NRC PDR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 7, 1981

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U.S. NUCLEAR REGULATORY COMMISSION

MEMORANDUM FOR: L. Rubenstein, Assistant Director for Core and Containment Systems, DSI

FROM: T. Novak, Assistant Director for Operating Reactors, DL

SUBJECT: PHYSICS STARTUP TEST PROGRAM FOR RELOAD PWRS

You requested that I make comments on the March 10, 1981 Memorandum from M. Chatterton through D. Fieno to W. Johnston on the stated subject. I agree that much work has been done on physics startup tests for PWRs and that a review of the concept should be made at this time. However, my conclusion is considerably different than that presented in Ms. Chatterton's Memorandum.

A good program should receive attention at each of the following points:

1. A set of standards to require the necessary startup tests to prove the reload core is loaded correctly and is in agreement with the calculated physics parameters;

2. Detailed review of the licensee's physics startup test program procedures including the acceptance criteria;

3. On-site observation of the startup testing as necessary to insure procedures are followed; and

4. Review of the results of the physics startup testing program.

Please note that the above points include the four items Ms. Chatterton believes all licensees should submit information on for each reload (even for those performed under 10 CFR 50.59). This method seems very repetative and makes work for your staff when it is not necessary. Let me discuss how I believe each of the above points should be handled.

<u>Point 1</u>: The Technical Specifications (TS) for all operating facilities should presently specify the necessary physics startup tests. If the CPB believes that this is not true, then a generic issue exists that requires resolution in the normal licensing manner; i.e. generic letter requesting TS changes, staff review and issuance of license amendments.

<u>Point 2</u>: The review of all operating procedures has, in the past, been the responsibility of IE. Enclosure 1 is the pertinent pages of the March 13, 1981 inspection report for the Calvert Cliffs units. Note that the Region I - based inspector specializing in procedure reviews spent around 32 hours (3 different trips to Calvert Cliffs) reviewing sections of the licensee's Startup Testing Program.

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Point 3: Enclosure 2 provides an index of IE inspection modules and some of the individual modules inspectors perform for each reactor startup following a core reload. Note the detail of the IE procedures and the reliance on acceptance criteria. Also note the Enclosure 3, IE Procedure No. 72700 requirement for an inspector to observe at least five of the eight specific test to be performed.

Point 4: As indicated in the reference memorandum, unofficial (not required by TS or Regulations) reporting requirements have been put on the licensees. This is an unacceptable practice that should be discontinued. TS for all operating plants require reporting of reactivity anomalies and errors discovered in the transient or accident analyses. In addition the record retention TS requires retention of records of reactor tests and experiments for at least five years. Thus, it would be a simple matter for an inspector to review the data during or shortly after the performance of startup tests.

At this time when the staff workload is beyond our capabilities I recommend that since much of the review suggested by the reference memorandum is controlled by TS and presently reviewed by IE inspectors, this entire review area by turned over to IE. I suggest this recommendation be presented at the next NRR/IE interface meeting.

Thomas M. Novak, Assistant Director
for Operating Reactors
Division of Licensing

Enclosures: As stated

ENCLOSURE 1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

MAR 2 0 1981

Docket Nos. 50-317
50-318

Baltimore Gas and Electric Company
ATTN: Mr. A. E. Lundvall, Jr.
Vice President, Supply
P. O. Box 1475
Baltimore, Maryland 21203

Gentlemen:

Subject: Inspection Number 50-317/81-03 and 50-318/81-03

This refers to the routine safety inspection conducted by Mr. W. M. Troskoski of this office on January 26-30, 1981 at Calvert Cliffs Nuclear Power Plant Units 1 and 2 Lusby, Maryland of activities authorized by NRC License Nos. DPR-53 and DPR-69 and to the discussions of our findings held by Mr. W. M. Troskoski with Mr. L. B. Russell of your staff at the conclusion of the inspection.

Areas examined during this inspection are described in the Office of Inspection and Enforcement Inspection Report which is enclosed with this letter. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observations by the inspector.

Based on the results of this inspection, it appears that one of your activities was not conducted in full compliance with NRC requirements, as set forth in the Notice of Violation, enclosed herewith as Appendix A. This item of noncompliance has been categorized into the levels described in the Federal Register Notice (45 FR 66754) dated October 7, 1980. You are required to respond to this letter and in preparing your response, you should follow the instructions in Appendix A.

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and the enclosures will be placed in the NRC's Public Document Room. If this report contains any information that you (or your contractor) believe to be proprietary, it is necessary that you make a written application within 20 days to this office to withhold such information from public disclosure. Any such application must be accompanied by an affidavit executed by the owner of the information, which identifies the document or part sought to be withheld, and which contains a statement of reasons which addresses with specificity the items which will be considered by the Commission as listed in subparagraph (b) (4) of Section 2.790. The information sought to be withheld shall be incorporated as far as possible into a separate part of the affidavit. If we do not hear from you in this regard within the specified period, the report will be placed in the Public Document Room.

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

Region I

Report No. 50-317/81-03
50-318/81-03

Docket No. 50-317
50-318

License No. DPR-53
DPR-69 Priority ________ Category C

Licensee: Baltimore Gas and Electric Company

P.O. Box 1475

Baltimore, Maryland 21203

Facility Name: Calvert Cliffs Units 1 and 2

Inspection at: Lusby, Maryland

Inspection conducted: January 26-30, 1981

Inspectors: W. Troskoski
W. Troskoski, Reactor Inspector — 3/10/81 date signed

Approved by: D. L. Caphton
D. L. Caphton, Chief, Nuclear Support Section No. 1, RO&NS Branch — 3/10/81 date signed

Inspection Summary:

Inspection on January 26-30, 1981 (Combined Inspection Report Nos. 50-317/81-03 and 50-318/81-03)

Areas Inspected: Routine, unannounced inspection of licensee actions on previous items; fuel handling operations, and surveillance testing related to refueling Technical Specifications for Unit 2; startup testing and data reduction for Unit 1; IE Circulars; and, administrative controls. The inspection involved 32 inspector-hours onsite by a region-based NRC inspector.

