipment errors are errors associated with the installation of unique equipment required to support startup testing. The use of test equipment constitutes a unique operating practice that may have a credible likelihood of initiating operation outside the safety analysis. Unique operating practices include the use of a reactivity computer. Appendix A provides the results of a review of industry problems and Table A-9 summarizes the results for test performance problems. The review identified twelve instances of test equipment errors. Six occurred during the CEA Worth test and one occurred during the CEA Flux Symmetry test. Specific tests were not identified for the remainder, but all the test equipment errors occurred during low power physics tests. Six of the errors involved the reactivity computer directly and most involved CEAs. It is likely that none of the test equipment errors would have occurred had the STAR Program been used.

The impact of the change on the likelihood of initiating operation outside the safety analysis due to test equipment errors is determined to be acceptable based on the evaluation performed in Section E.2.3.1 of Appendix E. [

]

[

]

4.3.2.2 Impact on Test Process Errors

Test process errors are errors associated with performing the maneuvers that are required to support startup testing. These maneuvers may involve unique operating practices that are not otherwise used during operation as well as normal operating practices, and may have a credible likelihood of initiating operation outside the safety analysis. Unique operating practices include unique CEA configurations and the frequent interaction between operations and test personnel to determine plant operating maneuvers. Normal operating practices include reactivity maneuvers that require changes in CEA position, boron concentration, and temperature. Appendix A provides the results of a review of industry problems and Table A-9 summarizes the results for test performance problems. The review identified ten instances of test process errors. Six occurred during the CEA Worth test and one occurred during the ITC test. Specific tests were not identified for the remainder, but all the test process errors occurred during low power physics tests. Two of the errors involved the reactivity computer directly and most involved CEAs. It is likely that none of the test process errors would have occurred had the STAR Program been used.

The impact of the change on the likelihood of initiating operation outside the safety analysis due to test process errors is determined to be acceptable based on the evaluation performed in Section E.2.3.2 of Appendix E. [

]

4.3.2.3 Impact on Test Result Errors

Test result errors are errors associated with the measured results for parameters from startup testing. These errors can be caused by hardware malfunctions, as well as improper calibration, connection, operation, and reading of equipment used in the test. Although all measurements are subject to error, the more complex equipment such as the reactivity computer have a greater potential for causing test measurement errors than using normal plant instrumentation for startup test measurements. Test result errors have the potential of impacting plant operation through the substitution of measured values for predicted values in operating instructions. Operating instructions are judged to have a minimal likelihood of initiating operation outside the safety analysis because the test result error would have to involve a significant nonconservative measurement error and be within acceptance criteria for the test. In addition, predicted values rather than measured values are typically used in operating instructions when the test result is less conservative. Appendix A provides the results of a review of industry problems and Table A-9 summarizes the results for test performance problems. The review identified three instances of test result errors. All three occurred during the MTC Surveillance test. The likelihood of test result errors associated with these tests would be similar when using the STAR Program.

The impact of the change on the likelihood of initiating operation outside the safety analysis due to test result errors is determined to be acceptable based on the evaluation performed in Section E.2.3.3 of Appendix E. [

]

Table 4-1 Summary of Impacts on the Ability to Ensure Uncertainties are Bounded by the Safety Analysis

[1] Definitions and evaluations of these problems are provided in the report sections indicated by the number before the problem title.

Table 4-2 Summary of Impacts on the Overall Effectiveness in Detecting As-Built Core Problems

[1] Definitions and evaluations of these problems are provided in the report sections indicated by the number before the problem title.

Table 4-3 Summary of Impacts on the Likelihood of Initiating Operation Outside the Safety Analysis

[1] Definitions and evaluations of these problems are provided in the report sections indicated by the number before the problem title.

FIGURE 4-1 Flowchart of Design Prediction Problem Evaluation Process

FIGURE 4-2 Flowchart of As-Built Core Problem Evaluation Process

FIGURE 4-3 Flowchart of Test Performance Problem Evaluation Process

5.0 CONCLUSIONS

This section summarizes conclusions from the evaluation of the STAR Program in Sections 4.0 and Appendix F. The STAR Program consists of the tests in Table 3-3 and the STAR Applicability Requirements in Table 3-4. The STAR Program involves the following changes to the Generic Program:

1. Elimination of the CEA Worth test at HZP
2. Elimination of the ITC test at HZP
3. Elimination of the MTC Surveillance test at HZP[16]
4. Addition of a MTC Alternate Surveillance test at HZP
5. Addition of an ITC test at intermediate to HFP
6. Addition of a ΔCBC HZP-HFP test at HFP
7. Addition of Core Design Applicability Requirements
8. Addition of Fabrication Applicability Requirements
9. Addition of Refueling Applicability Requirements
10. Addition of a Startup Testing Applicability Requirement
11. Addition of CEA Lifetime Applicability Requirements

This change eliminates the following two measurements at HZP:

1. CEA Worth
2. ITC

The elimination of these two measurements at HZP eliminates the need for the reactivity computer and permits the use of normal operating practices during startup testing. The CEA worth measurement is eliminated from the startup test program while an ITC measurement is performed later in the program during power operation. Additional tests may be performed when using the STAR Program including tests that require the use of the reactivity computer. Thus, elimination of only one of the above measurements at HZP is an option when using the STAR Program.

5.1 IMPACT OF CHANGES TO GENERIC STARTUP TEST PROGRAM

This section summarizes conclusions on the impact of the changes to the Generic Program on design prediction, as-built core, and test performance problems. The impact of the changes to the Generic Program evaluated in Section 4.0 demonstrate the following:

- The ability of the STAR Program to prevent operation with problems is essentially the same as, or better than, the Generic Program.

In this report, problems are core configurations that are not explicitly accounted for in the safety analysis. Thus, the prevention of problems results in safety analysis conformance and assurance that the core can be operated as designed. This satisfies the purpose of startup testing which is to ensure the core can be operated as designed. The results of the evaluation of the impact of the changes to the Generic Program are summarized in Table 5-1. Conclusions with respect the changes are presented below.

[16] For plants that have eliminated the MOC MTC Surveillance test contingent on the results of the BOC MTC Surveillance tests at HZP and power in accordance with Reference 2, reliance on the MTC Surveillance test at power to make this determination is acceptable.

5.1.1 Conclusions from the Evaluation of Design Prediction Problems

The following was concluded from the evaluation of design prediction problems:

- The uncertainties of parameters that are measured in the Generic Program, but not measured in the STAR Program, are bounded by the safety analysis when using the STAR Program.

The change does not impact the accuracy that can be supported for the core design methods used to predict core parameters. The ITC accuracy that can be supported remains unchanged by adding an ITC Test at power to replace the ITC Test at HZP. [

]

In general, the following can be concluded relative to design prediction problems:

- The incidence of significant problems associated with predictions of CEA worth and MTC has been very low.

5.1.2 Conclusions from the Evaluation of As-Built Core Problems

The following was concluded from the evaluation of as-built core problems:

- The ability of the STAR Program to prevent operation with as-built core problems is essentially the same as, or better than, the Generic Program.

[

] The ability to detect MTC Noncompliance problems at HZP is preserved by requiring an MTC Alternate Surveillance test at HZP. [

]

The following can be concluded relative to as-built core problems:

- The STAR Program has the same ability to detect problems using ITC and MTC Surveillance tests, although some may be detected at power instead of HZP.

- The STAR Program is expected to detect problems related to the power distribution that would be detected by the CEA worth tests at HZP.

- The STAR Core Design Applicability Requirements enhance the detection of ITC, MTC and CEA worth error problems in the core design process prior to HZP.

- [

  ]

- The eliminated CEA worth test at HZP has not been effective in detecting as-built core problems.

5.1.3 Conclusions from the Evaluation of Test Performance Problems

The following was concluded from the evaluation of test performance problems:

- The STAR Program decreases the likelihood of operation outside the safety analysis due to test performance problems.

The likelihood of operational problems is reduced due to the elimination of unique operating practices during startup testing. Unique operating practices include the use of the reactivity computer, unique CEA configurations, and the frequent interaction between operations and test personnel on plant operating maneuvers. The STAR Program does retain some reactivity maneuvers to support ITC measurements but does not require unique operating practices to support testing.

The following can be concluded relative to test performance problems:

- Elimination of tests that use the reactivity computer results in startup tests that require only normal plant operating practices[17].

- Performing startup tests using normal plant operating practices decreases the likelihood of having operational problems associated with testing.

- Problems related to tests that involve CEA worth measurements and the reactivity computer at HZP have resulted in operational problems and test delays.

5.2 ACCEPTABILITY OF STAR PROGRAM

This section presents conclusions concerning the acceptability of the STAR Program for implementation in both the participating and non-participating PWR plants. The STAR Program consists of the tests in Table 3-3 and the STAR Applicability Requirements in Table 3-4. The acceptability of the changes is based on an evaluation of the impact the changes have on safety analysis conformance. The results for the evaluation in Section 4.0 of the acceptability of the changes to the Generic Program are summarized in Table 5-1. The results for the evaluation in Appendix F of the acceptability of the changes to deviations from the Generic Program for Participating Plants are summarized in Table 5-2.

5.2.1 Acceptability of STAR Program for Participating Plants

The following conclusions address the acceptability of the STAR Program for Participating Plants:

- **Implementation of the STAR Program in the Participating Plants is acceptable.** This conclusion is based on the evaluations summarized in Table 5-1 and 5-2 that demonstrate acceptable results for the impact of the STAR Program on safety analysis conformance. These results are demonstrated for both the changes to the Generic Program and the elimination of deviations from the tests in Generic Program by Participating Plants.

[17] The proposed startup test program does not preclude the use of unique operating practices. Technical Specification special test exceptions may be used to perform measurements. For instance, the use of CEA configurations outside the safety analysis using special test exceptions may continue for CEA worth measurements at HZP or the ITC measurement at power.

- **A CEA Flux Change test based on either measured reactivity changes or startup rates is an acceptable alternative to the CEA Drop Characteristics test.** This is based on the conclusion in Appendix F that the CEA Flux Change Test is an effective means of detecting CEA uncoupling. Modifying the CEA Flux Change test at HZP to measure startup rate instead of reactivity using the reactivity computer was also found to be acceptable in Appendix F.

- **The continued elimination of the MOC at power ITC measurement to verify EOC MTC Technical Specification compliance is acceptable for plants that have already eliminated this measurement in accordance with Reference 2.** For these plants it is acceptable to rely on the BOC MTC Surveillance test at power to determine if the criteria for eliminating the MOC MTC Surveillance test is satisfied. This is based on the conclusion in Appendix B that the ITC startup test data between different operating conditions is poolable.

5.2.2 Acceptability of STAR Program for Non-Participating PWR Plants

The following conclusions address the acceptability of the STAR Program for non-participating PWR plants:

- **Implementation of the STAR Program in the non-participating PWR plants[18] is acceptable provided there are no relevant unique design features that require additional startup testing.** This conclusion is based on the evaluations summarized in Table 5-1 that demonstrate acceptable results for the impact of the STAR Program on safety analysis conformance. These results are demonstrated for the changes to the Generic Program but not for the elimination of additional tests that deviate from the Generic Program. Any changes to deviations from Generic Program by non-Participating Plants would have to be evaluated on an individual basis.

- **A CEA Flux Change test based on either measured reactivity changes or startup rates is an acceptable alternative to the CEA Drop Characteristics test.** This is based on the conclusion in Appendix F that the CEA Flux Change Test is an effective means of detecting CEA uncoupling. Modifying the CEA Flux Change test at HZP to measure startup rate instead of reactivity using the reactivity computer was also found to be acceptable in Appendix F.

[18] This includes CE Plants, Westinghouse Plants and B&W Plants.

Table 5-1 Impacts associated with Changes to the Generic Program

CATEGORY	PROBLEM	SECTION[1]	IMPACTED	CRITERIA SATISFIED	RESULT
Design Prediction	CEA Worth Inaccuracy	4.1.2.1	Yes	Yes	Acceptable
	CBC Inaccuracy	4.1.2.2	No	Not Applicable	Acceptable
	ITC Inaccuracy	4.1.2.3	Yes	Yes	Acceptable
	Power Distribution Inaccuracy	4.1.2.4	No	Not Applicable	Acceptable
As-Built Core	CEA Worth Error	4.2.2.1	Yes	Yes	Acceptable
	CBC Error	4.2.2.2	Yes	Yes	Acceptable
	ITC Error	4.2.2.3	Yes	Yes	Acceptable
	Power Distribution Error	4.2.2.4	Yes	Yes	Acceptable
	MTC Noncompliance	4.2.2.5	Yes	Yes	Acceptable
	SDM Noncompliance	4.2.2.6	Yes	Yes	Acceptable
	Fuel Fabrication Error	4.2.2.7	Yes	Yes	Acceptable
	Fuel Misloading	4.2.2.8	Yes	Yes	Acceptable
	Fuel Distortion	4.2.2.9	Yes	Yes	Acceptable
	Fuel Poison Loss	4.2.2.10	Yes	Yes	Acceptable
	Fuel Crudding	4.2.2.11	Yes	Yes	Acceptable
	CEA Fabrication Error	4.2.2.12	Yes	Yes	Acceptable
	CEA Misloading	4.2.2.13	Yes	Yes	Acceptable
	CEA Uncoupling	4.2.2.14	Yes	Yes	Acceptable
	CEA Distortion	4.2.2.15	No	Not Applicable	Acceptable
	CEA Absorber Loss	4.2.2.16	Yes	Yes	Acceptable
	CEA Finger Loss	4.2.2.17	Yes	Yes	Acceptable
	RCS Anomaly	4.2.2.18	No	Not Applicable	Acceptable
	RCS B-10 Depletion	4.2.2.19	No	Not Applicable	Acceptable
Test Performance	Test Equipment Error	4.3.2.1	Yes	Yes	Acceptable
	Test Process Error	4.3.2.2	Yes	Yes	Acceptable
	Test Result Error	4.3.2.3	Yes	Yes	Acceptable

[1] Definitions and evaluations of these problems are provided in the indicated report sections.

Table 5-2 Impacts associated with Changes to the Generic Program Deviations[1]

(Applicable to Participating[2] Plants Only)

CHANGES TO DEVIATION	SECTION[3]	CRITERIA SATISFIED	RESULT
Eliminating or Modifying the CEA Flux Change Test at HZP	F.3.2.1	Yes	Acceptable
Eliminating the Rodded CBC Test at HZP	F.3.2.2	Yes	Acceptable
Eliminating the IBW Test at HZP	F.3.2.3	Yes	Acceptable
Eliminating the SDM Surveillance Test at HZP	F.3.2.4	Yes	Acceptable
Eliminating the CEA Flux Symmetry Test at HZP	F.3.2.5	Yes	Acceptable

[1] Deviations in startup testing by Participating Plants from the Generic Program.
[2] The Participating Plants are the subset of CE Plants that are participants in CEOG Task 1173, Startup Test Reduction. These plants are ANO 2, Waterford 3, Millstone 2, SONGS 2 &3, Calvert Cliffs 1 & 2, St. Lucie 1 & 2 and Ft. Calhoun
[3] Definitions and evaluations of these problems are provided in the indicated report sections.

6.0 REFERENCES

1. ANSI/ANS-19.6.1-1997, "American National Standard Reload Startup Physics Tests for Pressurized Water Reactors," August 22, 1997.

2. Amendment 1 to CE NPSD-911-P-A, "Analysis of Moderator Temperature Coefficients in support of a Change in the Technical Specification End of Cycle Negative MTC Limit," January 1998.

3. NUREG-1432, Rev.2, "Standard Technical Specifications Combustion Engineering Plants," June 2001.

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APPENDIX A

REVIEW OF INDUSTRY PROBLEMS

A.1 INTRODUCTION

A.1.1 Background

This appendix provides the results of a review of past industry problems obtained from searches of various industry databases. The methods that detected the problems are identified where possible. In addition, other information that is relevant to the impact of the STAR Program on problems is summarized. Included are the causes of the problems and corrective actions to prevent recurrence.

This review is based on a sampling of industry problems and does not necessarily include the worst-case instance for each problem. The review does provide a reasonable sampling of past problems and is thus representative of industry experience. The sampling included a search of NRC and INPO databases to identify problems experienced in the industry.

The NRC and INPO databases where chosen since the problems of interest are limited to significant problems that remained uncorrected prior to the beginning of Startup Tests. Such problems are, in general, expected to be reportable under NRC Regulations and likely reported to INPO because of the potential impact on the industry. Although a search of Westinghouse's and/or licensees' corrective action program databases would have identified additional problems, most of these would not be applicable to the STAR Program for the following reasons:

- The problem was not significant, i.e., small compared to parameter uncertainties and other margins in the safety analysis.
- The problem was detected and corrected prior to the initiation of startup testing.
- The problem was not relevant to the STAR Program.

Therefore, many of the errors and other problem associated with analyses and measurements reported in Westinghouse's and/or licensees' corrective action program databases, that may otherwise be expected to be identified, are not identified since they are, in general, inconsequential. Note that in specific circumstances, which relate most directly to the tests impacted by the STAR Program, the NRC and INPO databases are supplemented by information on specific problems either (a) from Westinghouse's reports to its customers on various technical issues related to CE Plants or (b) supplied by Participating Plants. Examples are information pertaining to the following as-built core problems:

- CEA Worth errors
- CBC errors
- ITC errors
- Power Distribution errors
- Fuel Fabrication Errors
- Fuel Misloadings
- CEA Uncoupling
- CEA Distortion
- CEA Absorber Loss

A.1.2 A.1.2 Purpose

The purpose of this appendix is to provide information from past industry experience that is relevant to the impact of the STAR Program on problems.

A.2 SEARCH RESULTS

The principal sampling of industry experience included searches of Nuclear Regulatory Commission (NRC) and Institute for Nuclear Power Operations (INPO) databases. The searches focused on design prediction, as-built core and test performance problems relevant to the STAR Program. The specific databases searched included the NRC's Public Document Room and INPO's (SOER, EPIX, etc.). In addition to these industry databases, Westinghouse searched its internal corporate technical issue databases for CE Plants (i.e., TechNotes, InfoBulletins) and reviewed information on problems provided by participating plants. In some instances overlap exists in the documents addressing specific problems. Furthermore, some documents, such as Generic Letters, Information Notices and INPO SOERs, address similar problems at different plants. Therefore, the number of entries in tables providing search results may not reflect the total number of problems for each problem area.

A.2.1 Description Of Review Process

Keywords, phrases and strings were developed to execute the database searches in a focused and efficient manner to identify potential documents addressing STAR problems. In some cases, these keywords, phrases and strings were linked using logical connectors (e.g., and, or, near, etc.) to further zero in on results directly applicable to STAR problems. Because the capabilities or characteristics of the search engines employed by the various databases explored varied, it was necessary to adjust the nature of the searches between databases in order to optimize search performance and results.

A.2.1.1 Nuclear Regulatory Commission (NRC) Database Searches

The NRC Public Document Room (PDR) legacy database was searched from the time of the database origin to approximately the present time (actually ~mid-1970s through 1999). The STAR Program keywords, phrases and strings in Table A-1 were used in the search. Time was not spent searching the ADAMS database (i.e., 1999 to present) since the INPO database search covered from the early 80s to present. For the PDR database, only selected document types were searched. For example, Licensee Events Reports (LERs) were determined to be a likely high yield source of information in comparison to Topical Reports which were felt to be a source of little to no viable information with respect to the goals of the STAR Program. The PDR document types searched using the keywords, phrases and strings is presented in Table A-2.

A.2.1.2 Institute of Nuclear Power Operations (INPO) Database Searches

A search of the INPO Web Site was made for the STAR Program keywords, phrases and strings in Table A-3. The topic areas of the INPO Web Site that were searched are identified in Table A-4. Note that the search for selected keywords in Table A-3 were limited to "All OE Topics" including Operating Experience Reports (OEs), Operations And Maintenance Reminders (O&MRs), Significant Event Notifications (SENs), Significant Event Reports (SERs), and Significant Operating Experience Reports (SOERs). This was done to minimize identification of documents relating to routine topics areas identified in Table A-4. Note that limiting the search to "All OE Topics" is considered acceptable since all significant events should be captures under this topic area. The searches were conducted from the Advanced Search Page of the Nuclear Network Page at the INPO Web Site. The INPO searches covered the period from the early 80s to present and overlapped the search of the NRC PDR database between the early 80s to approximately 1999. In addition, the following relating to the INPO database search is noted:

- Knowing that there are misspellings and to get variations in the wording, the search string was sometimes a subset of the actual term. For example, "misload" was searched to find "misloading" on the basis that this would locate instances in which *misload, misloaded* and *misloading* were present.
- Based on reading the abstracts returned or the actual documents, specific documents were copied to a common file.
- Documents with abstracts which were insufficient to identify their content were opened to determine their applicability to the search criteria.

It should be noted that the INPO search engine limits the returns to 300 documents. In cases where 300 documents were identified, there may be more but the search was not expanded to capture all potential documents on the premise that the original 300 documents was a reasonable sample size. An exception was the search for "reactivity" which was broken into two periods to capture all of the applicable documents.

The INPO search involved fewer keywords than the NRC search because it did not use "and" logic to limit the number of hits that were not applicable to the STAR Program problems. In addition, some keywords were not used in the INPO search because they resulted in excessive hits that were not applicable to the STAR Program problems and were considered to be embodied in the results of the other searches conducted.

A.2.1.3 Westinghouse Technical Issue Databases

Starting in ~1979, CE issued reports to its customers on various technical issues related to CE Plants that would be of interest to them, including the fuel and CEAs. These reports were transmitted as either a InfoBulletin (1979-1999) or TechNote (1990-1999). Since theses databases are relatively small in comparison to the NRC and INPO databases, the keywords, phrases and strings listed in Tables A-1 and A-4 were not employed. Rather, the title indices for the documents contained in these databases was reviewed and potentially relevant documents were identified and extracted for a detailed review to determine whether or not they were actually pertinent to the goals of the STAR Program. Fifteen (15) potentially applicable reports were identified, retrieved and reviewed in order to assess actual applicability. In addition, one problem relating to a CE plant was reported in a Westinghouse Nuclear Safety Advisory Letter (NSAL). Finally, Westinghouse Pittsburgh personnel, who support Startup Testing at Westinghouse plants, identified additional events that have occurred in Westinghouse plants that are applicable to the STAR Program problems.

A.2.1.4 Participating Plant Information

Participating plants were requested to respond to a data request and survey to support the STAR Program. The data request and survey included a request to provide information relating to the following:

- Startup Tests that failed the acceptance criteria for the last five startups.
- Core misloading (misloaded or misrotated assemblies) events at their plants that were not identified before or during core verification.
- Known core misloading events at other plants.
- Misloading of fuel pellets or fuel rods at Hematite or Columbia Fuel Fabrication Facilities.
- Misloading of fuel pellets or fuel rods at other Fuel Fabrication Facilities.
- Uncoupled CEA events (uncoupled CEAs that were not identified by checking of heights and weights following coupling) at their plants.
- Known uncoupled CEA events at other plants.

- Events at their plants that resulted in loss of CEA integrity due to fabrication, handling damage or usage.

The responses from the participants were used to supplement the information from the NRC and INPO databases searched and are included in the search results.

A.2.2 Search Results

By exercising the keywords, phrases and strings in the various databases, over 5700 documents were identified, most of which had no applicability to problems relevant to the STAR Program. A review of the raw search results by document title and "abstract" was the principal method employed to pare down the extensive lists of search results to likely candidates for document retrieval and subsequent review (i.e., a 'hit'). These 'hits' were then retrieved from their respective source (NRC, INPO or Westinghouse) and reviewed to determine their applicability to STAR problems. The review of these selected documents resulted in further paring down of the 'hits' since the actual review, most times, revealed that they were not applicable to STAR problems. In the end, approximately 450 documents were retrieved and reviewed of which approximately 110 documents were found to be applicable to STAR problems. Tables A-5 and A-6 provide lists of the documents actually employed (i.e., those 'hits' that actually bore fruit) in establishing the industry problems that are applicable to STAR problems.

Tables A-5 and A-6 summarize the search results for industry problems and identify the relevant database identifier (e.g., ascension, INPO OE, SEN and SOER number), date of issuance, plant and plant type (where applicable), abstracted text, STAR problem area and estimated impact. Entries that were solely obtained from INPO documents do not identify the plant. Table A-5 provides the information for design prediction and as-built core problems along with the detection method while Table A-6 provides the information for test performance problems along with the initiating test. Tables A-7 and A-8 tabulate the methods that have detected past industry design prediction and as-built core problems respectively. This information is used to form conclusions about the frequency of detectable problems and the ability of the STAR Program to prevent operation with these problems. Table A-9 tabulates the tests that have initiated past test performance problems. This information is used to form conclusions about the initiation of test performance problems and the ability of the STAR Program to prevent operation with these problems.

A.3 CONCLUSIONS

The following is concluded based on the results of the review of past industry problems obtained from searches of various industry databases summarized in Tables A-5 through A-9:

1. The incidence of significant problems associated with predictions of CEA worth and MTC has been very low. This conclusion is based on the observation that only three CEA worth prediction problems and no ITC or MTC prediction problems were identified in the sample search. Although 30 instances of potential MTC noncompliance were identified, none involved predictions that were inconsistent with the typical uncertainties associated with MTC. Instead, the core designs involved predictions close to the Technical Specification limit and the measured values were within expectations based on uncertainties. In all cases corrective actions were implemented and no Technical Specification violations were identified.

2. The eliminated CEA worth test at HZP has not been effective in detecting as-built core problems. This conclusion is based on the observation that none of the ninety-three as-built core problems in the sample search were detected by the CEA worth test.

3. Problems related to tests that involve CEA worth measurements and the reactivity computer at HZP have resulted in operational problems and test delays. This conclusion is based on the observation that most of the twenty-six test performance errors identified in the sample search involved CEA worth measurements or the reactivity computer.

The above conclusions are based on a sampling of industry problems that does not include all instances of each problem. Regardless, a more exhausted search is not expected to change these conclusions.

Table A-1 Summary of NRC Database Search Keywords/Phrases/Strings

SUBJECT	SUBJECT ID (AND)	SUBJECT PROBLEM (OR)	GENERAL PROBLEM DESCRIPTIOR
CBC	CBC, boron, B10, B-10	Worth, reactivity, critical, concentration, rundown, depletion, measurement,	Problem, error, compliance, event, incident, surveillance, inspection, examination, violation, nonconservative, deficiency, defect, detect, damage, anomaly, margin, acceptance criteria, reactivity, occurrence
ITC	Isothermal temperature coefficient, ITC, moderator temperature coefficient, MTC	Measurement, Technical Specification, Tech Spec	
Power Distribution	Power distribution, peaking, incore analysis, Fr, FR, FRT, Fdh, FDH	tilt, roll, penalty, symmetry, asymmetry	
CEA	CEA, control element, RCCA, rod cluster, CRA, control rod, core verification, SDM, shutdown margin	Misloading, integrity, failure, strain, swelling, crack, interference, bowing	
Fuel	Fuel assembly, fuel element, fuel rod, nuclear fuel, enrichment, fuel bundle, core verification,	Misloading, bowing, crud, pressure drop, delta P	
Fuel Poison	Poison, burnable shim, WABA, burnable absorber, IFBA, erbia, gadolinia, gadolinium	Misloading, integrity, failure, strain, swelling, crack, interference, bowing	
RCS Anomaly	Coolant system anomaly, RCS anomaly, RCS temperature anomaly, RCS flow anomaly, temperature asymmetry, flow asymmetry	Asymmetry	
Startup Tests	Startup test, SUT, prediction, physics test, low power physics, LPPT, power ascension test, power escalation test, reactivity computer	Measurement, test result	

NOTE: A hit consists of one of the following:

1. A keyword from the **SUBJECT ID** column **AND** a keyword from the **SUBJECT PROBLEM** column **OR**
2. A keyword from the **SUBJECT ID** column **AND** a keyword from the **GENERAL PROBLEM DESCRIPTOR** column.

Table A-2 NRC Topic Areas Searched

- Licensee Events Reports (10 CFR50.72)
- Defects and Non-Compliance Reports (10 CFR 21)
- Deficiency Reports (10 CFR 50.55e)
- NRC Inspection Reports
- Reportable Occurrence Reports
- Abnormal Occurrence Reports
- Safety Evaluation Reports
- Notices of Violation
- Correspondence
- Generic Communications (Generic Letters, Bulletins, Information Notices, etc.)
- Investigation Reports
- NUREG Reports
- Test/Inspection/Operating Procedure

Table A-3 Summary of INPO Database Search Keywords/Phrases/Strings

SUBJECT	KEYWORD 1 (NEAR)	KEYWORD 2
CBC	CBC, boron worth, reactivity*, rundown*, depletion*	
ITC	isothermal temperature coefficient, ITC*, moderator temperature coefficient, MTC*	
Power Distribution	power distribution Fr*, FRT*, FDH*, tilt*, penalty*, asymmetry	peaking
CEA	control rod CEA*, control element*, RCCA, rod cluster, CRA, misload, strain*, swelling*, asymmetry, reactivity*	uncoupled, unlatched
Fuel	swelling*, bowing*, misload, crud*, asymmetry, reactivity*	
Fuel Poison	burnable, erbi, gadolin, misload, strain*, swelling*, bowing*, reactivity*	
RCS Anomaly	asymmetry, reactivity*	
Startup Tests	startup test*, physics test, reactivity*, ZPPT	

* Search for keywords limited to All OE Topics including Operating Experience Reports (OEs), Operations And Maintenance Reminders (O&MRs), Significant Event Notifications (SENs), Significant Event Reports (SERs), and Significant Operating Experience Reports (SOERs). Section A.2.1.2 provides further information on these searches

Table A-4 INPO Topic Areas Searched

- **Source Area**
  - Nuclear Network Web Pages only
  - Licensee Event Reports (LERs)
- **General Topics**
  - Coordination With INPO General
  - Meeting Information Announcements
  - Meeting Information Summaries
- **Emergency Preparedness**
  - General
  - Drills
  - Hotline
- **Technical Exchange**
  - **All Technical Exchange Topics**
- **Chemistry**
  - General
- **Computer Technology**
  - General
  - Business
  - Process Control
- **Corrective Action Programs**
  - General
- **Daily Plant Status**
  - **All Daily Plant Status Topics**
  - Events
  - Full Report
  - Preliminary Notification Of Occurrences
  - Scrams
  - Vendor Notifications
- **Engineering**
  - Design Engineering
  - Fuel Management
  - Inspection And Testing
  - System Engineering
- **Equipment Performance**
  - **All Equipment Performance Topics**
  - General
  - Breakers
  - Electrical
  - Heat Exchangers Or Steam Generators
  - Instrumentation And Control
  - Other Mechanical
  - Pumps
  - Structures
  - Switchyard
  - Turbines
  - Valves
- **Fire Protection**
  - General
- **Human Performance**
  - General

Table A-4 INPO Topic Areas Searched

- **Industrial Safety And Medical**
  - General
- **Just-in-Time Operating Experience**
  - Training/Briefing Material and Equipment Failure Experience
- **Licensing And Nuclear Safety**
  - **All Licensing And Nuclear Safety Topics**
  - General
  - Decommissioning
  - Plant Life Extension
  - Probabilistic Safety Analysis
  - Regulatory Issues
  - Safety Analysis
  - Technical Specifications
- **Maintenance Processes**
  - **All Maintenance Processes Topics**
  - General
  - Foreign Material Exclusion
  - Predictive
  - Preventive
- **Operations**
  - **All Operations Topics**
  - General
  - Operations Management
  - Reactivity Management
  - Refueling Activities
  - Surveillance Testing
- **Operating Experience Programs**
  - General
- **Planning And Scheduling**
  - General
  - OnLine
  - Outage
- **Plant Event Reports**
  - **All OE Topics**
  - Operating Experience Reports (OEs)
  - Operations And Maintenance Reminders (O&MRs)
  - Significant Event Notifications (SENs)
  - Significant Event Reports (SERs)
  - Significant Operating Experience Reports (SOERs)
- **Procurement**
  - General
  - Commercial Dedication
  - Parts

Table A-4 INPO Topic Areas Searched

- **Radiation Protection**
  - **All Radiation Protection Topics**
  - General
  - Contamination Control
  - Dosimetry
  - Instrumentation
  - Personnel Exposure
  - Radiological Effluents
- **Radioactive Waste**
  - General
- **Records Management**
  - General
  - Document Retrieval
  - Procedure Management
- **Regulatory Reports**
  - **All Regulatory Reports Topics**
  - Bulletins
  - Generic Letters
  - Information Notices
  - Morning Reports
  - Regulatory Issue Summaries
- **Security**
  - **All Security Topics**
  - General
  - Fitness For Duty
- **Self Assessment**
  - General
  - Quality Assurance Archive (Read-only)
  - Benchmarking
- **Training**
  - **All Training Topics**
  - General
  - Contractor Training
  - Control Room Operator
  - General Employee Training
  - Plant Personnel
- **Archived Topics**
  - Design Engineering & Configuration Management
  - Exchange of Miscellaneous Information
  - Meeting Announcements & Summaries
  - NRC Daily Plant Status Report
  - Regulatory Information Transmittal
  - Technical Support Information Exchange

Table A-5 Summary of Industry Design Prediction and As-Built Core Problems									
#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Detection Method
1	4.1.2.1	Dominion	-	1984	Millstone Unit 2	CE	During Cycle 6 CEA Group Worth tests, Group 4 measured worth (0.170%Δρ or 15.7%) failed the acceptance criteria for individual group worth (larger of ±0.1% Δρ or ±15%). Millstone consulted with Westinghouse and reviewed safety analysis. It was concluded that Group 4, which was in fuel assemblies discharged at EOC1 (four cycles earlier), had an error in the predicted worth since predictions did not include effect of Pu decay. The methodology was revised to properly account for PU decay.	CEA Worth
2	4.1.2.1	W-Pittsburgh	-	1990 (approx.) and Earlier	Vogtle and Other Plants	W	Misprediction of rod worths in new fuel assemblies. Reference banks (highest worth) would typically be located over new fuel locations. Measurements of the bank would have a bias error, and at times exceed the 10% review criterion. Total bank worth would be acceptable. The core design code and/or model inputs were changed to resolve the discrepancy.	CEA Worth
3	4.1.2.1	W-Pittsburgh	-	1996	Wolf Creek and Other Plants	W	Misprediction of assembly powers in periphery in W 4-Loop plants. Specifically, the first application of DRWM at Wolf Creek was troubled with a few issues. The measured worths showed a marked bias (in-out) that was supported by Rod Swap data. Westinghouse concluded that prediction obtained from the spatial calculations were incorrect due to in-out bias associated with treatment of the baffle reflector. Previous predictions for In-Out fuel management were not a problem. However implementation of Low Leakage fuel management resulted in larger misprediction of group worths. Core engineering revamped their process for the baffle reflector constants, which provided a marked improvement in the predicted results.	CEA Worth
4	4.1.2.2	8205140500	LER 82-006-01T-0	05/07/82	Prairie Island Nuclear Station, Unit 1	W	LER 82-006-01T-0. On 820423 during surveillance test Sp 1104 measured reactor coolant boron concentration about 120 ppm (1% Δρ) higher than the original predicted value. Caused by miscalculation of predicted worth and/or depletion rate of gadolinium in fresh fuel assemblies. Analysis being performed to monitor disagreement. Results of analysis indicate that the disagreement should start to converge at about 6200 MWD/MTU and disappear at EOC. **Note that the initial correction for the rundown prediction was only 18 ppm. Therefore, there does not appear to have been any significant deviation in the BOC HZP CBC measurement.**	Boron Rundown
5	4.1.2.3						None Identified	
6	4.1.2.4						None Identified	
7	4.2.2.1						None Identified	
8	4.2.2.2						None Identified	
9	4.2.2.3						None Identified	
10	4.2.2.4	8002040556	RO 80-17	01/23/80	North Anna Power Station, Unit 1	W	RO 80-17: On 800123 Zero power flux tilt irregularities encountered following refueling. Specifically, the measured $F^N_{\Delta H}$ exceeded the established design values tolerance at hot-zero-power with D Bank inserted and with Banks C and D at the insertion limits. Caused by quadrant power	Incore Flux Symmetry