Results: Of the five areas inspected, no items of noncompliance were identified in four of the areas, one item was found in one area (level 5, failure to follow procedures, Detail 6).

b. Reactivity Control Systems

Through discussions with licensee representatives and review of control room panels and controls, the inspector verified that the limiting conditions of operation for the refueling mode (mode 6) were met in that:

(1) High Pressure Safety Injection Pump No. 23 provided a flow path from the refueling water tank to the Reactor Coolant System (TS 3.1.2.1).

(2) The refueling water tank provided a borated water source that met volume, concentration and temperature requirements (TS 3.1.2.7).

No items of noncompliance were identified.

c. Inverse Multiplication (1/M)

During the fuel loading, the inspector reviewed the sections of Fuel Handling Procedure FH-6 (Rev. 6) that dealt with neutron flux monitoring. Observations of data being taken and 1/M computations being made were conducted on 1/28/81 to verify procedure adherence. Independent 1/M calculations were also made by the inspector as a check on the licensee's calculations. No discrepancies were identified.

4. Startup Testing - Unit 1

Scope

Sections of the licensee's Startup Testing Program were reviewed to verify that the tests were performed in accordance with technically adequate and approved procedures and Technical Specification requirements. Test data were also reviewed to verify that the results meet acceptance criteria.

Findings

The inspector reviewed Post-Startup Test Procedure (PSTP)-2, Unit 1, Cycle 5, Initial approach to Criticality and Low Power Physics Testing, Rev. 1. Data and acceptance criteria were compared for:

(1) Critical boron (ARO, 532°F)

(2) Isothermal Temperature Coefficient

(3) CEA Group Worths

(4) Critical boron (532°F; 5, 4, 3, 2, 1, Full Inserted)

Each was within its defined specifications.

Technical Specification 3.1.1.1 requires that the shutdown margin shall be determined to be $\geq$ 4.3% $\Delta\rho$ before exceeding 5% of rated thermal power. The licensee successfully demonstrated this by meeting the above acceptance criteria parameters of procedure PSTP-2, that were presented in the following Baltimore Gas and Electric documents prepared by Combustion Engineering:

1. BG&E-9676-468, 10/17/80, "Calvert Cliffs Unit 1 Cycle 5 HZP Critical Boron Concentration".

2. BG&E-9676-452, "Calvert Cliffs Unit 1 Cycle 5 Licensee Submittal".

The inspector notes that Baltimore Gas and Electric is to submit a summary report of plant startup and power escalation testing following modifications of the plant due to the new core design. Pending NRC review of this startup report, the inspector has no further questions at this time.

5. IE Circulars

IEC: 80-17, Fuel Pin Damage Due to Water Jet From Baffle Plate Corner, was issued July 23, 1980. This circular identified a fuel pin failure mechanism that has appeared only in certain Westinghouse PWR's. However, it has been distributed to all PWR's since there may be other plant specific designs of the 'as built' core baffle that could contribute to similar fuel pin failures. Recommended actions included (1) determination of core locations that might be subject to water jet impingement upon fuel pins that could potentially be damaged by fretting, (2) examination of fuel pins that were discharged from those locations, or are now at those locations (during the next refueling outage), and (3) take appropriate actions to correct/prevent occurrence of this problem.

The inspector discussed these problems with licensee representatives. These representatives stated that to date, there has been no observed fuel pin damage due to water impingement. Selected fuel assemblies have been discharged and inspected for this specific phenomena during past refuelings, with negative results. The licensee further stated that the fuel vendor, Combustion Engineering, had been contacted when the circular was issued. The fuel vendor indicated to the licensee that fuel pin damage of the kind addressed by the circular had not occurred at any of the C-E plants. When the inspector requested documentation of the licensee - fuel vendor discussions, the licensee's representative stated that they would request a written letter from Combustion Engineering. Based on these discussions, Circular No. 80-17 is closed.

INSPECTION PROCEDURE NUMBER	TITLE	INSPECTION FREQUENCY
61701	Surveillance	R
61702	Surveillance of Core Power Distribution Limits	X+
61703	Calibration of the Local Power Range Monitoring System	X++
61704	APRM (Average Power Range Monitor) Calibration	X++
61705	Incore/Excore Detector Calibration	X
61706	Core Thermal Power Evaluation	X
61707	Determination of Reactor Shutdown Margin	X+++
61708	Isothermal Temperature Coefficient of Reactivity Measurement (PWR)	X
61709	Power Coefficient of Reactivity (PWR)	X
61710	Control Rod Worth Measurement (PWR)	X
61711	Target Axial Flux Difference Calculation (W NSSS)	X++++

+Should also be completed quarterly during operating cycle

RFollowing initial fuel loading and all subsequent refuelings as described in Module 72700

++Should also be completed at mid point of operating cycle

+++Should also be completed following detection of an inoperable control rod

++++Should also be completed following power transients greater than 50% and start up following a unit trip

*Eligible for reduced frequency

SECTION I

INSPECTION OBJECTIVES

1. Verify that the plant is being operated within the licensed power distribution limits.
2. Determine whether the means utilized to confirm operation within these limits have been submitted by the licensee to NRR for review.
3. Verify that changes or alterations to calculational methods are reviewed by the licensee for correctness.

SECTION II

INSPECTION REQUIREMENTS

Complete the portions of this procedure pertaining to the NSSS for the facility being inspected.

A. WESTINGHOUSE NSSS

1. Determine from the licensee which data analysis code is used to process the information obtained by the movable incore instrumentation. Determine whether the analysis code has been submitted to and reviewed by NRR for approval.

2. From a characteristic flux map printout (preferrably >50% power), verify the following:

   a. That the control rod insertions, core power level and burnup at the time of the flux map were part of the input to the code calculations.

   b. That the predicted two-dimensional power distribution analytical data for all fuel sources and fluxes measured in the thimble locations for each axial region in the core are part of the input.

3. Verify from the full flux map printout in item 2 above that all detectors independently traversed some reference calibration instrument tube for that particular flux map. Examine the normalized detector data and verify that the relative set of measured reaction rates (fission rates as seen by detector) for each thimble location following normalization are printed out.

4. Examine the printout in item 2 above and review the predicted versus measured reaction rates data and obtain an explanation from the responsible reactor engineer for any apparent anomalies.

5. Hot Channel Factors (1 month sample)

   a. Verify that the values of the applicable technical specifications hot channel factors calculated by the analysis code and recorded in the reactor engineering logs (or equivalent) are within the prescribed limits.

   b. Ascertain that the calculated values reflect applicable uncertainty and/or penalty factors.

   c. Examine the printout edits for the highest of each of the hot channel factors and verify that the licensee has accounted for all observed anomalies.

6. Axial Flux Differences

   a. Ascertain from the operations log books (or equivalent) that the Axial Flux difference limits are being maintained within their applicable ranges. (1 month sample)

   b. Review a recent load change (>20%) and verify that the axial flux difference and the mechanics associated with the logging of the applicable penalty deviations are in accordance with the requirements.

7. Ascertain from the operations logs (or equivalent) that the Quadrant Power Tilt limits are being observed and no apparent anomalies exist. (1 month sample).

8. Identify primary personnel responsible for the major steps in obtaining the results of the computer analysis code calculations from the initial incore flux map data.

9. Assess the apparent technical competence of the site reactor engineering staff regarding the particular core analysis code being used at the facility, including capabilities and limitations of the method.

10. Examine the licensee's procedures for evaluating changes or alterations to calculational methods.

E. GENERAL ELECTIRC NSSS

1. Examine the data monitored in performance of a recent LPRM (Local Power Range Monitor) calibration and BASE distribution calculations, as well as the results of those calculations as typed out by OD-1, "Whole Core LPRM Calibration and BASE Distributions" on the on-demand typewriter. Investigate alarms, error and other in-process messages that may be typed out during the course of the program.

2. For the printout in item 1, ascertain that the TIP (Traversing Incore Probe) machine normalization factors were properly obtained for all machines by traversing each probe, one at a time, through the common calibration tube.

3. Verify for the item 1 OD-1 printout, that TIP (Traversing Incore Probe) data for all LPRM locations has been accepted by the computer.

4. Verify from a recent P-1 the following:

   a. Conformance with the Linear Heat Generation Rate (LHGR) limit.

   b. If the CMPF (core maximum peaking factor) is above the design value Total Peaking Factor for that class of fuel ascertain that the APRM setpoints were adjusted (as required) by the appropriate amount specified by the Technical Specifications.

   c. Following such an APRM gain adjustment, verify that the "APRM GAF" on the succeeding P-1 reflects such a change.

   d. Examine a P-1 showing several BASE CRIT CODE entries and a high CMPF. Examine a subsequent P-1 once the Base Crit Codes have been cleared by running the necessary TIP traces and note the effect on the CMPF.

5. Examine the OD-6, "Thermal Data in a Specified Fuel Bundle," printout associated with the P-1 selected in item 4 and ascertain that the minimum critical power ratio (MCPR) and the limiting Average Planar Linear Heat Generation Rate (APLHGR) are within their prescribed limits.

6. Examine the adequacy of the licensee's plans for ascertaining operation within licensed limits under circumstances where the process computer is unavailable.

7. Verify, over a one month period, that each time the computer recovered from an outage, OD-15, "Computer Shutdown and Outage Recovery Monitor," was called in.

8. Verify by examining the records of the three most recent LPRM gain changes that an OD-1 or OD-2 was successfully run subsequent to the changes made.

9. Assess the apparent knowledgeability of the site reactor engineering staff regarding the particular core analysis code being used at the facility, including capabilities and limitations of the method.

10. Examine the licensee's procedures for evaluating changes or alterations to calculational methods.

C. COMBUSTION ENGINEERING NSSS

1. Determine from the licensee which data analysis code is used to process the information obtained by the incore instrumentation.

2. From a characteristics flux map printout verify:

   a. That the control rod insertions, core power level and burnup at the time of the flux maps were part of the input to the code calculations.

   b. That the predicted power distribution analytical data for all fuel sources and fluxes in the instrumented locations for each axial region in the core is part of the input.

3. a. For early C-E reactors employing four segment fixed detector strings and no moveable chambers, determine how the readings from the detector strings are intercalibrated.

   b. For later C-E reactors employing five segment fixed detector strings coupled with a traveling detector system, ascertain that the procedure for intercalibration is being followed.