Table A-5 Summary of Industry Design Prediction and As-Built Core Problems								
#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Detection Method
							tilt identified by incore flux map analyses. Based on there measurement results, it was determined that the control rod insertion limits in Tech. Specs. were no longer appropriate. A safety evaluation was performed that justifies power operation with revised insertion limits. Flux map taken at 3% power at the revised insertion limit and the measured $F^N_{\Delta H}$ was determined to be acceptable. The magnitude of the anomaly was not identified. However, it likely was less than the uncertainty on Power Distribution since the Tech. Spec. limit on tilt apparently was met. This is classified as a Power Distribution Error since it resulted from failure to account for non-symmetric isotopic distribution as a result of a quadrant power tilt that was present during the previous cycle. **Detected by Incore Flux Symmetry not the ITC, MTC Surveillance or CEA worth measurement results.**	
11	4.2.2.5	7811170184	LER	11/13/78	North Anna Power Station, Unit 1	W	LER: Moderator Temp Coefficient (MTC) More Positive Than Limits In Tech Spec 3.1.1.4. The measured HZP ARO MTC was determined to be +1.126 pcm/°F versus the limit of <0.0 pcm/°F. Based on this measurement result, temporary rod withdrawal limits were determined to be needed for the first 3,000 MWD/MTU. The report does not state that failure to meet the Tech. Spec. was result of a calculation error in CBC or MTC.	MTC Surveillance
12	4.2.2.5	8002050538	RO	01/25/80	Donald C. Cook Nuclear Power Plant, Unit 2	W	RO: During Zero Power Physics Testing & W-All Rods Withdrawn the Moderator Temp Coefficient was measured to be +0.71 pcm/°F. The Tech. Spec. most positive limit was 0.0 pcm/°F. Rods to be repositioned so boron concentration is below 1480 PPM. The MTC will be negative whenever the reactor is critical after 800 MWD/MT.	MTC Surveillance
13	4.2.2.5	8007250499 * 8008260543	SR * SR	07/15/80 * 08/18/80	Sequoyah Nuclear Plant, Unit 1	W	Special Rept: Moderator Temp Coefficient (MTC) More Positive Than Limits In Tech. Spec. 3.1.1.3. The measured HZP ARO MTC was determined to be $+0.85\times10^{-5}$ delta k/k/°F versus the limit of <0.0 delta k/k/°F. Based on this measurement result, temporary rod withdrawal limits were determined. The report does not state that failure to meet the Tech. Spec. was result of a calculation error in CBC or MTC.	MTC Surveillance
14	4.2.2.5	8112140150	SR	08/25/81	William B. McGuire Nuclear Station, Unit 1	W	Special Rept: Moderator Temp Coefficient (MTC) More Positive Than Limits In Tech. Spec. 3.1.1.3. The measured HZP ARO MTC was determined to be +1.44 pcm/°F versus the limit of <0.0 pcm/°F. Based on this measurement result, temporary rod withdrawal limits were determined to be needed for the first 40 EFPDs. The report does not state that failure to meet the Tech. Spec. was result of a calculation error in CBC or MTC.	MTC Surveillance
15	4.2.2.5	8111200764	SR	11/13/81	Sequoyah Nuclear Plant, Unit 2	W	Special Rept: Moderator Temp Coefficient (MTC) More Positive Than Limits In Tech Spec 3.1.1.3. The measured HZP ARO MTC was determined to be $+0\ 65\times10^{-5}$ delta k/k/°F versus the limit of <0.0 delta k/k/°F. Based on this measurement result, temporary rod withdrawal limits were determined. The report does not state that failure to meet the Tech. Spec. was result of a calculation error in CBC or MTC.	MTC Surveillance

Table A-5 Summary of Industry Design Prediction and As-Built Core Problems

#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Detection Method
16	4.2.2.5	8204080144	RO	03/12/82	Joseph M. Farley Nuclear Plant, Unit 1	W	RO: During Low Power Physics Testing for Cycle 4 Moderator Temp Coefficient (MTC) in Hot Zero Power Beginning of Cycle Condition was measured greater than required by Tech. Spec. The MTC Tech. Spec. positive limit was 0.0 pcm/°F while the measured MTC was 0.78 pcm/°F. Administrative limit on rod insertion was established for Cycle 4 to assure the LCO was not violated with an assumed uncertainty of 0.1 pcm/°F. The report does not state that failure to meet the Tech. Spec. was result of a calculation error in CBC or MTC.	MTC Surveillance
17	4.2.2.5	8207060017	RPT	03/29/82	Sequoyah Nuclear Plant, Unit 2	W	Inspection Report: Moderator Temp Coefficient (MTC) More Positive Than Limits In Tech Spec 3.1.1.3. The measured HZP ARO MTC was determined to be positive versus the limit of <0.0 delta k/k/°F. Based on this measurement result, temporary rod withdrawal limits were determined. The report notes that there was a failure to maintain the control rod withdrawal limits established by the action requirements of Tech. Spec. 3.1.1.3. The report does not state that failure to meet the Tech. Spec. was result of a calculation error in CBC or MTC. There had been no training for the licensed operators regarding the control rod withdrawal limits requirements.	MTC Surveillance
18	4.2.2.5	8303140583	LER	03/01/83	Salem Nuclear Generating Station, Unit 1	W	LER: Moderator Temp Coefficient (MTC) More Positive Than Limits In Tech. Spec. The measured HZP ARO MTC was determined to be slightly positive versus the limit of <0.0 delta k/k/°F. Based on this measurement result, temporary rod withdrawal limits were determined. The report does not state that failure to meet the Tech. Spec. was result of a calculation error in CBC or MTC. The report says that the measured result was expected.	MTC Surveillance
19	4.2.2.5	8309230389	LER 83-039-031-0	09/07/83	William B. Mcguire Nuclear Station, Unit 2	W	LER 83-039-031-0: On 830809 During unit startup rod withdrawal limits established to prevent exceeding positive Moderator Temp Coefficient. Caused by control operator failure to follow startup procedure. A conservative estimate of MTC showed that the value was less position than the Tech. Spec. Limit, 0.0. The subject rod withdrawal limits were transmitted to the NRC via a Tech. Spec. 3.1.1.3a/6.9.2 Special Report submitted 06/02/83. The reason why rod withdrawal limits were established was not identified but was likely associated with a higher than CBC. It is assumed that the difference was less than the uncertainty. W-830907 Ltr.	MTC Surveillance
20	4.2.2.5	8407130308	LER	07/10/84	Turkey Point Plant, Unit 4	W	LER: Moderator Temp Coefficient (MTC) More Positive Than Limits In Tech Spec 3.1.2 at 70% power. The measured ARO MTC at 70% power was determined to be slightly positive versus the limit of <0.0 delta k/k/°F. The problem arose when boron was added to control power, compensating for the reduced xenon concentration, resulting in the MTC Tech. Spec. violation. Based on this measurement result, rod position limits, required xenon build-up, and boron concentration were determined that would produce a negative MTC. The report does not state that failure to meet the Tech. Spec. was result of a calculation error in CBC or MTC.	MTC Surveillance

#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Detection Method
Table A-5 Summary of Industry Design Prediction and As-Built Core Problems								
21	4.2.2.5	8509160103	LER	10/04/84	North Anna Power Station, Unit 1	W	LER: Moderator Temp Coefficient (MTC) More Positive Than Limits In Tech Spec. Measured HZP ARO MTC was determined to be $+0.67\times10^{-5}$ delta k/k/°F versus the limit of <0.0 delta k/k/°F. Based on this measurement result, temporary rod withdrawal limits were determined. The report does not state that failure to meet the Tech. Spec. was result of a calculation error in CBC or MTC.	MTC Surveillance
22	4.2.2.5	8502210304	SR	01/28/85	Catawba Nuclear Station, Unit 1	W	Special Rept: Moderator Temp Coefficient (MTC) More Positive Than Limits In Tech Spec 3.1.1.3 when applying measurement uncertainty. The measured HZP ARO MTC was determined to be -0.02 pcm/°F. Based on this measurement result and the extra conservatism that was desired, temporary rod withdrawal limits were determined to be needed for 85 EFPDs. The report does not state that failure to meet the Tech. Spec. was result of a calculation error in CBC or MTC.	MTC Surveillance
23	4.2.2.5	8507010629	SR	06/10/85	Wolf Creek Generating Station	W	Special Rept: Moderator Temp Coefficient (MTC) More Positive Than Limits In Tech Spec 3.1.1.3. The measured HZP ARO MTC was determined to be $+1.03\times10^{-5}$ delta k/k/°F versus the limit of <0.0 delta k/k/°F. Based on this measurement result, temporary rod withdrawal limits were determined. The report does not state that failure to meet the Tech. Spec. was result of a calculation error in CBC or MTC.	MTC Surveillance
24	4.2.2.5	8512310262	SR	12/20/85	Virgil C. Summer Nuclear Station, Unit 1	W	Special Rept SPR 85-018: On 851216 during Zero Power Physics Testing under surveillance test procedure Stp-210.002 positive moderator temp coefficient for reload core of Cycle 3 was identified. Although the abstract from the NRC data base stated that the discrepancy was Part 21 related the document does not mention anything related to a Part 21. The MTC Tech. Spec. positive limit was 0 0 pcm/°F while the measured MTC was 0.32 pcm/°F. Administrative limit on rod insertion was established for Cycle 3 to assure the LCO was not violated. The report does not state that failure to meet the Tech. Spec. was result of a calculation error in CBC or MTC.	MTC Surveillance
25	4.2.2 5	8603070509	SR	02/03/86	Millstone Nuclear Power Station, Unit 3	W	Special Rept: Positive Moderator Temp Coefficient (MTC) measured above Tech Spec Limits. The measured HZP ARO MTC was determined to be +0.92 pcm/°F versus a Tech. Spec. limit of 0.0 pcm/°F. Rod withdrawal & boron concentration limits established to prevent Moderator Temp Coefficient from becoming positive.	MTC Surveillance
26	4.2.2.5	8605090148	SR	04/17/86	Callaway Plant, Unit 1	W	Special Rept: Moderator Temp Coefficient (MTC) More Positive Than Limits In Tech. Spec. 3.1.1.3.a. The measured HZP ARO MTC was determined to be +0.01 pcm/°F versus the limit of <0.0 pcm/°F. Based on this measurement result, temporary rod withdrawal limits were determined to be needed for the first 4,000 MWD/MTU. The report does not state that failure to meet the Tech. Spec. was result of a calculation error in CBC or MTC.	MTC Surveillance
27	4.2.2.5	8605280096 *	LER	05/12/86	Oconee Nuclear	B&W	LER: HFP most negative Moderator Temp Coefficient (MTC) was more	MTC

#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Detection Method
		8612080407	* SE	* 11/26/86	Station, Unit 1		negative than limits in Technical Specification. The measured EOC ARO MTC was determined to be -3.36×10^{-4} $\Delta\rho/°F$ versus the limit of $>-3.0 \times 10^{-4}$ $\Delta\rho/°F$. The report states that failure to meet the Tech. Spec. was not result of a calculation error in CBC or MTC. The root cause was an inadequate vendor database of measured MTC data. Hence, a revision to the bias applied to EOC MTC will occur for future reloads.	Surveillance
28	4.2.2.5	8704030503	LTR	03/27/87	Alvin W. Vogtle Nuclear Plant, Unit 1	W	Letter: Moderator Temp Coefficient (MTC) More Positive Than Limits In Tech. Spec. 3.1.1.3. The measured HZP ARO MTC was determined to be +0.89 pcm/°F versus the limit of <0.0 pcm/°F. Based on this measurement result, temporary rod withdrawal limits were determined to be needed up to a 30% power level for the first 6,000 MWD/MTU. The report does not state that failure to meet the Tech. Spec. was result of a calculation error in CBC or MTC.	MTC Surveillance
29	4.2.2.5	8704140140	LTR	04/09/87	Waterford Generating Station, Unit 3	CE	Letter: Positive Moderator Temp Coefficient (MTC) measured above Tech. Spec. Limits. The measured HZP ARO MTC was determined to be $+0.505 \times 10^{-4}$ delta k/k/°F versus a Tech. Spec. limit of $+0.5 \times 10^{-4}$ delta k/k/°F. Control rods were inserted in order to reduce the Moderator Temp Coefficient. The letter does not state that failure to meet the Tech. Spec. was result of a calculation error in CBC or MTC.	MTC Surveillance
30	4.2.2.5	8707200441	SR	07/09/87	North Anna Power Station Unit 1	W	Special Rept: Moderator Temp Coefficient (MTC) more positive than limits in Tech. Spec. The measured HZP ARO MTC was determined to be +1.24 pcm/°F versus the limit of <0.0 delta k/k/°F. Conservative calculations demonstrated the MTC will be negative at 70% power. Therefore, control rod withdrawal limits are not necessary since these values are within Tech. Spec. limits. The report does not state that failure to meet the Tech. Spec. was result of a calculation error in CBC or MTC.	MTC Surveillance
31	4.2.2.5	8711190307	SR	11/16/87	Callaway Plant, Unit 1	W	Special Rept: Positive Moderator Temp Coefficient (MTC) measured above Tech. Spec. Limits. The measured HZP ARO MTC was determined to be +0.372 pcm/°F versus a Tech. Spec. limit of <0.0 pcm/°F. Rod withdrawal and boron concentration limits established to prevent Moderator Temp Coefficient from becoming positive. The report does not state that failure to meet the Tech. Spec. was result of a calculation error in CBC or MTC.	MTC Surveillance
32	4.2.2.5	8811220004	SR	11/17/88	Sequoyah Nuclear Plant, Unit 1	W	Special Rept: Moderator Temp Coefficient (MTC) More Positive Than Limits In Tech. Spec. 3.1.1.3. The measured HZP ARO MTC was determined to be +0.64 pcm/°F versus the limit of <0.0 pcm/°F. Based on this measurement result, temporary rod withdrawal limits were determined to be needed for the first 2,700 MWD/MTU. It was known prior to measurement that a violation would occur. Due to earlier shutdown of previous cycle, which caused the presence of additional excess reactivity, a positive MTC was predicted.	MTC Surveillance
33	4.2.2.5	8901180412	SR	01/12/89	Wolf Creek	W	Special Rept: Moderator Temp Coefficient (MTC) More Positive Than	MTC

Table A-5 Summary of Industry Design Prediction and As-Built Core Problems								
#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Detection Method
					Generating Station		Limits In Tech. Spec. 3.1.1.3. The measured HZP ARO MTC was determined to be +0.65 pcm/°F versus the limit of <0.0 pcm/°F. Based on this measurement result, temporary rod withdrawal limits were determined to be needed for the first 4,851 MWD/MTU. The report does not state that failure to meet the Tech. Spec. was result of a calculation error in CBC or MTC.	Surveillance
34	4.2.2.5	8903030521	SR	02/15/89	Catawba Nuclear Station, Unit 1	W	Special Rept: Moderator Temp Coefficient (MTC) More Positive Than Limits In Tech. Spec. 3.1.1.3. The measured HZP ARO MTC was determined to be +4.61 pcm/°F. Based on this measurement result, temporary rod withdrawal limits were determined to be needed at or above the 90% power level for 34 EFPDs. The report does not state that failure to meet the Tech. Spec. was result of a calculation error in CBC or MTC.	MTC Surveillance
35	4.2.2.5	8903230497	RO	03/10/89	Byron Station, Unit 2	W	RO: During Zero Power Physics Testing and with All Rods Withdrawn, the Moderator Temp Coefficient was measured to be +0.0435 pcm/°F. The Tech. Spec. most positive limit was 0.0 pcm/°F. Based on this measurement result, temporary rod withdrawal limits were determined to be needed for the first 1,000 MWD/MTU. The report does not state that failure to meet the Tech. Spec. was result of a calculation error in CBC or MTC.	MTC Surveillance
36	4.2.2.5	8904170097	SR	04/06/89	Alvin W. Vogtle Nuclear Plant, Unit 2	W	Special Rept: Moderator Temp Coefficient (MTC) More Positive Than Limits In Tech. Spec. 3.1.1.3. The measured HZP ARO MTC was determined to be +0.9 pcm/°F versus the limit of <0.0 pcm/°F. Based on this measurement result, temporary rod withdrawal limits were determined to be needed up to a 30% power level for the first 6,000 MWD/MTU. The report does not state that failure to meet the Tech. Spec. was result of a calculation error in CBC or MTC.	MTC Surveillance
37	4.2.2.5	9004230549	SR	04/16/90	Comanche Peak Steam Electric Station, Unit 1	W	On 900404 positive Moderator Temp Coefficient (MTC) noted during Low Power Physics Testing due to high boron concentration in moderator. The measured HZP ARO MTC was determined to be +1 pcm/°F versus a Tech. Spec. limit of 0 pcm/°F. Rod withdrawal limits were imposed per Tech. Spec. 3.1.1.3a Action A.1 for 4450 MWD/MTU. However, inspection of attached table implies that CBC only needs to be reduced by 140 ppm (~3 pcm/°F) to remove rod withdrawal limits. Impact appears to be much smaller than 4450 MWD/MTU. However, the 4450 MWD/MTU likely corresponds to HZP No Xexon condition. The report does not state that failure to meet the Tech. Spec. was result of a calculation error in CBC or MTC.	MTC Surveillance
38	4.2.2.5	9005080381	SR	04/30/90	Catawba Nuclear Station, Unit 1	W	Special Rept Re Moderator Temp Coefficient (MTC) More Positive Than Limits In Tech. Spec. 3.1.1.3 Figure 3.1-0. The measured HZP ARO MTC was determined to be +3.3 pcm/°F. Based on this measurement result temporary rod withdrawal limits were determined to be needed at 95 and higher power levels for 4 EFPDs or the HFP boron concentration is less than	MTC Surveillance

#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Detection Method
Table A-5 Summary of Industry Design Prediction and As-Built Core Problems								
							1363 ppm boron. The report does not state that failure to meet the Tech. Spec. was result of a calculation error in CBC or MTC.	
39	4.2.2.5	9010040161	SR	09/21/90	Catawba Nuclear Station, Unit 2	W	Special Rept: Moderator Temp Coefficient (MTC) More Positive Than Limits In Tech. Spec. 3.1.1.3 Figure 3.1-0. The measured HZP ARO MTC was determined to be +5.11 pcm/°F. Based on this measurement result, temporary rod withdrawal limits were determined to be needed at 0 and between 85 and 100% power levels for 4 EFPDs or the HFP boron concentration is less than 1438 ppm. The report does not state that failure to meet the Tech. Spec. was result of a calculation error in CBC or MTC.	MTC Surveillance
40	4.2.2.5	9104180168	SR	04/12/91	South Texas Project, Unit 1	W	Special Rept: on 910402 during low power physics testing positive moderator temp coefficient measured high critical boron concentration in reactor coolant system at beginning of 18 month fuel cycle caused MTC to be positive. The MTC Tech. Spec. positive limit was 0.0 pcm/°F while the measured MTC was 0.4 pcm/°F. Administrative limit on rod insertion was established to assure the LCO was not violated. The higher boron concentration than expected likely contributed to the failure to meet Tech. Spec. LCO. Furthermore, the report does not state that failure to meet the Tech. Spec. was result of a calculation error in CBC or MTC. Note that STAR's MTC Alternate Surveillance corrects the predicted MTC for measured CBCs higher than predicted CBC. Therefore, application of STAR would likely have accounted for the discrepancy in CBC. W-910412 ltr.	MTC Surveillance
41	4.2.2.6	INPO - SER 11-95	-	1988	Plant X, Units 1 and 2	CE	The fuel vendor's reactivity analysis assumed the minimum cold shutdown boron concentration was 4,000 ppm. However, the technical specifications only directed a minimum boron concentration of 2,150 ppm or the boron concentration corresponding to Keff less than or equal to 0.95, whichever was greater. On at least two occasions since initial plant startup, the shutdown boron concentration was reduced below the 4,000 ppm value assumed in the initial safety analysis, but was maintained above the 2,150 ppm value in the technical specifications. In 1988, boron concentration decreased to less than 4,000 ppm for an extended time; however, subsequent review of these plant conditions confirmed that safety limits were maintained even though boron concentration was less than the amount assumed in the safety analysis.	Core Design QA
42	4.2.2.7	ETP-02-310	-	2001	Westinghouse Columbia SC Fuel Manufacturing Plant	W	The customer had specified a unique top nozzle identification to be followed with a suffix, to indicate the enrichment level of the fuel assembly. The top nozzle unique number had been specified, however the suffix was incorrectly specified. The nozzles were fabricated as listed on the bill of materials. The nozzles were determined to be usable and the core loading plan and associated documentation were modified to reflect the as-built top nozzle identifications.	Fuel Fabrication QA

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Table A-5 Summary of Industry Design Prediction and As-Built Core Problems								
43	4.2.2.7	ETP-02-310	-	2001	Westinghouse Columbia SC Fuel Manufacturing Plant	W	In this instance the loading pattern of the WABA core component has been incorrectly specified in the bill of material. Part of the product had been fabricated and shipped to the customer. The WABA pattern was specified with the correct number of rods. However, the wrong loading pattern was used. All the misloaded assemblies were re-configured with the correct loading pattern.	Receipt Inspection
44	4.2.2.7	ETP-02-310	-	1999	Westinghouse Columbia SC Fuel Manufacturing Plant	W	In this instance the loading pattern of the WABA core component has been incorrectly specified in the bill of material. None of the product had been shipped. The WABA pattern was specified with the correct number of rods. However, the wrong loading pattern was used. All the misloaded assemblies were re-configured with the correct loading pattern.	Fuel Fabrication QA
45	4.2.2.7	PAE-JJB-02-004	-	1999-2002	Westinghouse Columbia SC Fuel Manufacturing Plant	PWR	Occasionally fuel rods are rejected by the active gamma scanner. The reject could be due to fuel or burnable absorber pellet misloads, undersized diameter pellets, pellet density deviations, or false rejects due to conservative scanner settings. The annual reject rate for ADU rods is less the 0.15% (less than 500 rods/year) and less than 0.31% (less than 400 rods/year) for ZrB2 IFBA rods.	Fuel Fabrication QA
46	4.2.2.7	PAE-JJB-02-004	-	1999-2002	Westinghouse Columbia SC Fuel Manufacturing Plant	W	During processing, one ZrB2 IFBA pellet coater run experiences a number of equipment failures causing many startups and shutdowns. This occurred through many shifts resulting in confusion about the run's total coating time. As a result the actual B10 coating was off by a factor of 10. The overcoated pellets were subsequently loaded into approximately 150 rods. The gamma scanner subsequently rejected all the rods containing the overcoated pellets.	Fuel Fabrication QA
47	4.2.2.7	8207010407	LD-82-062	06/23/82	Plants using CE fabricated fuel	CE	Part 21 Report addressing final confirmatory analysis of burnable boron pellets in C-E supplied fuel not properly generated. CE concluded that the extreme case of either no boron being present or excess boron being present would be clearly noticeable as a result of normal testing. With respect to normal plant operation, it is CE's opinion that even if there are local variations, they would not cause clad damage or elevate the local heat flux sufficiently to cause thermal hydraulic problems. Customers should conduct re-review physics test. No substantial safety hazard.	Fuel Fabrication QA
48	4.2.2.7	8311030013	NS-EPR-2843	10/25/83	Sequoyah Unit 2	W	Letter informing NRC that 19 fuel rods containing pellets of an improper diameter and enrichment found to be loaded into Fuel Assembly P39 does not represent a substantial safety hazard but does represent a failure to comply with 10CFR50.46. Nineteen fuel rods containing pellets of an improper diameter and enrichment were found to be loaded into Fuel Assembly P39 that had been delivered to TVA. Specifically, it amounted to approximately a 24 inch length of pellets beginning 12 inch from the bottom in each of 19 fuel rods. The errant pellets were of a smaller diameter and a lower enrichment than the proper pellets. The incident was discovered on 08/24/83 through a routine overcheck evaluation of inspection records and	Fuel Fabrication QA

#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Detection Method
							documentation by a QC engineer. The incident was discovered and corrected prior to loading of the fuel in the reactor. An analysis was conducted to assess the impact on reactor performance had the assembly operated in the core. The evaluation showed that center temperature melting point would have been exceeded based on current methods. Similarly, the PCT as contained in the current ECCS analysis would also exceed the Appendix K limit of 2200 °F. However, overall coolability of the assembly and the core would not be affected. An investigation was also performed to determine whether or not this event represented an isolated incident. Specifically, a detailed review of manufacturing records for the time period which included the manufacturing of Sequoyah 2 reload and included several thousands lots (25 rods per lot) of fuel rods. No evidence of mislaoded pellets due to tray mixup was noted. This information, coupled with an assessment of the manufacturing process, led to the conclusion that this was an isolated event. Westinghouse has taken corrective actions which result in additional controls to prevent recurrence.	
49	4.2.2.7	PAE-JJB-02-004	-	08/2000	Westinghouse Columbia SC Fuel Manufacturing Plant	PWR	A number (20) of fuel rods were rejected by the gamma scanner for suspected enrichment deviations. Destructive evaluation of rods from this population revealed that the reject condition was actually caused by strings of undersized pellets. All fuel rods were ground on the same pellet line. The cause was determined to be over grinding due to pellet backup into the grinder. Subsequent analysis of the impact of undersized pellets that may have been loaded but not rejected (i.e., beyond the resolution of the scanner) indicated no significant impact on any operating parameter or the safety analysis.	Fuel Fabrication QA
50	4.2.2.7	9302220351 • 9307220073	LER 93-001 LER 93-001-01	02/16/93 07/15/93	Turkey Point Plant, Unit 3	W	On 01/14/93 while Units 3 & 4 were at 100% power Westinghouse notified FPL that Wet Annular Burnable Absorber Assemblies (WABA) were not mfg. per design specs. This was caused by failure to translate design requirement to drawing absorber design modified. Specifically, the WABA Assemblies were not redesigned as required to accommodate a longer fuel rod lower end cap on the last batches of fuel supplied for each unit. The anomaly was initially detected during review of the Unit 3 flux map at 100% power. Specifically, the local peaking factor (Fq) was higher and axial power distribution more topped peaked than expected. Further investigation by Westinghouse lead to identification of the WABA offset anomaly. Westinghouse evaluated the impact of the discrepancy on Unit 4 and concluded that all statements and conclusions in the original RSE remain valid for the entire operating cycle. Somewhat similar conclusions were reached for Unit 3. The results of FAC analysis performed by Westinghouse for Unit 3 concluded that Fq*K(z) could potentially have been violated by up to ~11% based on the WABA offset if the unit was operated in a load-	Incore Power Distribution

Table A-5 Summary of Industry Design Prediction and As-Built Core Problems								
#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Detection Method
							follow mode with the worst combination of axial power shape and rod position. FPL concluded that the offset WABA rods were of minor safety significance. W-930713 Ltr.	
51	4.2.2.7	9312020238 * INPO SER 2-94	Part 21 * PI	11/18/93 * 11/16/93	H.B. Robinson Plant, Unit 2	W	9312020238 & INPO SER 2-94: The results of 30% flux map following startup from Refueling Outage 15, performed on November 15, 1993, indicated peaking factors and radial tilts greater than expected. Subsequent investigation determined that six new fuel assemblies contain burnable poison fuel pins located in the wrong position in the assemblies. This misposition of the burnable poison fuel pins accounts for the power distribution anomaly differences.	Incore Power Distribution
52		9312020249	Part 21	11/19/93	H.B. Robinson Plant, Unit 2		9312020249: As-Loaded Gadolinia pin locations did not comply with design specifications. Pin locations were in incorrect quadrant of six fuel assemblies in reload batch ROB-13 supplied by Siemens Power Corporation. The defect resulted from inadequate communication of design spec info from fuel designer to mfg. within vendor. This condition may have result in thermal or safety limit violations which may not be detected by power distribution monitoring systems. The defect was detected during Power Distribution tests due to higher peaking than anticipated. Initial corrective action was to exchange the location of the defective assemblies to achieve as-designed conditions.	
53		9401100075	LER 93-020-00	12/31/93	H.B. Robinson Plant, Unit 2		LER 93-020-00: On 931119 TS violation occurred at 30% power due to exceeding F-Delta-H Hot Channel Factor in COLR by 0.36%. The reason that the limit was exceeded was that fuel rods in certain assemblies in the core were not loaded as designed which had the effect of accentuating power peaking in the core, causing the thermal limit to be exceeded. The data from the first 30% power flux map indicated a power tilt of 2.8%, which exceeded the acceptance criteria of less than 2%. The map also indicated that the peaking factors were higher than expected but less than Tech. Specs. A comparison of "predicted" to "measured" relative powers indicted higher than predicted powers (approximately 14%) in the core areas surrounding certain fuel assemblies (later to be determined to have been misfabricated) and also indicated lower than expected relative powers (approximately 10%) in other localized areas of the core. A second flux map at 30% power yielded similar results. Analysis confirmed that this event had no impact on plant safety. Furthermore, continued operation of the core, as misloaded, could not have created a safety hazard. Specifically, analysis indicated that if the core had operated at full power, this would have resulted in a "true" F-Delta-H of 1.797 vs 1.70 (+5.7%) but there would have been no violation of the core safety limits since the cycle specific undetected bundle misleading event was analyzed to yield a maximum F-Delta-H of 1.82. The misleading event was bounded by the static RCCA Misalignment Analysis (F-Delta-H of 1.94). Caused by management deficiency. Six misloaded fuel assemblies	

Table A-5 Summary of Industry Design Prediction and As-Built Core Problems

#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Detection Method
							repositioned in core to compensate for power anomaly. W-931231 Ltr.	
54		9401040023	Final Part 21	12/31/93	H.B. Robinson Plant, Unit 2		Final Part 21 Rept concluding that deviation in nuclear fuel assemblies mfg by Siemens Power Corp could not have created Substantial Safety Hazard as defined by 10CFR21 & that subject 10CFR21 defect did not exist. As-Loaded Gadolinia pin locations did not comply with design specifications. Pin locations were in incorrect quadrant of six fuel assemblies in reload batch ROB-13 supplied by Siemens Power Corporation. The defect resulted during fabrication due to an incorrect manual transposition of bundle design information, by a Siemens employee, onto marked-up Bundle Load Maps. This information is used as the input for the Siemens automatic fuel assembly manufacturing system (BADL - Bundle Assembly Data Logger). The fuel mis-fabrication was identified through normal plant testing and analysis of inappropriate flux map results following approximately 3 days of power ascension. The defect was detected at 30% power due to higher peaking than anticipated. Initial corrective action was to exchange the location of the defective assemblies to achieve as-designed conditions. Analysis confirmed that no substantial safety hazard existed and that continued operation of the core could not have created a substantial safety hazard.	
55	4.2.2.7	ETP-02-310	-	2001	Westinghouse Columbia SC Fuel Manufacturing Plant	W	In this instance, the IFBA pattern had been incorrectly specified in the bill of materials. The fuel assemblies were specified with the correct number of IFBAs, but with the wrong configurations. The cause was related to errors in the SAP Bill of Materials for these components. All the fuel assemblies had been fabricated and shipped to the customer. Physics modeling of the as built configurations showed minimal impact on the core parameters. However it was necessary to update the incore detector constants in order not to introduce error into the power distribution measurement. Note that this event was identified as a result of Westinghouse's Corrective Action Program (CAP).	Fuel Fabrication QA
56	4.2.2.7	8602260287	DR 86-01	02/11/86	Braidwood Station, Unit 1	W	Final Deficiency Rept 86-01 addressing misorientation of burnable poison rods A6P2D in Westinghouse fuel assembly C54S initially reported on 01/14/86. The FSAR discusses the inadvertent loading and operation of a fuel assembly in an improper position. Included is the loading of one or more fuel assemblies into a new core without their required burnable poison rods. The moveable incore detector system is capable of revealing any assembly enrichment error or loading error which causes power shapes to be peaked in excess of the design value. The FSAR concludes that the resulting power distribution effects will either be readily detected by the moveable incore detector system or will cause a sufficiently small perturbation to be acceptable within the uncertainties allowed between nominal and design power shape. Westinghouse also analyzed the effects on peaking factor	Receipt Inspection

#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Detection Method
							margin and flux perturbation if a fuel assembly with a misoriented burnable poison is moved toward the core center after residing on the periphery during cycle one. Westinghouse concluded that there would have been no violation of peaking factor for cycle one and all subsequent cycles. Assembly A6P2D was returned to mfg for repair. Vendor will change mfg & inspection practices.	
57	4.2.2.7	INPO - OE9646	IO1	01/15/99	Plant X	W	On January 15, 1999, with Plant X Units 1 and 2 at 100 percent power, reactor engineering personnel discovered, during new fuel receipt inspection, that the burnable poison rodlets (BPR) in two fuel assemblies did not agree with the core loading plan diagram (CLPD). The fuel vendor, Westinghouse, was notified that the configuration of the eight outer rodlets on each of the two 12-rodlet BPR assemblies received were not in accordance with the vendor's CLPD. It appears that the deficient BPR assemblies received were manufactured in accordance with the incorrect drawing reference number cited on the CLPD. The vendor's preliminary root cause investigation indicates a possible problem with the ANCHOR code software that automatically provides the drawing reference numbers on the CLPD.	Receipt Inspection
58	4.2.2.7	INPO - OE14297	IO1	6/11/02	Plant X, Units 1&2	W	On June 11, 2002, with Plant X Units 1 at 100 percent power, core data indicated a larger axial offset deviation (AOD) than expected following a refueling outage. Unit 2 also has experienced this anomaly following its most recent refueling outage. Currently, Unit 1 is operating with all rods out. The core axial offset is at minus 11 percent and trending slightly more negative. The cause of the larger deviations have not yet been confirmed. However, it is likely associated with deposition of ZB_2 during rod loading.	Incore Power Distribution
59	4.2.2.8	INPO - OE8730	IO1	11/06/97	Plant X Unit 2	W	Plant X Unit 2 identified an inconsistent core flux distribution while at 30 percent power following startup from a refueling outage. In consultation with the fuel vendor, the station determined that 16 new fuel assemblies, with gadolina burnable poison rods, had been loaded into core locations different than planned. The burnable poison rods have two different patterns (eight assemblies each) and the loading procedures, supplied by the vendor, had the patterns reversed.	Incore Power Distribution
60	4.2.2.8	7905150468	LER 79-014-01T-0	05/11/79	Prairie Island Nuclear Station, Unit 1	W	LER 79-014-01T-0: On 790427: Informed by Exxon of error in core loading pattern. Specifically Gadolinia-Bearing assemblies not properly located. Exxon submitted a revised loading pattern to NSP which exchanged the mislocated gadolinia-bearing assemblies with other new Exxon assemblies. The loading pattern was checked independently against as-built loadings and verified to be correct and consistent with the reported safety analysis. The assemblies repositioned on 04/30/79. Subsequent zero power physics testing on may 6 and at-power on May 7 confirmed the gadolinia-bearing assemblies were properly located (no core power tilts were measured).	Core Design QA