4. a. Ascertain from the core performance logs (or equivalent) that the axial shape index is being maintained within the allowable limits. (1 month sample)

   b. Ascertain that the various uncertainty factors and flux peaking augmentation factors have been included in the setting of the incore detector Local Power Density alarms as required by the Technical Specifications.

5. Hot Channel Factors (1 months sample)

   a. Verify that the values of the applicable Technical Specifications hot channel factors calculated by the analysis code and recorded in the reactor engineering logs (or equivalent) are within the licensed limits.

   b. Ascertain that the calculated values reflect applicable uncertainty and/or penalty factors pertinent to the license.

   c. Examine the printout edits for the highest of each of the hot channel factors calculated from the item 2 flux map and verify that the licensee has accounted for all observed anomalies.

6. Ascertain from the operation logs (or equivalent) that the azimuthal power tilt limits are being observed and that no apparent anomalies exist. (1 month sample)

7. Identify and document the major steps and primary personnel responsibility in the overall process of obtaining the results of the computer analysis code calculations from the initial incore flux map data.

8. Assess the apparent knowledgeability of the site reactor engineering staff regarding the particular core analysis code being used at the facility, including capabilities and limitations of the method.

9. Examine the licensee's procedures for evaluating changes or alterations to calculational methods.

D. BABCOCK AND WILCOX NSSS

1. Determine from the licensee which data analysis code is used to process the information obtained by the incore instrumentation.

2. Obtain a printout of the applicable subroutine [see reference I.5.D.(4) of Section III] and verify:

   a. That the control rod insertions, core power level and burnup at the time of the flux map were part of the input to the code calculations.

   b. That the predicted power distribution analytical data for all fuel sources and fluxes in the instrumented locations for each axial region in the core are part of the input.

3. Verify that the incore detector calibration procedure is being followed.

4. Ascertain from the core performance logs (or equivalent) that the axial power imbalance is being maintained within the licensed limits. (1 month sample)

5. Hot Channel Factors (1 month sample)

   a. Verify that the values of the applicable technical specifications hot channel factors calculated by the analysis code and recorded in the reactor engineering logs (or equivalent) are within the licensed limits.

   b. Verify that the calculated values reflect applicable uncertainty and/or penalty factors pertinent to the license.

   c. Examine the printout edits for the highest of each of the hot channel factors and verify that the licensee has accounted for any apparent anomalies.

6. Verify from the operations logs (or equivalent) that the Quadrant Power Tilt limits are being observed and that no apparent anomalies exist. (1 month sample)

7. Identify and document the major steps and primary personnel responsibility in the overall process of obtaining the results of the computer analysis code calculations from the initial incore flux map data.

8. Assess the apparent knowledgeability of the site reactor engineering staff regarding the particular core analysis code being used at the facility, including capabilities and limitations of the method.

9. Examine the licensee's procedures for evaluating changes or alterations to calculational methods.

Verify that the parameters specified in the most current cycle reload report have been implemented into the computer software and verified by a test case.

SECTION I

INSPECTION OBJECTIVE

Verify that the calculation of core thermal power is technically correct and power level instruments indicate reactor power within prescribed limits.

SECTION II

INSPECTION REQUIREMENTS

NOTE: Inspection requirements for BWRs and PWRs are provided in Part A and Part B of this section respectively.

A. BWR Inspection Requirements

1. Review the licensee's core thermal power evaluation procedure for technical adequacy and review the results for a specific evaluation at >50% power.

   a. Examine the "Core Performance Evaluation" data sheet, or equivalent, and verify that correct units have been used for the various operating parameters used to compute core thermal power, and that the initial conditions required in the plant procedures are adequate and were met.

   b. Where required, verify that physical properties obtained from figures and curves corresponding to specific reactor conditions have been accurately established, properly translated and recorded on the data forms.

   c. Verify that test instruments utilized meet the applicable accuracy and calibration specifications.

   d. Verify the correctness of the licensee's equation. Review the calculations and ascertain the correctness of the results.

   e. Verify power level instrument settings.

2. Verify that the frequency of evaluations is as prescribed by the facility's Technical Specifications. (1 month sample)

3. Independently calculate a heat balance on the nuclear boiler using the licensee's procedure for manual calculations.

B. PWR Inspection Requirements

1. Review the licensee's core thermal power evaluation procedure for technical adequacy and review the results for a specific evaluation at >50% power.

   a. Examine the secondary heat balance data sheet, or equivalent and verify that correct units have been used for the various operating parameters used to compute core thermal power and that the initial conditions required in the plant procedure are adequate and were met.

b. Verify that physical properties obtained from figures and curves corresponding to specific reactor conditions have been accurately established and are properly translated and recorded on the data forms.

c. Verify that the configuration of the Steam Generator Blowdown System is established in the procedure, and during the data acquisition period, was as required by the plant's procedure.

d. Verify that test instruments utilized meet the applicable accuracy and calibration specifications.

e. Verify the correctness of the licensee's equation. Review the calculations and ascertain the correctness of the results.

f. Verify power level instrument settings.

[illegible]. Verify that the frequency of evaluations is as prescribed by the Technical Specifications. (1 month sample)

[illegible]. Independently calculate a secondary heat balance using the licensee's procedure for manual calculations.

SECTION III

INSPECTION GUIDANCE

A. GENERAL BWR GUIDANCE

The thermal power of the reactor core is determined by a heat balance on the nuclear boiler using operating data. Under steady state conditions, the nuclear boiler heat output is obtained as the difference between the total heat removed from the boiler system and the total heat added in the flow streams returning to the boiler.

1a. Operating data normally recorded for core performance evaluation are the following:

- Reactor pressure (psig)
- Feedwater flow (10^6 lbs/hr)
- Control rod drive water flow (gpm)
- Total steam flow (lbs/hr)
- Bypass valve position (% open)
- Control valve position (% open)
- Feedwater temperature (°F)
- Inlet temperature to recirculation pumps (°F)
- Recirculation pump power (MW)
- Jet pump flow (10^6 lbs/hr)
- Core delta-P (psi)
- Cleanup system heat exchanger ΔT (°F)
- Cleanup system flow (gpm)
- Condenser vacuum (in Hg)
- Drive water flow (10^6 lbs/hr)
- Reactor water level (inches of water)
- Gross electrical output (MWe)

- Net electrical output (MWe)
- APRM (Average Power Range Monitor) readings (%)

b. Physical properties of concern are the enthalpies of:

- Feedwater
- Steam
- Jet pumps (i.e., core inlet flow)
- Cleanup system inlet/outlet
- Control rcd drive water

c. Accuracy requirements are normally found in the SAR or Bases of the TS. Witness calibration of process instrumentation if possible.

c. Core thermal power equals the difference between the total energy out and total energy in.

Total energy OUT consists of the sum of the steam energy rate, the cleanup system energy rate and the fixed losses energy rate. Total energy IN consists of the sum of the feedwater energy rate, the recirculation pump energy rate and the control rod drive flow energy rate. In symbol form, the equation is:

$$Q_{core} = (Q_s + Q_{cu} + Q_{fl}) - (Q_{FW} + Q_{RD} + Q_{pumps})$$

Where: Q_s = steam energy rate

Q_{cu} = heat loss in cleanup system

Q_{fl} = miscellaneous fixed heat losses

Q_{FW} = feedwater energy rate

Q_{RD} = rod drive cooling water energy rate

Q_{pumps} = recirculation pumping power input to water

NOTE: Energy rate is the product of mass flow rate times the enthalpy of the flow stream, (for example: $lbs/_{hr}$ X $BTU/_{lbs}$ = $BTU/_{hr}$)

e. Average Power Range Monitors are adjusted to agree with the results of the heat balance as required by the Technical Specifications (TS).

3. The manual calculation should agree within ± 5% of the computer calculated thermal power (MWe).

B. GENERAL PWR GUIDANCE

Thermal power measurements are utilized in the checks and calibrations of the Nuclear Power Range Instrument Channels. In the thermal calibration of the Nuclear Instrumentation System, the reactor power may be obtained either from the plant computer's calorimetric program or by a manual method of calculation. The latter is normally required when the computer program is not working, or to double check results obtained from the computer.

1a. Typical initial conditions for a core thermal power determination are:

- The reactor is critical and in power operation.
- At the desired power level, the plant has been operated for a sufficient length of time to show that steady state operating conditions have been attained.
- Feedwater flow, water levels, and all controllable temperatures and pressures shall remain, as nearly as possible, unchanged throughout the data acquisition period. This can be accomplished by minimizing rod movement and changes in boron concentration.
- Steam generator blowdown may or may not be allowed by the particular procedure.

b. The physical properties normally obtained from the plant curve book are:

- The thermal expansion factor of the feedwater flow nozzle which is plotted versus the feedwater temperature. This parameter is a factor in the determination of the feedwater flow.
- The feedwater density (equivalent to "specific weight")
- Enthalpy of feedwater
- Enthalpy of steam
- Reactor coolant pump power

c. The Steam Generator Blowdown System is designed to continuously process steam generator blowdown flow that could contain radioactive contaminants in the event of a steam generator primary to secondary leak. A consideration of the mass flow and enthalpy of this blowdown flow is necessary in a thermal power evaluation if the system is in operation.

c. Accuracy requirements are normally found in the SAR or Bases of the TS. Witness calibration of process instrumentation if possible.

e. A heat balance across the secondary side of the steam generators is the starting point to determine the core thermal power in a PWR. This heat balance is modified by the following to obtain core thermal power:

- Letdown energy loss
- Reactor Coolant Pump energy input
- Fixed energy losses (radiation)

In symbolic form, the equation is:

$$Q_{core} = (Q_s + Q_{LD} + Q_{BD} + Q_{FL}) - (Q_{FW} + Q_p)$$

Where: Q_s = Steam energy rate

Q_{LD} = Letdown flow energy rate

$*Q_{BD}$ = Steam generator blowdown energy rate

Q_{FL} = Fixed heat losses (supplied by NSSS vendor or determined experimentally)

Q_{FW} = Feedwater energy rate

Q_p = Reactor coolant pump energy input

* Not required for plants with once-through steam generators.

f. Power range nuclear instruments are adjusted to agree with the results of the heat balance as required by the Technical Specifications (TS).

3. The manual calculation should agree within ± 5% of the computer calculated power (MWe).

SECTION I

INSPECTION OBJECTIVES

1. To verify that the licensee is ensuring adequate shutdown margin throughout the operating cycle.

2. Verify that the calculation of the reactor shutdown margin is technically correct and in accordance with the facility's Technical Specifications and procedures.

3. Verify that the SHUTDOWN MARGIN determination has been performed at the frequency required by the plant Technical Specifications.

SECTION II

INSPECTION REQUIREMENTS

NOTE: Inspection requirements for the PWR and BWR are provided in part A and part B of this section respectively.

A. PWR Inspection Requirements

1. Review the licensee's shutdown margin determination procedure for technical adequacy.

2. For a specific shutdown margin determination, verify that the most recent critical conditions prior to the shutdown have been accurately recorded.