Table A-5 Summary of Industry Design Prediction and As-Built Core Problems								
#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Detection Method
							Therefore, it is concluded that there was no impact	
61	4 2 2.8	W-Pittsburgh	-	11/30/71	Beznau	W	NFD-NE-345: During startup testing for Beznau Unit 1 Cycle 2 flux maps indicated a power distribution anomaly. Specifically, a gross power tilt around Location G12 (location one row in from the periphery) was observed. Differences between measured and predicted power of +14% were observed in the vicinity of G12 while the symmetric locations directly opposite was -5%. Based on these results an investigation was initiated. Upon investigation it was determined that Assembly 161 was 3.35 (As-Built 3.395 w/o) and had been incorrectly loaded into Location H12. Assembly 161 was the only assembly fabricated at 3.35 w/o for Cycle 2 to match 3 spare assemblies (Assemblies 122-124) fabricated for Cycle 1 at 3.35 w/o. The remaining 36 assemblies (Assemblies 125-160) were fabricated at 3.18 w/o. It was incorrectly assumed that assemblies 122 through 125 were the 3.35 w/o and these were not loaded in the core. As a result Assembly 161 (3.395 w/o) rather than Assembly 125 (3.18 w/o) was placed in Location H12. The head was removed and Assembly 161 was replaced with Assembly 125. Following this interchange, differences between measured and predicted were reduced by more than one-half and were typical of past WEC startup experience. The enrichment difference between the final and misloaded assemblies was ~0.2 w/o; considerable less than 04 to 0.8 w/o difference normally existing between regions of fuel. Moreover, the misleading occurred near the periphery of the core were analytical accuracy is relatively poor. Nevertheless, the loading error was evident in the flux map, and on this basis WEC concluded that an undetected loading error is extremely improbable.	Incore Flux Symmetry
62	4.2.2.8	W-Pittsburgh	-	1985 (Before)	Farley Unit 1	W	A depleted asymmetric Burnable Poison Rod Assembly was loaded backwards in a peripheral assembly such that the rodlets were to the outside instead of the interior. The core operated the cycle with the misloaded insert, even though it was found during the cycle by surveillance testing using flux maps (Incore Power Distribution). It was not identified during the startup measurements. Consequences were minimal to that cycle. No asymmetric inserts were ever used again at Farley.	Incore Power Distribution
63	4.2.2 8	INPO - SER 11-95 OE6743	IBS	06/16/94	Plant X	W	According to the approved core reload design, part-length hafnium rods (inserts) should have been installed in eight fuel assemblies on the periphery of the core to suppress neutron flux in the vicinity of the reactor vessel belt-line weld area. However, when the vendor prepared the candidate loading pattern transmittal, some of the figures in the document did not show the hafnium inserts. Although the vendor did not intend for the document to be used for that purpose, the candidate loading pattern transmittal was used by site personnel to prepare the nuclear component transfer list. Later, the nuclear component transfer list, which also did not include the hafnium	Incore Flux Symmetry

Table A-5 Summary of Industry Design Prediction and As-Built Core Problems								
#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Detection Method
							inserts, was used as a reference to reload the reactor core. Consequently, the reactor core was reloaded without hafnium inserts.	
64	4.2.2.9	INPO-OE7719 INPO-OE7631	LER50-498 LER50-498	12/18/95 * 12/18/95	Plant X, Unit 1	W	INPO-OE7719: This Operating Experience entry is a follow-up to OE-7631, where several control rods failed to fully insert. The rod drop test results were evaluated against the approved safety evaluation (USQE) and the acceptance criteria contained in the Reactor Startup procedure. Based on these previously established criteria, the rod drop time and shutdown margin restart criteria were met by a large margin. INPO-OE7631: On 12/18/95, Unit 1 tripped due to a switchyard lockout. Following the trip, three control rods (one in Control Bank C -F10, and two in Shutdown Bank B - C09 and N07) were observed to be at 6 steps by Digital Rod Position Indication (DRPI). Preliminary manufacturing records review has not produced any adverse indications. A safety evaluation was performed that demonstrates that shutdown margin is met with all control rods. Note that the subject document did not identify how much time was lost as a result of this event.	CEA Trip
65	4.2.2.9	INPO - OE8467 INPO - OE7844	-	07/09/97 * 08/26/95	Plant X, Unit 2	W	OE8467: During unit trips and the refueling outage, control rod drop time measurements were performed (8/26/95, 1/17/97, 3/1/97 and 3/11/97). These measurements showed that the drop times for some rods had increased and they indicated incomplete insertion. These data and examination of traces revealed that the problem was related to rod friction. It was determined that assembly bowing increased the frictional drag on the control rod. Plant safety was not affected. OE7844: Following initiation of the refueling outage and with the unit in mode 3, a rod drop test was performed. During this test a control rod was discovered not fully inserted. This incomplete insertion was due to the presence of an obstruction inside one of the fuel assembly guide tubes. Inspection of the fuel assembly showed that it was bowed with respect to a vertical reference. Westinghouse considered that, according to their experience, the magnitude of the bowing was not significantly greater than what might be expected from the most bowed assemblies of any other core. The affected assembly was not scheduled to be inserted into the core. Note that the subject document did not identify how much time was lost as a result of this event. However, it is estimated that several hours to a day may have been lost.	CEA Drop Time
66	4.2.2.9	INPO - OE10675	LER 289-990911-1 LER 99-011-00	09/11/99	Plant X, Unit 1	B&W	On 9/11/99, during an optional performance of Control Rod Drop Time Tests, control rod group 2 rod 2 (2-2) and control rod group 5 rod 2 (5-2) did not insert full length as expected. Control rod 2-2 stopped at approximately 7% withdrawn and control rod 5-2 stopped at approximately 26% withdrawn. Control rods were then dropped a second time with the same results. It was determined that assembly bowing increased the frictional	CEA Drop Time

Table A-5 Summary of Industry Design Prediction and As-Built Core Problems								
#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Detection Method
			LTR192 0-99-20549 CAP T1999-0722 RCE 180				drag on the control rod. The fuel bow visual observations correlate with the control rod drop. The safety significance of this event was minimal since both rods inserted to a position where most of their rod worth would be effective.	
67	4.2.2.9	INPO Document	LER 50-302	10/01/99	Plant X, Unit 3	B&W	Two control rods exceeded the Improved Technical Specification (ITS) limit of 1.66 seconds during the performance of optional Control Rod Drop Time Test. The contributing causes of the slow rod drop times are: (1). Thermal barrier degradation due to crud accumulation in the check valve region causing the check valve balls to stick, and resulting in the higher flight time due to increased drag forces. (2). Deformation of the control rod guide tube caused the mechanical drag in the control rod guide tubes, which resulted in slow rod drop times due to the increased drag force. These failure mechanisms are industry wide issue. No human error occurred during this event. The fuel design limits would not have been exceeded for normal shutdown or any anticipated operation occurrences during the cycle as a result.	CEA Drop Time
68	4 2 2.9	INPO - OE7813	-	08/22/94	Plant X, Units 3 and 4	W	After a scram at Plant X Unit 4 in August 1994, where one control rod jammed 18 steps from full insertion, extensive investigations were launched. The investigations revealed that assemblies bowed in an S-shape caused the jamming. As the fuel in Plant X Unit 3 was identical to the fuel at Plant X Unit 4, bowed fuel assemblies also could be expected at Plant X Unit 3. Therefore drop tests were performed at Plant X Unit 3 as well. During drop tests at Plant X Unit 3 in February 1995, two control rods had longer drop times than the established surveillance criteria. The new fuel assemblies supplied to Plant X Units 3 and 4 have been modified in order to reduce the axial force on the assembly and increase the lateral stiffness. Note that the subject document did not identify how much time was lost as a result of this event.	CEA Trip
69	4.2.2.9	INPO - OE7861	LER50-498	12/18/95	Plant X, Unit 1	W	During the Plant X shutdown for refueling on May 18, 1996, more testing was performed for the incomplete reactor control rod insertion issue. The reactor was in Mode 3 Hot Standby, full flow conditions. After dropping all shutdown banks simultaneously, 6 shutdown bank rods did not fully insert; 5 were at 6 steps from rod bottom and 1 at 12 steps. No cause is known. It is estimated that several hours may have been lost.	CEA Manipulations
70	4.2 2 9	INPO - OE8032	LER50-412	09/09/96	Plant X, Unit 2	W	During the 6th-refueling, Plant X performed control rod drop and drag testing in fulfillment of commitments to NRC Bulletin 96-01. Three assemblies exhibited higher than expected drag forces during testing. No cause is known. It is estimated that several hours may have been lost.	CEA Manipulations

#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Detection Method
Table A-5 Summary of Industry Design Prediction and As-Built Core Problems								
71	4.2.2.9	INPO - OE1750	-	09/06/85	Plant X, Unit 3	Mitsu bishi	In one of fuel assemblies, one of two leaf spring cramps was out of place and was founded laying down on the vane which connects with spider hub of Control Rod Cluster due to a broken spring screw. One of 16 vanes of Control Rod Cluster was slipped out of spider hub. The broken spring screw has an indication of intergranular corrosion, which was presumably attributable to the fact that the spring screw had been tightened excessively in the assembling stage to produce a residual high stress as well as intergranular corrosion.. The disconnection of the spider hub in question was presumably caused by the restricted movement of the cluster inside its guide tube as a result of a foreign material trapped therein. Note that this event is also categorized as a CEA Finger Loss problem which is considered to be a consequence of this initiating event.	CEA Inspection
72	4.2 2.10						None Identified	
73	4.2.2.11	INPO - OE5782	IOI	05/92	Plant X	W	Plant X has been tracking a trend in both Axial Flux Difference (AFD) and Estimated Critical Position (ECP) measurements. The trend was noted during Cycle 5 (11/90-3/92) and was characterized by the AFD trending more negative than predicted. A consequence of this trend was an increasing error in Estimated Critical Position calculation.	Incore Power Distribution
74	4.2.2.11	INPO - OE6911	IOI	10/20/94	Plant X, Unit 1	W	Utility X believes that Plant X Unit 1 is experiencing the same axial offset phenomena that has previously been seen at Callaway, Vogtle, Harris, and Millstone Unit 3. Plant X Unit 1 is currently about 280 Effective Full Power Days (EFPD) into Cycle 8. The cycle length is 390 EFPD. At about 150 EFPD into this cycle, the measured axial offset (AO) was observed to be growing more negative than predicted. This difference has continued to increase as the cycle operated and following power runbacks took some step changes more negative. The incore measured AO at 255 EFPD was -8.5% versus a predicted value of about -4%. In addition to the negative AO, incore power distributions show the flux is depressed between the upper grid spans of the high powered assemblies. The most plausible theory for this phenomena is that a boron-lithium compound is being concentrated in the crud layer on the top of high powered fuel rods.	Incore Power Distribution
75	4.2.2.11	9712090146	INFO NOTICE 97-085	12/11/97	GENERIC	PWR	NRC INFO NOTICE 97-085 'Effects of Crud Buildup & Boron Deposition on Power Distribution & Shutdown Margin.' Addresses potentially significant problem pertaining to anomalous behavior of the core axial power distribution and erosion of shutdown margin (SDM) attributed to crud buildup on the nuclear fuel and subsequent boron deposition in the crud layer. The anomaly is characterized by a gradual unexpected power shift toward the bottom of the core and was first detected in Callaway Cycle 4 at approximately 7000 MWD/MTU. The power shift continued until burnup effects became dominant and cause the power to shift back to the top of the core nearing EOC. In addition to the anomalous power distribution,	Incore Power Distribution

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#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Detection Method
							deviations were observed in the Estimated Critical Position (ECP) of the control rods. Deviations over 0.5%Δρ were observed in Cycles 4 and 5. The crud buildup that causes the anomalous power distribution is particularly pronounced in high-power assemblies. After analyzing relevant data, performing scoping calculations, and reviewing industry experience, UE and Westinghouse concluded that the power distribution anomaly was most likely caused by the formation of crud and deposition of lithium borate initiated by subcooled nucleate boiling in the upper portion of the core. The ECP deviations were another effect of this anomaly. Incore detector indications of flux depressions between fuel grids in high power fuel assemblies, as well as visual examinations showing crud deposits on fuel pins, supported these conclusions. This resulting power shift causes a reduction in SDM and an increase in local peaking factors. Near the end of cycle, excess burnup in the bottom of the core and reduced boron and lithium concentrations in the reactor coolant system cause the power distribution to shift back toward the upper portion of the core, partially restoring the burnup distribution. A number of Westinghouse plants have experienced AOAs ranging from -3 percent to -15 percent. These plants include Callaway, Catawba 1, Comanche Peak 2, Millstone 3, Seabrook, Vogtle 1 and 2, and Wolf Creek. AOAs ranging from -3 to -15 percent have been observed. Core power was reduced to 95 and then 70% in response to a -15% AO at Callaway. Shutdown margin was decreasing at a rate of 3-4 pcm/day. Corrective actions to restore SDM include modifying rod insertion limits and rod worth uncertainties in the SDM calculation. **The notice did not identify any anomaly associated with radial power distribution or the ability to detect the anomaly during LPPTs.**	
76	4.2.2.11	INPO - OE9417 OE7651	IO1	05/10/98	Plant X	W	Plant X continues to experience an axial offset anomaly (AOA) during reactor Cycle 10 although so far the severity of the anomaly is less than Cycle 9. SEN 170 describes the severe AOA during Cycle 9 which caused eroding shutdown margin. Compensatory actions included operating at a reduced reactor power (about 70%) for approximately one third of the cycle. Root cause of the anomaly is boron precipitated in crud on the upper region of fuel rods	Incore Power Distribution
77	4.2.2.11	INPO - OE11095	IO1	10/13/99	Plant X, Unit 2	CE	On October 13, 1999, at approximately 160 EFPD (Effective Full Power Days), Reactor Engineering concluded that Palo Verde Unit 2 Cycle 9 was exhibiting indications of an axial shape trend that appeared to be similar to the axial offset anomaly (AOA) experienced by other nuclear facilities. The combination of high, multi-cycle steaming rates, i.e., subcooled boiling, sufficiently high beginning of cycle (BOC) boron concentration, and higher than normal mobile CRUD concentrations caused by an early cycle trip, made U2C9 susceptible to AOA.	Incore Power Distribution

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#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Detection Method
78	4.2.2.12	INPO - OE10274 * 499-990831-1	IA	08/31/99	Plant X, Unit 2	W	On August 31, 1999, with Plant X Unit 2 at 100 percent power, Westinghouse informed the station that they have discovered dimensional quality problems with some chrome-plated rodlets supplied from a subcontractor. Westinghouse determined that approximately 10 percent of the work-in-process rodlets had larger than anticipated diameters (approximately 0.3855 inch versus the specification of 0.3845 inch). The cause of this event was the subcontractor use of an oversize ring gauge.	CEA Fabrication QA
79	4.2.2.13						None Identified	
80	4.2.2.14	7810180107	LER 78-048-03L-0	09/18/78	Crystal River Nuclear Plant, Unit 3	B&W	LER 78-048-031-0: On 09/18/78 quadrant power tilt exceeded transient limit but not max limit. Caused by uncoupling of Group 5 Rod 1. The acceptance criteria for the Control Rod Drop Test procedure was revised and review of the new rod coupling procedure was planned to preclude recurrence of this event. Based on review of this LER it is concluded that symptoms of an uncoupled rod were present in the rod drop time test results. Similar events in the future would likely be detected by CEA Drop Characteristics.	Incore Flux Symmetry
81	4.2.2.14	8306270228	Part 21	06/22/83	Connecticut Yankee	W	Part 21 addressing improper latching of RCCAs identified on 03/15/83 during Zero Power Physics Tests. The specific test was not identified in the Part 21. Subsequent investigation confirmed that 4 RCCAs were unlatched. The RCCAs were latched using a revised installation procedure that emphasizes proper orientation of the CRDS to the RCCA while latching. The connect/disconnect CRDS buttons were examined for proper heights to confirm correct installation and latching. Additional information obtained for Connecticut Yankee indicates that the uncoupled rods were discovered during an "ejected rod worth test" a test similar to the CEA Flux Symmetry test. The test identified large measured reactivity differences between symmetric rods. In addition several rods did not exhibit any reactivity change when the rod (or in this case just the extension shaft) was moved. At some later time the rod drop traces were reviewed and clearly identified that the rods were uncoupled. Therefore, it is concluded that this event was detectable by both CEA Drop Characteristics and CEA Flux Change.	CEA Flux Symmetry
82	4.2.2.14	9011150125	SR	11/01/90	Yankee-Rowe Nuclear Power Station	W	Special Rept: On 11/01/90 Control Rod 24 was found disconnected from its drive shaft. Drive Shaft latching will be initiated. An orderly plant shutdown and cooldown were initiated after the determination that the drive shaft was not coupled was made. The subject document does not state how the uncoupled control rod was detected. Additional information obtained from Yankee Rowe indicated that the uncoupled control rod was first detected during control rod exercising prior to criticality. Specifically, CEA Position Indication was suspect. Furthermore, examination of rod drop time for Control Rod 24 indicted that the insertion was slower than normal. Yankee Rowe subsequently performed a CEA Flux Change Test on Control Rod 24, a test not normally performed, and confirmed that it was not	CEA Position Indication

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#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Detection Method
							coupled.	
83	4 2.2.14	9110080100	Insp Repts 50-280-91-20 & 50-281-91-20	09/16/91	Surry Power Station, Unit 1 Surry Power Station, Unit 2	W	Insp Repts 50-280-91-20 & 50-281-91-20: On 910708 12 Violations noted. major areas inspected: startup tests thermal power analysis power distribution monitoring nuclear instrument calibrs & followup of unresolved items. Following completion of low power physics tests for Unit 2, Cycle 11, it was determined that control rod F-6 (Control Bank D) had been unlatched and fully inserted throughout the tests. The Hot Rod Drop test results were reviewed and the trace for control rod F-6 was not as smooth, prior to deceleration into the dashpot, as that of other rods. The test engineer concluded following the test that the trace was the result of otherwise undetected electrical noise and was not unreasonable; the drop time for F-6 was within the span of the other results. Subsequent Zero Power Testing yielded measurements within the acceptance criteria for all parameters (CBC, ITC, DBW, Reference Bank Worth and Total Worth of all Banks). The first flux map at ~30% power revealed that control rod F-6 was fully inserted (unlatched). A comparison of measurements to predictions for Zero Power Tests with F-6 inserted was then made. This comparison demonstrated that insertion of F-6 did not greatly change the predicted values, and the measured values satisfied the acceptance criteria for both sets of calculations. The report did not identify the discrepancy in Control Bank D worth. However, it likely was within the typical acceptance criteria for a Test Bank for rod swap of ±15%. Furthermore, the report did not identify the impact of the uncoupled CEA on radial peaking	Incore Power Distribution
84	4.2.2.14	NED-DEN-02-0035	-	12/15/78	Fort Calhoun	CE	On December 15, 1978, the plant was placed into cold shutdown after completion of low power physics testing to re-couple control rod drive mechanism 31. The uncoupled CEA was not discovered during coupling weight and height checks nor during rod drop testing. The result was discovered during full length CEA symmetry checks. Steps were not implemented into the procedure to ensure complete coupling of the CEA. Personnel error was also a contributing factor due to inadequate training.	CEA Flux Symmetry
85	4.2.2.14	RAC-2-88130	-	05/08/88	Arkansas Nuclear One	CE	On May 8, 1988, during startup testing, it was discovered that CEA 2-8 was not coupled. During coupling, the CEA was not completely coupled due to inadequate procedures. Steps were not implemented into the procedure to ensure complete coupling of the CEA. Personnel error was also a contributing factor due to inadequate training.	CEA Flux Symmetry
86	4.2.2.14	9307010013	LER 93-005-00	06/24/93	St. Lucie Plant Unit 1	CE	LER 93-005-00: On 05/29/93 discovered that one CEA of dual element-control element drive mechanism potentially unlatched. Possibly caused by personnel error. CEA relatched & latching procedure modified. During Low Power Physics Tests (CEA Flux Symmetry for Dual CEAs), core neutron flux asymmetries were identified that indicated one CEA of a dual CEA drive mechanism (CEDM) #7 was potentially uncoupled.	CEA Flux Symmetry

#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Detection Method
							Subsequently computer analysis and flux distribution modeling confirmed an unlatched inboard CEA of CEDM #7 as the most probable cause of the observed flux distributions. The reactor was disassembled and ensuing inspections confirmed that the inboard CEA of CEDM #7 was unlatched. The most likely cause of the event was due to personnel error on the part of contractor and plant personnel in not adequately executing CEA latching. Contractor personnel performing the latching may not have correctly evaluated the full engagement of both CEAs. Additionally, post-latching verification as indicated by the extension shaft pin position may have been adequately performed. Both of these checks could have alerted the personnel that the CEA was not fully latch. Contributing factors included inadequate procedural guidance on methods of ensuring proper latching, human factor concerns in performing the indicator pin verification, and lack of independent verification of position indicator pin location. Results of the initial latch verifications (pin position and weights) and subsequent core physics tests (initial results of CEA Symmetry Checks for CEDM #7) suggested that both CEAs of CEDM #7 were initially attached, but that one CEA had become unlatch during Low Power Physics Tests. Corrective Actions committed to included upgrading of the coupling procedure to provide increased assurance of proper CEA coupling. Changes that were to be considered include (a) elevation measurements to verify post coupling position, (b) recording of coupling tool indicator position as an additional confirmation of position indicator pin location and (c) ensuring sufficient slack exists in the coupling tool cables while withdrawing gripper plungers during coupling. Also being considered are human factors improvements to the position pin verification including (a) additional lighting and (b) independent verification. **Note: Although the detection method for this particular unlatched CEA was CEA Flux Symmetry, the coincidence of Low Power Physics Tests and the release of the partially latched CEA were fortuitous. In the event that the CEA became unlatch following CEA symmetry checks, but before CEA Group Worth measurement, it is unlikely that the unlatch CEA would be detected by CEA Group Worth measurement, if measured. This is because the discrepancy in group worth would likely be within the acceptance criteria. Regardless, the uncoupled CEA would have been either detected during the Incore Flux Symmetry test at ~30% power or included as a penalty in the radial peaking factor.** W-930624 Ltr.	
87	4.2.2.14	INPO - OE8587	IO2	06/16/97	Plant X, Unit 3	CE	On June 16, 1997 with Unit 3 completing a Refueling Outage and in reactor startup testing, it was determined that CEA 91 was not coupled to its extension shaft. The plant was returned to refueling mode, the reactor head was removed, and an inspection of CEA 91 determined that one of the CEA fingers indicated that the extension shaft had inadvertently been inserted and	CEA Position Indication

#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Detection Method
							latched between the center hub and an adjacent finger instead of inside the center hub. The CEA 91 and extension shaft were replaced. The outage was extended 25 days. Steps were not implemented into the procedure to ensure complete coupling of the CEA. Personnel error was also a contributing factor due to inadequate training. Note that Plant X's Corrective Action Program documentation relating to this event states that CEA 91 (a) does not show the dashpot effect exhibited by all of the other full strength CEAs and (b) bounced higher and more times than the other three four-finger CEAs. Furthermore, unlike the other three four-finger CEAs, CEA 91 did not result in any change in excore detector count rate during insertion. Therefore, it is concluded that this event was detectable by both CEA Drop Characteristics and CEA Flux Change.	
88	4.2 2 15	CE NPSD-955-P	-	1986	Calvert Cliffs Unit 1	CE	CEA inspections were performed at EOC 8 using ECT on 68 CEAs in Calvert Cliffs Unit 1. Axial cracking of CEA clad was observed in the center B_4C fingers of 4 CEAs at the elevation of the lowest B_4C pellet. The axial cracking was confirmed by visual inspections. The axial cracks were tight and there was no evidence of poison loss. No circumferential cracking was observed in any of the fingers. No strain or cracks were observed in the outer four AgInCd fingers of any CEA.	CEA Inspections
89	4.2 2 15	CE NPSD-955-P	-	1987 & 1989	Calvert Cliffs Unit 2	CE	CEA inspections were performed at EOC 7 and EOC 8 using ECT on 68 CEAs in Calvert Cliffs Unit 2. Axial cracking of CEA clad was observed in the center B_4C fingers of 29 CEAs at the elevation of the lowest B_4C pellet. The axial cracking was confirmed by visual inspections. The axial cracks were tight and there was no evidence of poison loss. No circumferential cracking was observed in any of the fingers. No strain or cracks were observed in the outer four AgInCd fingers of any CEA.	CEA Inspections
90	4.2.2.15	CE NPSD-955-P	-	1990	St. Lucie Unit 1	CE	CEA inspections were performed at EOC 9 using ECT on 50 CEAs in St. Lucie Unit 1. Axial cracking of CEA clad was observed in the center B_4C fingers of 5 CEAs at the elevation of the lowest B_4C pellet. The axial cracking was confirmed by visual inspections. The axial cracks were tight and there was no evidence of poison loss. No circumferential cracking was observed in any of the fingers. No strain or cracks were observed in the outer four AgInCd fingers of any CEA.	CEA Inspections
91	4.2.2.15	IN 87-19	-	1987	Multiple	W	This notice is provided to inform recipients of a potentially significant safety problem that could result from the perforation and cracking of the rod cluster control assemblies (RCCAs) in Westinghouse PWRs.	CEA Inspections
92	4.2.2.15	INPO-OE2495	-	09/30/77	Plant X, Unit 1	MHI	During the 9th refueling and maintenance outage at Plant X Unit 1 of Utility X, Inc. measurement of outside diameter of 29 control rods revealed the thinning of cladding and swelling. Control rod thinning was caused by the friction with control rod's guide-plate located between upper core plate and upper core support plate. The swelling were due to absorber (Ag-In-Cd)	CEA Inspections

#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Detection Method
							irradiation by neutron for long time.	
93	4.2.2.15	INPO-OE3019	-	09/03/85	Plant X	W	During the current refueling outage inspection of the Rod Cluster Control Assemblies (RCCA), eddy current testing has identified numerous anomalous indications on essentially every rodlet of each RCCA. These anomalies consist of interior clad wall indications and increases in overall rodlet diameter. The indications include cross-sectional clad area losses of up to 40%, with some localized areas approaching through-wall from the clad interior. Rodlet diameter increases approach 30 mils in some cases.	CEA Inspections
94	4.2.2.15	8605050015	LER 86-015-00	04/23/86	Connecticut Yankee Atomic Power	W	LER 86-015-00: On 860319 during the EOC 13 refueling Eddy Current Inspection performed on Rod Cluster Control Assemblies (RCCAs) indicated clad cracking sliding wear & Guide Card Wear. Thirty two (32) of the forty seven (47) RCCA's examined showed signs of rodlet cracking with the worst case RCCA having thirteen (13) of twenty (20) rodlets cracked. These indications have been confirmed by visual techniques. A significant amount of clad cracking was observed just above the end plug region of the RCCA rodlet. This phenomenon is a direct result of AgInCd swelling whereby the effect of fluence causes the material to contact the clad. The resulting stress, in conjunction with the material properties to the stainless steel tube, creates an IASCC condition. Cracking was axial in nature and little or no diameter changes to the rodlets occurred. Since the poison material in the RCCAs appear to be limited to AgInCd, and there was no concerns relating to the ability to insert the RCCAs or rodlet loss identified, no impact on CEA Worth or peaking is assumed. Worst case RCCAs replaced with Westinghouse onsite spares. W-860423 Ltr.	CEA Inspections
95	4.2.2.15	INPO-OE2343	-	01/19/88	Plant X, Unit 1	MHI	During the 10th refueling and maintenance outage at Plant X Unit 1 of Utility X, Inc. measurement of outside diameter of control rods revealed the thinning of cladding and swelling. Fourteen control rods whose changes were considered to be relatively large (10 control rods: thinning, 4 control rods: swelling) were replaced with new ones.	CEA Inspections
96	4.2.2.15	INPO-OE2491	-	02/28/88	Plant X, Unit 2	MHI	During the 9th refueling and maintenance outage at Plant X Unit 2 of Utility X, Inc. measurement of outside diameter of 48 control rods revealed the thinning of cladding and swelling. The swelling were presumed to be due to neutron absorber (Ag-In-Cd) irradiation by neutron. The thinning was presumed to be due to its cladding interference with guide tube caused by coolant turbulence for long time.	CEA Inspections
97	4.2.2.15	8901260047	Part 21	01/09/89	Maanshan, U1 Wolf Greek Callaway	W	Part 21 Rept summarizing full length hafnium rod cluster control assemblies anomaly update. Status of work listed. Hot Cell examination of Maanshan Unit 1 rodlets has verified that the swelling phenomenon is predominately located in the upper sections of the rodlets. Hafnium hydrid has been identified to be the cause of the swelling. Hydriding was also observed on the bottom surface of the absorber tip. The stainless steel clad near the tip	CEA Inspections

Table A-5 Summary of Industry Design Prediction and As-Built Core Problems								
#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Detection Method
							apparently became sensitized due to irradiation, similar to the material changes noted with longer term operating AgInCd RCCA's. These facts suggest that the bent and broken tips are the result of cracks formed in the clad, which are further exacerbated by some combination of plant scram, heatups/cooldowns, or other events leading to relative thermal expansion stresses between the absorber, cladding, and end plug. No loss of poison was identified. However, swelling in the range of 8 - 23 mils have been observed and significant bulging of the clad is noted (0.404 inch). Categorized as CEA damage since swelling and bulging may interfere with travel.	
98	4.2.2.15	INPO-OE4413	-	02/02/91	Plant X, Unit 1	CE	The core center Control Element Assembly (CEA) failed to fully insert during Unit 1 shutdown. Readings of the Control Element Drive Motor coil traces indicate the control rod is binding in the buffer region of the guide tube. This binding is believed to be due to swelling of the zircaloy slugs of the control rod. Similar swelling has been experienced on three CEAs. The center CEA was designed for power distribution control early in the life of the core and now provides very little reactivity control. Only the center one of the five fingers serves any reactivity function; the remaining four are filled with aluminum oxide pellets with a zircaloy plug at the bottom of each finger.	EOC CEA Insertion
99	4.2.2.15	INPO-OE13420	INPO Events Database 348-011031-1	10/31/01	Plant X, Unit 1	W	On October 31, 2001 at Plant X Unit 1, during routine Cycle 18 start-up rod operability testing one of the Rod Control Cluster Assemblies (RCCAs) would not fully insert (exhibited Incomplete Rod Insertion in the Dashpot, IRID). This RCCA was located in a fresh fuel assembly and stopped in the dashpot region of the fuel assembly at approximately step 24 (about 15 inches from bottom). RCCA drop testing revealed that 3 RCCAs were not fully inserting into the dashpot region. Two of the RCCAs stopped at approximately step 24. The third RCCA stopped at approximately step 18, and fully inserted after 1 1/2 minutes Four (4) other RCCAs exhibited slow times (> 1 0 second) through the dashpot region when compared to the other RCCAs. All of the slow or incompletely inserted RCCAs had been in service for 18.8 effective full power years of reactor operation. The cause of the incomplete and slow insertions was determined to be swelling of the RCCA rodlet tips. This event is not significant because the incomplete and slow insertion problem among the remaining original RCCAs was discovered and corrected before the unit returned to power following the refueling. This event is noteworthy because the refueling outage was extended several days to replace the remaining RCCAs.	CEA Manipulation
100	4.2 2.16	LER 2001-003-00 NSAL-01-5,	-	12/04/01 * 09/26/02	Palo Verde Unit 3	CE	LER 2001-003-00: On October 10, 2001 Palo Verde Unit 2 was in Mode 1 (POWER OPERATION), operating at approximately 100 percent power when Control Room personnel were advised of a potentially transportable	CEA Inspections

#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Detection Method
		Rev. 2					condition affecting the integrity of Control Element Assemblies (CEAs). Preliminary inspection of Unit 3 CEAs during refueling revealed one CEA with cracks. Due to the similar design and operating history of CEAs, similar cracks were assumed to also be present in Unit 2. The CEA degradation LER reports APS' activities for Units 1, 2, and 3. The degradation was also discovered in Unit 1 and Unit 3. LCO 3.0.3 was not applicable at the times of discovery for Units 1 and 3's CEA degradation. The time of discovery for Units 1 and 3 occurred during refueling activities (MODE 6) where the operability of the CEAs is not required. Also, during the Unit shutdowns prior to the Unit 1 and Unit 3 refueling outages, the CEAs performed their design functions. Due to the conservative decision to declare the Unit 2 CEAs inoperable, it is considered that the same condition prohibited by Technical Specifications existed in Unit 1 and Unit 3. During initial inspections of Unit 3 CEAs, several fingers of one CEA were observed to emit a small stream of bubbles from the top of a crack-like indication. All full length CEAs were replaced during the outage. NSAL-01-5, Rev. 2: The Palo Verde Unit 3 CEAs were found to exhibit cracked cladding and one (1) finger was missing its nose cap. No loss of neutron absorber material was detected. Based on visual inspections and detailed inspections performed in August 2001 and January 2002, and evaluation of operating histories, it was judged that the Palo Verde CEA finger damage is the result of IASCC. As originally reported in NSAL-01-5, Rev. 0, Westinghouse evaluated the available information and determined that the CEA damage condition as it existed at Palo Verde 1 did not constitute a substantial safety hazard. Based on the additional information from the Palo Verde 2 and 3 CEAs, Westinghouse has determined that for the existent conditions the same conclusion (i.e., not constituting a substantial safety hazard) also applies to these Units.	
101	4.2.2.16	LER 2001-003-00 NSAL-01-5, Rev. 2	-	12/04/01 * 09/26/02	Palo Verde Unit 2	CE	LER 2001-003-00: On October 10, 2001 Palo Verde Unit 2 was in Mode 1 (POWER OPERATION), operating at approximately 100 percent power when Control Room personnel were advised of a potentially transportable condition affecting the integrity of Control Element Assemblies (CEAs). Preliminary inspection of Unit 3 CEAs during refueling revealed one CEA with cracks. Due to the similar design and operating history of CEAs, similar cracks were assumed to also be present in Unit 2. The CEA degradation LER reports APS' activities for Units 1, 2, and 3. The degradation was also discovered in Unit 1 and Unit 3. LCO 3.0.3 was not applicable at the times of discovery for Units 1 and 3's CEA degradation. The time of discovery for Units 1 and 3 occurred during refueling activities (MODE 6) where the operability of the CEAs is not required. Also, during the Unit shutdowns prior to the Unit 1 and Unit 3 refueling outages, the	CEA Inspections