3. For the above selected shutdown margin determination, verify that the Core Reactivity change from the most recent critical due to the following factors has been properly obtained:

   a. Reactivity change due to Boron.

   b. Reactivity change due to Full Length Control Rod Banks worth changes as a result of position, boration, etc.

   c. Reactivity change due to Shutdown Bank Rods.

   d. Reactivity change due to Part Length Rods (if in use; either way, verify that their operational status is reflected in the shutdown margin allowance).

   e. Reactivity change due to Temperature.

   f. Reactivity change due to Power.

   g. Reactivity change due to Xenon.

   h. Reactivity change due to Samarium and other fission products.

   i. Reactivity change due to fuel burnup and burnable poison depletion.

4. Examine the total shutdown margin calculation and verify that conditions and actions prescribed by the Technical Specifications are met.

5. Verify that Shutdown Margin calculations have been performed at the frequency specified in the plant's Technical Specifications.

6. Ascertain that changes made in boron concentration as a consequence of the shutdown margin calculation results are properly verified by chemical analysis. (Sample size dependent on frequency of shutdown margin determinations. See guidance for inspection requirement A.5.).

7. Ascertain that changes in shutdown margin due to rod misalignment have been addressed as required by the Technical Specifications.

B. GE-NSSS Inspection Requirements

1. Review the licensee's shutdown margin procedure for technical adequacy.

2. Examine the shutdown margin determination made at the beginning of the current operating cycle and verify that results are in agreement with Technical Specification requirements.

3. Examine the calculations made to determine the amount of control rod withdrawal required to correspond to the specified shutdown margin.

4. Verify that licensee has reviewed all data supplied by the fuel vendor which is utilized in the Shutdown Margin determination.

5. Verify that a shutdown margin determination took place after any recent incidence of a control rod's inability to insert. Ensure that consideration was given to the effects of temperature, Xenon, Samarium and other fission products, burnup and poison depletion on reactivity as appropriate.

6. Examine the licensee's analysis of a condition where the shutdown margin could not be met and evaluate the adequacy of corrective actions that were taken.

SECTION III

INSPECTION GUIDANCE

A. GENERAL PWR GUIDANCE:

1. Minimum shutdown margin as specified in the Technical Specifications is required for the power operating condition, the hot standby shutdown condition and the cold shutdown condition. In all analyses involving reactor trip, the single highest worth Rod Control Assembly is postulated to remain un-tripped in its full-out position.

   Two independent reactivity control systems are provided, namely control rods and soluble boron in the coolant. The control rod system can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no-load. In addition, the control rod system provides the minimum shutdown margin under Condition I, (normal operation and operational transients), events and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the highest worth control rod is stuck out upon trip.

   The boron system can compensate for all reactivity changes due to xenon burnout and buildup, temperature changes from hot shutdown to cold shutdown, fuel burnup, poison depletion, and fission product changes and will maintain the reactor in the cold shutdown condition. Thus, backup and emergency shutdown provisions are provided by a mechanical and a chemical shim control system.

2. Conditions such as boron concentration, full length rod position, part length rod position, reactor average temperature, power level, xenon and samarium concentrations burnup and poison depletions, are parameters contributing to the overall core reactivity and consequently to the determination of total reactivity change from critical conditions to shutdown. As such, a knowledge of these parameters for the most recent critical conditions preceding a shutdown is essential.

3. A calculation of the Total Reactivity Change requires analysis of each of the contributing factors listed in item 3 of the inspection requirements.

   The basic computation performed to determine the reactivity change associated with each parameter is to multiply the reactivity coefficient of each parameter times the parameter's change in going from the most recent critical condition to the shutdown conditions. Reactivity coefficients (or reactivity values) for the various

parameters involved can normally be obtained from curves found in the plant's technical data book. Rod worths and reactivity coefficients will vary with burnup and boron concentrations.

It should be noted that under shutdown conditions, in calculating the reactivity associated with each of the various parameters, negative reactivities imply positive shutdown margin and positive reactivities imply negative shutdown margins.

4. If the available shutdown margin resulting from the total reactivity change is insufficient to meet the Technical Specification, an additional amount of negative reactivity in the form of boration of the reactor coolant system must be added. A calculation of the boron concentration needed to meet the required shutdown margin involves determining the difference between the required shutdown margin and the available shutdown margin. Then, the difference between this result and the differential boron worth at the specific pre-boration conditions is equivalent to the required boration. Finally, the minimum boron concentration to satisfy the required shutdown margin would be equivalent to the addition of the actual boron concentration at the specific conditions plus the calculated boration.

   Some procedures require that calculations be checked by someone other than the one initially obtaining the results. If so, this procedural requisite should be confirmed.

5. Actual shutdown margin calculations are required by Technical Specifications for conditions such as:

   a. Prior to initial operation above 5% Rated Thermal Power after each fuel loading.

   b. After detection of an inoperable control rod.

Consequently, it is possible that only one actual calculation was performed for operating cycles lacking conditions of inoperable control rods. The sample size to satisfy this inspection requirement will therefore depend on the plant specific operating history. It should be noted that current Standard Technical Specifications require that overall core reactivity balances be compared to predicted values at least once per 31 Effective Full Power Days.

B. GENERAL BWR GUIDANCE:

The purpose of the shutdown margin test is to demonstrate that the reactor can be maintained subcritical by the margin specified in the Technical Specifications with the highest worth rod withdrawn and the core in its most reactive condition. Normally the core will be most reactive when Xenon-free with the moderator at cold (20°C) conditions.