Table A-5 Summary of Industry Design Prediction and As-Built Core Problems								
#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Detection Method
							CEAs performed their design functions. Due to the conservative decision to declare the Unit 2 CEAs inoperable, it is considered that the same condition prohibited by Technical Specifications existed in Unit 1 and Unit 3. During initial inspections of Unit 3 CEAs, several fingers of one CEA were observed to emit a small stream of bubbles from the top of a crack-like indication. All full length CEAs were replaced during the Unit 3 outage. Preliminary inspections of the Unit 2 CEAs during replacement activities found several CEA fingers with evidence of cracking near their lower ends. The nose cap of one finger of one CEA was also observed to have separated from the finger. All full length CEAs were replaced during the ensuing mid-cycle outage. NSAL-01-5, Rev. 2: No loss of neutron absorber material was detected from the Palo Verde Unit 2 CEAs. Based on visual inspections and detailed inspections performed in August 2001 and January 2002, and evaluation of operating histories, it was judged that the Palo Verde CEA finger damage is the result of IASCC. As originally reported in NSAL-01-5, Rev. 0, Westinghouse evaluated the available information and determined that the CEA damage condition as it existed at Palo Verde 1 did not constitute a substantial safety hazard. Based on the additional information from the Palo Verde 2 and 3 CEAs, Westinghouse has determined that for the existent conditions the same conclusion (i.e., not constituting a substantial safety hazard) also applies to these Units.	
102	4.2.2.16	9007190331	INPO-OE3993 INPO-OE4007 InfoBulletin 90-03 MY letter 07/05/90	06/11/90	Maine Yankee	CE	A problem with insertion/withdrawal of a Control Element Assembly (CEA) was encountered during pre-critical testing following the cycle 12 refueling outage at Maine Yankee. While performing CEA operability tests with the plant in a cold shutdown mode, it was discovered that one (of 85) Control Element Assembly became stuck at approximately 80% insertion. Each CEA is comprised of five 0.948" OD Inconel tubes containing boron carbide absorber pellets. Following withdrawal of the stuck CEA, it was discovered that the central tube end cap and the absorber pellet stack were missing. Thirty-three boron carbide pellets were found in the center guide tube of the fuel assembly containing this CEA. The cause for the end cap failure and the four-inch long axial crack observed on the lower end of the central CEA tube has not currently been identified. Testing revealed that two more CEAs were missing center finger end caps and six additional CEAs showed cracking. It is believed that some boron carbide pellets dissolved during operation and others are in the spent fuel pool. Based on limited number of affected fingers, it is judged that the impact was probably less than the uncertainty on total CEA worth.	CEA Manipulation
103	4.2.2.16	LER 2001-	-	12/04/01	Palo Verde	CE	LER 2001-003-00: On October 10, 2001 Palo Verde Unit 2 was in Mode 1	EOC CEA

Table A-5 Summary of Industry Design Prediction and As-Built Core Problems

#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Detection Method
		003-00 NSAL-01-5, Rev. 2		* 09/26/02	Unit 1		(POWER OPERATION), operating at approximately 100 percent power when Control Room personnel were advised of a potentially transportable condition affecting the integrity of Control Element Assemblies (CEAs). Preliminary inspection of Unit 3 CEAs during refueling revealed one CEA with cracks. Due to the similar design and operating history of CEAs, similar cracks were assumed to also be present in Unit 2. The CEA degradation LER reports APS' activities for Units 1, 2, and 3. The degradation was also discovered in Unit 1 and Unit 3. LCO 3 0.3 was not applicable at the times of discovery for Units 1 and 3's CEA degradation. The time of discovery for Units 1 and 3 occurred during refueling activities (MODE 6) where the operability of the CEAs is not required. Also, during the Unit shutdowns prior to the Unit 1 and Unit 3 refueling outages, the CEAs performed their design functions. Due to the conservative decision to declare the Unit 2 CEAs inoperable, it is considered that the same condition prohibited by Technical Specifications existed in Unit 1 and Unit 3. *NSAL-01-5, Rev. 2: Damaged CEAs were originally discovered by APS at Palo Verde 1 in March 2001 and resulted in the replacement of all the Unit's full-strength CEAs. Westinghouse supported an APS detailed inspection of the discharged Palo Verde 1 CEAs and discovered, in addition to the original visually observed damage (i.e., via video), that ten (10) CEA fingers had experienced a loss of some or all of their neutron absorber material (boron carbide, B4C). Based on visual inspections and detailed inspections performed in August 2001 and January 2002, and evaluation of operating histories, it was judged that the Palo Verde CEA finger damage is the result of IASCC As originally reported in NSAL-01-5, Rev. 0, Westinghouse evaluated the available information and determined that the CEA damage condition as it existed at Palo Verde 1 did not constitute a substantial safety hazard.	Insertion
104	4.2.2.17	8203240246	LER 82-003-99x-0	02/23/82	San Onofre Nuclear Station, Unit 1	W	LER 82-003-99x-0: On 811212 Control Rod C-7 stuck during startup & was freed after repeated manipulation. Previous incidences of dropped rodlets were caused by failure of the weld attaching the vane supporting two rodlets to the RCC Hub. Evidence indicates the likelihood of this mode of failure. Caused by dropped rodlets due to failure of weld attaching Supporting Vane To Rod Cluster Control (RCC) Hub. RCCA to be replaced. Note that the **existence of dropped rodlets was initially suspected by the RCCA becoming stuck (CEA manipulations). The number of dropped rodlets was subsequently determined to be one or two by surveillance techniques showing difference in temperature and flux levels (Incore Flux Symmetry & Incore Power Distribution). The impact on CEA worth and Power Distribution are assumed to be much less and less than the uncertainty of CEA worth and Power Distribution**	CEA Manipulation

Table A-5 Summary of Industry Design Prediction and As-Built Core Problems								
#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Detection Method
							respectively.	
105	4.2.2.17	W-Pittsburgh	-	1976 (approx.)	Salem Unit 1	W	During the Salem Unit 1 Cycle 1 startup, the results of measurements (rod worths and power distributions) were outside or right up against the review criteria. Since there didn't appear to be a global issue (tilt) or local issue (manufacturing), plant operation continued. During the first refueling, approximately 7 detached rodlets were discovered distributed throughout the core. The results of the flux maps were evaluated using contour analysis and the broken rodlets were "located". Based on the results of the investigation, the rodlets were determined to be detached prior to initial criticality. Failure mechanism was determined to be SCC at the joint where the rodlet is connected to the spider assembly.	CEA Inspection
106	4.2.2.17	INPO - OE1750	-	09/06/85	Plant X, Unit 3	Mitsu bishi	In one of fuel assemblies, one of two leaf spring cramps was out of place and was founded laying down on the vane which connects with spider hub of Control Rod Cluster due to a broken spring screw. One of 16 vanes of Control Rod Cluster was slipped out of spider hub. The broken spring screw has an indication of intergranular corrosion, which was presumably attributable to the fact that the spring screw had been tightened excessively in the assembling stage to produce a residual high stress as well as intergranular corrosion. The disconnection of the spider hub in question was presumably caused by the restricted movement of the cluster inside its guide tube as a result of a foreign material trapped therein. Note that this event is also categorized as a Fuel Distortion problem, which is considered to be the initiating event.	CEA Inspection
107	4.2.2.17	W-Pittsburgh	-	1977 (approx.)	Cook Unit 2	W	During the Cook Unit 2 Cycle 1 startup the methodology developed after the Salem Unit 1 Cycle 1 Finger Loss (above) was used to determine that several rodlets had become detached. Note that CEA Group Worth tests did not identify any anomaly. Results of later inspections confirmed that 2-3 rodlets had become detached due to the same problems at Salem.	Incore Power Distribution
108	4.2.2.18						None Identified	
109	4.2.2.19						None Identified	

Table A-6 Summary of Industry Test Performance Problems								
#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Initiating Test
1	4.3.2.1	8506040508	-	04/21/85	Oconee Unit 2	B&W	On April 21, 1985 at 0051 hours, Oconee 2 tripped during zero power physics testing (ZPPT), when Reactor Protection System (RPS) channels A and B sensed a flux level in excess of the setpoint. A malfunction in the power range recorder, which caused a less-than-actual indication of flux, led the operators to exceed the setpoint, causing the trip. No problems were identified with any of the equipment, but the chart for the power range recorder showed that the power range indicator apparently had stuck at 0.1 percent full power.	LPPT * Rx Trip involved
2	4.3.2.1	Participant	N/A	1981	Millstone Unit 2	CE	During Cycle 4 CEA Flux Symmetry tests, Groups 4 (-1.87¢), 3 (-1.56¢), 1 (-1.52¢) and A (-1.53¢) failed the acceptance criteria for individual group worth (±1.5¢). The results and measurement technique were reviewed and it was concluded that the discrepancy was likely related to the use of opposing Upper and Lower Excore channels, e.g., Upper Control Channel X & Lower Control Channel Y, input to the reactivity computer. Subsequent azimuthal tilt measurements at 50% (0.008) and 100% (0.003) power confirmed no significant asymmetry was present and that the discrepancy was due to the use of opposing Upper and Lower Excore channels.	CEA Flux Symmetry * Reactivity Computer involved
3	4.3.2.1	W-Pittsburgh	-	1986 (approx.)	Byron	W	The OTΔT channel coincident with the PR channel used for the LPPT was placed in trip mode. Another loop RTD failed, generating a trip signal which tripped the reactor during the LPPT. Tech Specs (new ITS) now allow testing by not placing those channels in trip mode. This is not universally in use by the plants.	LPPT * Rx Trip involved
4	4.3.2.1	8712140257	LER 87-010-00	12/07/87	Oconee Nuclear Station, Unit 1	W	LER 87-010-00: On 871105 manual reactor trip occurred due to component failure during Startup Physics Testing (Zero Power Physics Tests). Control Rod (CR) Groups 1 through 5 were withdrawn to 100% with no problems. CR Group 6 was pulled to 75% at which time Group 7 should have started to withdraw. When CR Group 6 reached 75% withdrawn control power to the control rods was lost. With control power lost, the rods could not be withdrawn or inserted while operating in this mode. The reactor was manually tripped. The rods fell immediately and no abnormal responses were received. Following investigation and restoration of control power clearance was provided to restart ZPPT. ZPPT was restarted and again Group 7 did not respond. The decision was made to select "Sequence Override" to allow testing to continue while I&C continued troubleshooting the system. The loss of control power was caused by a loose solder joint at Pin 1 of Control Rod Sequencer Card. Joint repaired. A review of the summary of incidents at Oconee revealed that there had been one other trip during ZPPT. This incident was reported in LER 270/85-02 and concerned a Reactor Trip on high flux indication. Although the LER did not identify any adverse impact on ZPPT the delays and use of "Sequence Override"	CEA Worth Test * CEAs involved

#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Initiating Test
							provided a less than desirable environment for testing. Therefore, this event has been included in this summary of operating experience and classified as a Test Equipment Problem. W-871207 Ltr.	
5	4.3.2.1	9101310207 INPO 370-901227-1	LER 370-901227-1	12/27/90 03/06/96	McGuire Unit 2	W	On December 27, 1990 operations (OPS) and performance (PRF) reactor group personnel were performing routine rod movement testing associated with Zero Power Physics Testing (ZPPT). OPS personnel attempted to insert Shutdown Bank E from the fully withdrawn position. Shutdown Bank 3 fell into the core taking the reactor subcritical. The operators then manually scrammed the reactor. An independent technical review was performed on the event and, consequently, a decision was made by station management personnel to restart the reactor. Unit 2 was returned to mode 2 (startup) operation about a day after the trip. Note that the subject document did not identify how much time was lost as a result of this event. However, it is estimated that several hours to a day may have been lost.	CEA Worth Test * CEAs involved
6	4.3.2.1	W-Pittsburgh	-	1990 (approx.)	Callaway	W	A Power Range channel not used by the reactivity computer generated a rate trip due to channel noise during the LPPT. Callaway (and others) developed and used a reactivity computer on the plant process computer. Tech Specs (new ITS) now allow testing by not placing those channels in trip mode. Callaway now uses DRWM using a PR channel.	LPPT * Rx Trip involved
7	4.3 2.1	Participant	-	1981	Millstone Unit 2	CE	During Cycle 4 CEA Group Worth tests, Group 4 measured worth (0.201%Δρ or 16.7%) failed the acceptance criteria for individual group worth (larger of ±0.1% Δρ or ±15%). In addition, the total group measured worth (Groups 7-2 +11.9%) failed the acceptance criteria for total group worth (±10%). The overlap data was reviewed and the safety analysis revised to reflect reduced rod worth. Subsequent evaluation determined that the cause for the lower measured worth was associated with the β-eff supplied by Westinghouse and use of two upper detectors in the non-overlap measurement (use on overlap measurements that used one upper and one lower detector yielded results within the acceptance criteria).	CEA Worth Test * Reactivity Computer involved
8	4.3.2.1	9706180291	-	05/14/97	McGuire Unit 1	W	On May 14, 1997, at 1104, the Unit 1 Reactor tripped during performance of Power Range (PR) Detector N42 Analog Channel Operational Testing (ACOT). Prior to the performance of the ACOT, PR Detector channel N41 had been bypassed due to being connected to the Reactivity Computer in preparation for Zero Power Physics Testing. The procedure chosen to bypass the N41 channel failed to bypass Permissive P-8, which unblocks reactor trip due to turbine trip. This event was caused by personnel utilizing an inappropriate procedure for the existing plant conditions. This event is considered to be of no significance with respect to the safety of the public.	LPPT * Reactivity Computer involved
9	4.3.2.1	INPO-OE12485	-	01/12/01	Plant X, Unit 1	B&W	While transitioning the core (O1C20) from Zero Power Physics Testing (ZPPT) to Power Escalation Testing (PET) the reactor was shutdown due to conflicting reactivity indications between the control room chart recorder	LPPT * Reactivity

Table A-6 Summary of Industry Test Performance Problems

Table A-6 Summary of Industry Test Performance Problems								
#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Initiating Test
							and the reactimeter. The same Nuclear Instrumentation (NI) detectors feed both control room and reactimeter indications. Control room indications did not fail. The control room indication provides wider band of power indication and is processed differently than the reactimeter circuit. Based on this, it was evident that the reactimeter signal was in error and a root cause investigation was warranted. Based on the investigation it was determined that the High Gain Bypass Filter (HGBF) circuit offset adjustment was not included in the calibration procedure for the NI circuits feeding the reactimeter. This lack of calibration caused the offset for each NI circuit to drift such that the voltage input to the reactimeter flat-lined, which produced a reactivity indication of zero.	Computer involved
10	4.3.2.1	INPO-OE13753	-	02/21/02	Plant X, Unit 1	W	Dynamic Rod Worth Measurement (DRWM) results for the cycle 5 reload core did not meet acceptance criteria. It was decided to measure rod worth using the rod swap method. Results from this method correlated very closely to design predictions (0.32 % above predicted for total rod worth; all individual bank worths met acceptance criteria). This event impacted the completion of low power physics testing by approximately 24 hours. The preliminary indication of the cause of this event is the leakage compensation (bucking) current was insufficient to compensate for leakage current. Reactor engineering personnel were not aware that adjusting leakage currents was a recommended iterative process to "fine tune" the bucking currents. Consequently, the compensation current was set but not readjusted. The procedure, written from the Westinghouse guideline, only requires adjusting leakage currents if more than four hours have elapsed since the bucking current was first set. A similar event may have been averted during the previous two refueling outages, because the bucking current was set and readjusted at least once due to the time criteria being (or nearly being) met.	CEA Worth Test * Reactivity Computer involved
11	4.3.2.1	INPO-OE13922	-	05/19/02	Plant X, Unit 2	W	On May 19, 2002, Plant X Unit 2 reactor was manually tripped as a result of a Rod Control System Urgent Failure Alarm in the 2BD power cabinet. During performance of low power physics testing, Operations personnel attempted to insert Shutdown Rod Bank B, when the rod control urgent alarm actuated. After actuation of the alarm, operators identified that Group 2 of Shutdown Rod Bank B and Control Bank D would not move. The most apparent cause of the condition appears to be an intermittent failure of the MXR2 multiplexing relay in the rod control system.	CEA Worth Test * CEAs involved
12	4.3.2.1	W-Pittsburgh	-	11/22/02	Indian Point 2	W	During Control Rod testing in Hot shutdown conditions, invalid RPS actuation was due to ongoing work. OTΔT channel 2 was placed in trip due to nuclear flux power range channel N 42 being removed from service to connect to the reactivity computer for low power physics testing. The reactor trip breakers were closed to support LPPT testing. While re-	CEA Worth Test * Rx Trip involved

#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Initiating Test
							terminating RCS narrow range RTD's, technicians landed a lead on loop 1 cold leg RTD, generating an Over Temperature Delta Temperature (OTΔT) signal. This resulted in a reactor trip due to a 2/4 channel OTΔT trip logic. Plant recovery was achieved in accordance with existing operating procedures.	
13 .	4.3.2.2	8311210163	LER 76-025-03L	12/20/76	Salem Nuclear Generating Station, Unit 1	W	LER 76-025-03L: On 761215 the stuck rod moved from 228 to 197 steps to maintain reactor critical while measuring stuck rod worth during Low Power Testing. Caused by slight overdilution of reactor coolant boron concentration. The amount of the overdilution was about 2 ppm boron which is less than the measurement uncertainty for boron titration. The reactor was immediately borated to return rod B-6 to 228 steps (ARO). Note that the test was a one time test and the overdilution was due to personnel error. W-770114 Ltr.	CEA Worth Test * CEAs involved
14	4.3.2.2	INPO - SER 15-90	-	04/16/90	Plant X, Unit 1	W	On April 16, 1990, Plant X Unit 1 was critical at 3.5 x E-8 amperes in the intermediate range when a control rod bank dropped into the core while performing low power physics testing. The reactor was returned to criticality by withdrawing another control rod bank. (This returned the overall rod bank configuration to one that had existed earlier in the test.) The control system for the dropped rod group was then repaired and a post-maintenance test was performed with the reactor critical. This event is significant because when control rods dropped during low power physics testing, the test was not terminated by fully shutting down the reactor. Further, criticality was restored rather than scramming or shutting down the reactor until the problem causing the dropped rods could be investigated and resolved.	CEA Worth Test * CEAs involved
15	4.3.2.2	W-Pittsburgh	-	1995 (approx.)	Comanche Peak	W	During the startup testing, the reactor engineer determined the current associated with nuclear heating based on pressurizer level increase instead of reactivity decrease due to doppler poisoning. As a result, the testing range was a full decade lower than previous measurement programs. The bank worths were in error by a significant amount due to gamma contamination and had to be remeasured with a lost time of about 12 hours. Subsequently, training packages were modified to address this problem.	CEA Worth Test * Reactivity Computer involved
16	4.3.2.2	INPO-OE9136	-	06/03/98	Plant X	CE	During low power physics testing after startup from a recent refueling outage at Plant X, a control rod manipulation error was made by a Nuclear Control Operator (NCO) which resulted in a single control rod insertion separate from the other rods in the rod group. This event is considered significant as it represents a core reactivity management issue. Analysis has shown that throughout the event, reactivity was maintained within physics testing limits, and adequate shutdown margin existed.	CEA Worth Test * CEAs involved
17	4.3.2.2	LER-01-003	INPO-OE1226	03/01/01	Virgil C. Summer	W	At 0017 hours on March 1, 2001, operations personnel initiated a manual reactor trip when two control rods remained fully inserted during control rod	CEA Worth Test