The shutdown margin requirement influences reactor design and operation. Following are some of the direct and indirect effects:

a) Ensures the reactor can be made subcritical from all operating conditions.

b) Ensures that postulated reactivity transients are controllab' within acceptable limits.

c) Permits rod withdrawal for maintenance during shutdown.

d) Limits reactivity of reload fuel.

e) Requires careful planning of fuel design and loading arrangement.

1. The procedures for shutdown margin tests will generally fall into one of three broad categories:

   a. Two Rod method - The rod calculated to have the highest worth is fully withdrawn. Either a face adjacent or diagonally adjacent rod is withdrawn to the position calculated to equal the specified shutdown margin. A variation of this method is to continue withdrawing the second rod and perhaps a third rod until the reactor is critical.

   b. In-sequence Critical - The rod calculated to have the highest worth is fully withdrawn. The reactor is then taken critical using a regular rod withdrawal sequence.

   c. Five Rod Critical - The rod calculated to have the highest worth is fully withdrawn. The four surrounding diagonal rods are withdrawn as a bank until the reactor is critical. A variation of this method involves a symmetric group of rods surrounding the highest worth rod being sequentially withdrawn until the reactor is critical.

   The above methods are not intended to be all inclusive and there may be some other variations.

2. Shutdown margin must be demonstrated at the beginning of cycle (BOC). If due to burnable poisons the core reactivity exhibits an increase with exposure (defined as the R-value), an additional increment of shutdown margin equal to this increase must be demonstrated at the BOC.

3. Cold shutdown margin calculations will involve the following:

   a. Location of the highest worth control rod.

   b. The maximum increase in core reactivity with exposure (R-value).

c. Predicted control rod(s) position at critical or when the specified shutdown margin has been inserted.

If rod worth curves supplied by the fuel vendor have been adjusted by the licensee, the reason for and validity of this adjustment should be determined.

4. The data provided by the fuel vendor normally includes:

   a. New core loading pattern.

   b. Location of highest worth control rod.

   c. Rod-worth curves.

   d. Increase in core reactivity with exposure (R-value).

5. Per Technical Specification surveillance requirement.

6. Under circumstances where the shutdown margin cannot be met, several items can be checked for anomalous conditions; for example:

   a. Rod drifting.

   b. Fuel assembly positions.

   c. Water temperature.

   d. Water chemistry (boron carbide tubes may have ruptured).

   e. Manufacturing (make sure manufacturing record matches actual fuel).

In any event, with the shutdown margin less than the license limit, specific action on the part of the licensee is prescribed by the facility's Technical Specifications.

SECTION I

INSPECTION OBJECTIVE

Verify that the measurement of the Isothermal Temperature Coefficient is technically correct and consistent with Technical Specification requirements.

SECTION II

INSPECTION REQUIREMENTS

1. Examine the adequacy of the licensee's procedure for measuring the Isothermal Temperature Coefficient of Reactivity and review the results for the most recent measurement.

   a. Verify that the prerequisites for the measurement as delineated in the procedure were met.

   b. Verify that during the measurement, precautions as may be indicated in the procedure were observed.

   c. Verify that plant conditions during actual measurement correspond to those plant conditions assumed in obtaining the analytical predictions, against which the actual measurements are compared.

   d. Verify that the values obtained for the Isothermal Temperature Coefficient have been correctly determined and are within the upper and lower limits used in the FSAR accident analysis and Technical Specifications.

   e. Verify that the licensee has properly accounted for any observed discrepancies between actual measurements and analytical predictions.

2. Verify that the frequency of measurement of the Isothermal Temperature Coefficient is as prescribed by Technical Specifications.

SECTION III

INSPECTION GUIDANCE

General

The kinetic characteristics of the reactor core determine the response of the core to changing plant conditions or to operator adjustments made during normal operation, as well as the core response during abnormal or accidental transients. These kinetic characteristics are quantified in reactivity coefficients. The reactivity coefficients reflect the changes in the neutron multiplication due to varying plant conditions such as power, moderator or fuel temperatures, or less significantly (in PWRs) due to a change in pressure or void conditions. Reactivity coefficients change during the life of the core and consequently ranges of coefficients are employed in transient analysis to determine the response of the plant throughout life.

The Isothermal Temperature Coefficient of Reactivity represents the combined effect on core reactivity of two discrete components, namely, the Fuel Temperature (Doppler) coefficient and the Moderator Temperature (Density) coefficient. The Fuel Temperature (Doppler) Coefficient is defined as the change in reactivity per degree change in effective fuel temperature and is primarily a measure of the Doppler broadening of U-238 and Pu-240 resonance absorption peaks.

An increase in fuel temperature increases the effective resonance absorption cross sections of the fuel and produces a corresponding reduction in reactivity. The Moderator Temperature (Density) coefficient is defined as the change in reactivity per degree change in the moderator temperature. A decrease in moderator density means less moderation which results in a negative moderator coefficient. The soluble boron used in the reactor as a means of reactivity control also has an effect on the moderator density coefficient, since the soluble boron poison density (boron atoms per unit volume; not ppm) as well as the water density is decreased when the coolant temperature rises. A decrease in the soluble poison density (boron atoms per unit volume) introduces a positive component in the moderator coefficient. Thus, if the concentration of soluble poison (in ppm) is large enough, yielding a high poison density (boron atoms per unit volume), the net value of the coefficient may be positive, thereby potentially necessitating a boron concentration reduction from the all-rods-out boron end point to ensure a negative moderator temperature coefficient.