#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Initiating Test
			6				withdrawal. All systems functioned as designed following the trip. The cause was determined to be the result of demanding control rod insertion beyond the fully inserted position (000 steps). This, in conjunction with geometric factors, placed the control rods in a position where the movable grippers would not properly engage to lift the rod when the lift coil energized.	* CEAs involved
18	4.3.2.2	8211170458	LER	11/08/82	St. Lucie	CE	On 10/18/82 during normal testing at 99% power the Pulse Counting function for CEA 41 was found deleted making the Pulse Counting Position Indication required by Tech. Spec. 3.1.3.3 inoperable. The function was immediately restored. The function is controlled by the plant computer and is one of two position indications available. The CEA Position Indication is believed to have been deleted while setting up for the ITC/MTC test two days earlier. Operators were instructed to run a deletion log following readjustment of CEA positions.	ITC Test * CEA Pulse Counting System
19	4.3.2.2	8201120337	LER 81-050-031-0	12/31/81	St. Lucie Plant, Unit 1	CE	LER 81-050-031-0: On 811201 During Zero Physics Testing surveillance requirement to suspend shutdown margin was not met when rods were not tripped within previous 24-H. Caused by failure to follow procedure. Procedure clarified. Subsequent surveillance confirmed that the rods were trippable before and after the event.	LPPT * CEAs involved
20	4 3.2.2	8403090139	-	01/28/84	Arkansas Unit 2	CE	On 1/28/84 at 0831 Unit 2 tripped from -3x10-3% FP during low power physics testing. The Core Protection Calculators (CPCs) generated an auxiliary trip when the integrated one pin peaking factor limit as calculated by the CPCs was exceeded due to CEA position during testing without the CPCs in bypass. Initial criticality after refueling was achieved using a physics testing procedure which required bypassing the CPC trips to allow certain physics test to be performed. The reactor was manually tripped as part of a physics test procedure. then the reactor was returned to criticality, the normal operations procedure was used. This procedure did not contain the provisions for special test exceptions (CPC bypasses) required for physics testing.	LPPT * CEAs involved
21	4.3.2.2	8509260446 * 8506060662	LER 85-003-01	04/21/85 * 09/19/85	Oconee Unit 2	B&W	LER 85-003-01: On 850421 during Power Escalation after Zero Power Physics Testing Control Rods were positioned beyond Tech Spec Rod Position Index Limit Curve. The rod withdrawal limit Statalarm did not actuate as it should when the limit was being approached, therefore, the violation was not discovered until 2250 hours (15 vs 5 % power) when the alarm did actuate. Boration of RCS was then started. Boron and control rod position were previously set to meet Tech. Spec. requirements for control rod position and established condition in preparation for Power Escalation Tests as outlined in the ZPPT procedure. Operations were told to continue in their procedures for power escalation. Rod index and boration requirements were not addressed as a prerequisite for power escalation. The	LPPT * CEAs involved

Table A-6 Summary of Industry Test Performance Problems								
#	Problem Area Report Section	Source ID Number	ALT ID Number	Date	Plant	Plant Type	DESCRIPTION	Initiating Test
							cause of the incident was failure of Operations personnel to maintain the control rod group position within the T.S. limits. An additional factor was that the ZPPT procedure did not address the rod index or boration requirements for power escalation subsequent to the test. Therefore, this is considered a Test Process Error. W-850919 Ltr.	
22	4.3.2.2	INPO SER-13-95 * 9505180052	-	03/23/95 * * 05/05/95	Catawba Unit 1	W	SER-13-95: On March 23, 1995, Catawba Unit 1 experienced a positive reactivity excursion while performing zero power reactor physics testing, causing reactor power to increase to approximately 3.5 percent power. During control rod reactivity worth testing, the test coordinator misinterpreted test instrumentation readings when the output range of the reactivity computer was exceeded and directed a reactor operator to continuously withdraw a shutdown (reference) rod bank for approximately 50 seconds. Because the reactor operator and test coordinator did not properly monitor nuclear instrumentation during rod bank withdrawal, a startup rate of approximately three decades per minute (DPM) (a reactor period of approximately nine seconds) was unknowingly established. *9505180052: Notice of Violation from insp on 950305-0408. Violation noted on 950323 during performance of control rod worth measurements Requirements of Procedure PT-O-A-4150-11B had not been properly implemented. Specifically, reactivity exceeded +40 pcm during performance of Control Rod Worth Measurements by Rod Swap, which resulted in over-ranging the reactivity instrumentation and misleading indication of actual reactivity. Misleading indication of reactivity contributed to actions which caused an inadvertent reactor power increase.	CEA Worth Test * Reactivity Computer involved
23	4.3.2 3	8103200582	IR	12/19/80	Haddam Neck Plant, Connecticut Yankee Atomic Power	W	Inspection Report: The calculations used to determine the MTC had misapplication of uncertainty applied. The inspector recalculated the MTC and verified it to be within the technical specification limits. The error is an example of inadequate review and approval.	MTC Surveillance
24	4.3.2.3	8609240136	LTR	09/16/86	Zion Nuclear Power Station, Unit 1 Zion Nuclear Power Station, Unit 2	W	The licensee responds to concerns raised by an inspector. Specifically, the inspector notes that test procedure steps were skipped and procedure steps were not signed prior to proceeding to the next step. There were also errors noted in some calculations. The licensee will introduce more training and some revision to procedures.	MTC Surveillance
25	4.3.2.3	8805030261 8804010014	LTR	04/19/88	Virgil C. Summer Nuclear Station, Unit 1	W	Letter: NRC found incorrect application of temperature scaling factor in the calculation of MTC. The licensee recalculated the MTC and verified it to be within the technical specification limits. More plant computer data from the startup was analyzed and reduced in order to revivify the acceptable results.	MTC Surveillance

Table A-7 Methods That Have Detected Past Industry Design Prediction Problems
(Number of Events Identified)[1]

PROBLEM	SECTION[1]	DETECTION METHOD																						Total by Problem
		CEA Drop Time	CEA Drop Characteristics	CEA Flux Change	CBC	IBW	CEA Worth	ITC	MTC Surveillance	SDM Surveillance	CEA Flux Symmetry	Incore Flux Symmetry	Incore Power Distribution	Low Power Physics Tests	Core Design QA	Fuel Fabrication QA	CEA Fabrication QA	Receipt Inspection	EOC CEA Insertion	CEA Manipulation	CEA Inspections	CEA Position Indication	CEA Trip	
CEA Worth Inaccuracy	4.1.2.1						3																	3
CBC Inaccuracy	4.1.2.2				1																			1
ITC Inaccuracy	4.1.2.3																							-
Power Distribution Inaccuracy	4.1.2.4																							-
Total by Detection Method	All	-	-	-	1	-	3	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	4

[1] Definitions and discussions of these problems are provided in the indicated report sections.

Table A-8 Methods That Have Detected Past Industry As-Built Core Problems
(Number of Events Identified)[1]

PROBLEM	SECTION[1]	DETECTION METHOD: CEA Drop Time	CEA Drop Characteristics	CEA Flux Change	CBC	IBW	CEA Worth	ITC	MTC Surveillance	SDM Surveillance	CEA Flux Symmetry	Incore Flux Symmetry	Incore Power Distribution	Low Power Physics Tests	Core Design QA	Fuel Fabrication QA	CEA Fabrication QA	Receipt Inspection	EOC CEA Insertion	CEA Manipulation	CEA Position Indication	CEA Inspections	CEA Trip	Total by Problem
CEA Worth Error	4.2.2.1																							-
CBC Error	4.2.2.2																							-
ITC Error	4.2.2.3																							-
Power Distribution Error	4.2.2.4											1												1
MTC Noncompliance	4.2.2.5								30															30
SDM Noncompliance	4.2.2.6														1									1
Fuel Fabrication Error	4.2.2.7												3			8		3						14
Fuel Misloading	4.2.2.8											2	2		1									5
Fuel Distortion	4.2.2.9	3																		2		1	2	8
Fuel Poison Loss	4.2.2.10																							-
Fuel Crudding	4.2.2.11												5											5
CEA Fabrication Error	4.2.2.12																1							1
CEA Misloading	4.2.2.13																							-
CEA Uncoupling	4.2.2.14										4	1	1								2			8
CEA Distortion	4.2.2.15																		1	1		10		12
CEA Absorber Loss	4.2.2.16																		1	1		2		4
CEA Finger Loss	4.2.2.17												1							1		2		4
RCS Anomaly	4.2.2.18																							-
RCS B-10 Depletion	4.2.2.19																							-
Total by Detection Method	All	3	-	-	-	-	-	-	30	-	4	4	12	-	2	8	1	3	2	5	2	15	2	93

[1] Definitions and discussions of these problems are provided in the indicated report sections.

Table A-9 Tests That Have Initiated Past Test Performance Problems
(Number of Events Identified)

PROBLEM	SECTION[1]	TEST														
		CEA Drop Time	CEA Drop Characteristics	CEA Flux Change	CBC	IBW	CEA Worth	ITC	MTC Surveillance	SDM Surveillance	CEA Flux Symmetry	Incore Flux Symmetry	Incore Power Distribution	Low Power Physics Tests[2]	Power Ascension Tests	Total by Problem
Test Equipment Error	4.3.2.1						6				1			5		12
Test Process Error	4.3.2.2						6	1						3		10
Test Result Error	4.3.2.3								3							3
Total by Test		-	-	-	-	-	12	1	3	-	1	-	-	8	-	25

[1] Definitions and discussions of these problems are provided in the indicated report sections.
[2] These occurred during startup tests but an individual test was not identified.

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APPENDIX B

REVIEW OF STARTUP TESTS

B.1 INTRODUCTION

B.1.1 Background

This appendix analyzes startup test data from Participating Plants to support the elimination of select startup test measurements. A large database that includes measurement results from multiple cycles for Participating Plants as well as some nonparticipating CE Plants is used to characterize the deviations between measurements and best estimate (BE) predictions for CEA worth and ITC. BE predictions are the predictions from core design methods corrected for the bias between past measurements and predictions. The distribution, variability, and poolability of the data are used to justify the elimination of the CEA worth and ITC measurements at HZP.

B.1.2 Purpose

The purpose of this appendix is to review recent startup test results to determine if the comparisons of measurement to prediction support the elimination of the CEA worth and ITC measurements at HZP from the Generic Program.

B.2 METHODS

B.2.1 Analysis of Data Distribution

The distribution of the data is a measure of the relative frequency with which deviations of a particular magnitude occur in the data. [

] A normal distribution is assumed when calculating a 95/95 tolerance and performing the tests used for poolability. [

] This conclusion is based on the deviations being normally distributed with a mean near zero. A normal distribution is characteristic of random errors in measurements and predictions. Although the individual uncertainties cannot be determined using the data, the uncertainty derived from benchmarking provides appropriate conservatism for either BE predictions or measurements because both the measurement and prediction uncertainty are present in the deviations. A zero mean indicates that systematic errors are not present in the data after correcting the prediction for any bias between prediction and measurement. A normality test is used to check the data for consistency with a normal distribution and the mean of the distribution is compared to the uncertainty from previous benchmarking to determine if a significant bias is present. A deviation of the mean from zero represents a potential change in the bias from previous benchmarking, and the significance of any change is determined by comparing the mean to the uncertainty.

Normality tests were conducted for all parameters and all data subsets. The data subsets include data for different operating conditions and core design methods. The Shapiro-Wilk's W-test for normality is performed for sample size $n<50$. The test calculates the quantity:

$$b = \sum_{i=1}^{k} a_{n-i+1}(x_{n-i+1} - x_i)$$

In which: $k = n/2$ if n is even or $k = (n-1)/2$ if n is odd.
x_i are the sorted residuals in increasing order
a_n are coefficients which can be found in Reference B-1.

Calculate also: $S^2 = (n-1)\ s^2$ where s^2 is the unbiased estimate of the population variance.
Finally calculate the test statistic:

$$W = b^2/S^2$$

In order for the test to pass, W must be greater than a critical value W_{crit} which can be found in Reference B-1.

The D' test for normality is used if the population size n>50. Calculate S^2 as in the previous test.
Then calculate:

$$T = \sum_{i=1}^{n} \{i - (n+1)/2\} x_i$$

In which x_i are the sorted residuals in increasing order.
The test statistic is:

$$D' = T / S$$

The test passes if: $D'_{crit}(0.01) < D' < D'_{crit}(0.99)$

B.2.2 Analysis of Data Variability

The variability of the data is characterized by standard deviations and 95/95 tolerances. [

] Uncertainties derived from previous benchmarking were based on 95/95 tolerances. Consistency between 95/95 tolerances derived from recent startup test data and uncertainties derived from previous benchmarking verifies the continued applicability of the uncertainties. [

]

B.2.3 Analysis of Data Poolability

The poolability of data subsets indicates whether the subsets all belong to the same population. [

] The poolability of data subsets is demonstrated using the Bartlett test. The Bartlett test for homogeneity of variances is designed to test for equality of variances across groups against the alternative that variances are unequal for at least two groups. The most common conclusion of the Bartlett test is that the test value is less than the critical Chi Squared value at the 5% significance level, indicating that the assumption of poolability can not be rejected. Thus all data subsets exhibit the same variability in the data and can correctly be pooled.

Having defined the subsets, the test shows that each subset is part of the total distribution. The test consists of evaluating the quantity:

$$\chi^2_{K-1} = \frac{\nu_e \ln S_p^2 - \sum_{i=1}^{K} \nu_i \ln S_i^2}{1 + \frac{1}{3(K-1)}\left(\sum_{i=1}^{K} \frac{1}{\nu_i} - \frac{1}{\nu_e}\right)}$$

in which K subsets have each ν_i degrees of freedom and a variance S_i^2, with a pooled variance:

$$S_p^2 = \frac{\sum_{i=1}^{K} \nu_i \; S_i^2}{\nu_e}$$

and a total degree of freedom:

$$\nu_e = \sum_{i=1}^{K} \nu_i$$

B.3 RESULTS

This section analyzes the distribution, variability, and poolability of deviations in startup test data for the following parameters:

- CEA worth
- ITC

The deviations in startup test data are the differences between the values measured for the parameters and the corresponding BE predictions. BE predictions using the following modern PWR methods are included in the data:

- DIT/ROCS
- PHOENIX/ANC
- CASMO/(SIMULATE or XGT or PRISM)

The source of the DIT/ROCS data includes data used in benchmarking and data used to justify the elimination of the EOC MTC surveillance in some CE Plants. The source of the PHOENIX/ANC data includes data used in benchmarking CE Plants. In addition, recent data from multiple cycles for Participating Plants as well as some nonparticipating CE Plants was included that involved predictions using DIT/ROCS, PHOENIX/ANC, and CASMO/(SIMULATE or XGT or PRISM)[19]. This data covers a wide range of core designs that include significant variations in fuel management, fuel enrichment, poison type, poison loading, and exposure. This data also reflects changes that have occurred as core designs have evolved with time.

[19] In some instances predictions were obtained from core design methods that were not the licensed methods for the particular plant and were not used in the startup tests to verify core design predictions

B.3.1 Review of CEA Bank Worth Data

Tables B-1, B-2, and B-3 provide the CEA bank worth data for DIT/ROCS, PHOENIX/ANC, and CASMO/(SIMULATE or XGT or PRISM) respectively. The CEA bank worth is the worth of an individual CEA group.

Figure B-1 provides a plot of CEA bank worth data for DIT/ROCS along with the uncertainty derived from previous benchmarking. Figure B-2 provides a plot of recent CEA bank worth data for DIT/ROCS, PHOENIX/ANC, and CASMO/(SIMULATE or XGT or PRISM) along with the 95/95 tolerance derived from the combined data.

B.3.1.1 Bank Worth Data Distribution

A normality test was performed on the CEA bank worth data in Figures B-1 and B-2 that confirmed the CEA bank worth data is consistent with a normal distribution. This included the individual subsets of data for each method in Figure B-2 as well as the combined data. The mean value of the recent DIT/ROCS data in Figure B-1 is [ ] which is small compared to the DIT/ROCS uncertainty of [ ] derived from previous benchmarking indicating a significant bias is not present. [ ]

B.3.1.2 CEA Bank Worth Data Variability

The preponderance of recent DIT/ROCS data in Figure B-1 is less than the uncertainty indicating consistency with previous benchmarking. [ ] In addition, the 95/95 tolerance derived from the combined DIT/ROCS, PHOENIX/ANC, and CASMO/(SIMULATE or XGT or PRISM) data in Figure B-2 is [ ] which is similar to the DIT/ROCS uncertainty [ ] derived from previous benchmarking. Although the DIT/ROCS uncertainty is somewhat larger, the difference is partly a result of fewer data points[20] being used to calculate the DIT/ROCS uncertainty. Also, the magnitude of the deviations varies with the worth of the measured banks. The deviations for low bank worths increase substantially because the measurement uncertainty causes the deviations expressed as a percentage of the predicted value to increase. When considering these influences on the calculated tolerances, [ ]

B.3.1.3 CEA Bank Worth Data Poolability between Methods

A poolability test was performed on the following subsets of CEA bank worth data in Figure B-2 that represent different modern core design methods:

- DIT/ROCS
- PHOENIX/ANC
- CASMO/(SIMULATE or XGT or PRISM)

[20] Fewer data points results in a larger 95/95 tolerance for the same standard deviation.