1. Experimentally, measurement of the Isothermal Temperature Coefficient consists of calculating the slope of a plot of core reactivity versus T_{avg} for various control rod bank configurations. The slope of these curves represents the Isothermal Temperature Coefficient for the specific control rod configuration.

a. Typical prerecuisites for this measurement are the following:

- Maintenance of Reactor Coolant System pressure and temperature within established values.

- Achievement of reactor criticality and specific power level with a given control rod configuration.

- Control of coolant temperature via secondary steam bypass to the condenser or steam dump to the atmosphere.

- Reactor Coolant Pumps are in operation.

- The Pressurizer boron concentration is within the allowable limits compared to the Reactor Coolant System boron concentration.

- Neutron flux and coolant temperature signals have been adequately connected for monitoring and recording.

b. Maintaining the plant operating status as specified in the Technical Specifications is of primary concern. Also, Reactor Coolant boration or dilution should be avoided during the performance of the test.

c. Validation of factors such as the Moderator Temperature Coefficient calculations is obtained by comparison with plant measurements at hot zero power. It is important to clarify whether the Doppler Coefficient contribution has been subtracted from the Isothermal Temperature Coefficient in order to make these comparisons meaningful. Comparisons between predicted and measured parameters should always correspond to equivalent plant conditions (e.g., normalized to consistent control rod bank configurations and boron concentration).

d. In conjunction with 1.c., verify that the slopes calculated from the plots of reactivity versus temperature have been properly determined and the resulting values are within the bounds of the accident analyses in the FSAR. (see Inspection Requirement, 1.c.)

e. Any apparent discrepancies between predictions and actual measurements should be satisfactorily accounted for by the licensee.

2. Determination of the Moderator Temperature Coefficient is typically required by the current Technical Specifications prior to initial operation above 5% of rated thermal power, after each fuel loading. In addition, because of the potential for a positive temperature coefficient near the end-of-core life, within 7 EFPDs after reaching a rated thermal power equilibrium boron concentration of 300 ppm, a measurement is normally required.

SECTION I

INSPECTION OBJECTIVE

Verify that the measurement of the Power Coefficient of Reactivity is technically correct and consistent with Technical Specification (TS) requirements.

SECTION II

INSPECTION REQUIREMENTS

1. Examine the adequacy of the licensee's procedure for measuring the Power Coefficient of Reactivity and the Power Defect, and review the results for the most recent measurements.

   a. Verify that the prerequisites and initial conditions for the measurements as delineated in the procedure were met.

   b. Verify that during the measurements, precautions as indicated in the procedure were observed.

   c. Verify that plant conditions during actual measurement correspond to those required by the procedure and assumed in the analytical predictions.

   d. If these conditions are different from procedural requirements, verify that any relaxations were approved by the responsible personnel and that TS limitations were observed as appropriate.

   e. Verify that the values obtained for the power coefficient and power defect are within the acceptance criteria. Verify the correctness of the calculations.

   f. If the difference between the measured and predicted values exceeds the acceptance criteria, verify that the licensee has accounted for the discrepancy. Verify adequacy of licensee's actions.

2. Verify that Technical Specification limits were met during the test.

SECTION III

INSPECTION GUIDANCE

GENERAL

During power level changes where the effects of xenon can be adequately accounted for, measurements are made of reactor power and the associated reactivity changes. From these results, the power coefficient of reactivity and power defect are determined. The total power coefficient is essentially the result of the combined effect of moderator temperature and fuel temperature changes as the core power level changes. It is expressed in terms of reactivity change per percent power change. These measurements are performed during power escalations at preselected levels or plateaus (such as 30%, 50%, 75%, and 100% power.)

1. One method for calculating the differential power coefficient of reactivity and the integral power defect is by following turbine load demands with the control bank, throughout the range of the programmed load changes. The main turbine is in automatic control and load changes are initiated at the turbine panel in the control room. The reactor is in manual control with T_{avg} maintained coincident with T_{ref} by the manual insertion or withdrawal of the control bank. Thermal power measurements should be performed before and after load changes. From the collected data and subsequent analysis for xenon, the power coefficient and integral power defect can be determined as functions of reactor power.

   a. Typical prerequisites and initial conditions for these measurements are as follows:

      - operational alignment of the neutron monitoring system has been satisfactorily completed.

      - the reactivity computer is installed and operational.

      - the detail reactor power history (power versus time) must be available for the approximately 48 hours prior to the start of the measurement in order to be able to construct the xenon reactivity history over the duration of the measurement.

      - a reactor thermal power measurement was performed prior to load changes.

      - actual rod bank configuration is as required by procedure.

- control of subsystems affecting overall plant transient response are left in automatic (e.g., pressurizer level, steam generator level and steam dump).

b. Typical precautions for this measurement are:

- procedural restrictions on the magnitudes and rates of power level changes should be observed (e.g., typical values are on the order of 1.0% per minute).

- primary system makeup during any power or load change should be avoided.

c. The power coefficient changes with core burnup, reflecting the combined effect of moderator and fuel temperature coefficients. As a result, the value of the coefficient (experimental or analytical) will depend on whether the transient of interest is examined at the beginning or end of core life.

d. Typically, deviations from these procedures require as a minimum, concurrence of the lead test engineer and the shift supervisor.

e. A typical Power Coefficient Calculation sheet would contain the following parameters:

- initial and final core thermal power
- average power level, $\overline{P}$
- initial and final xenon reactivity
- Reactor Coolant System boron concentration
- reactivity involved in control rod bank position changes

To perform the desired calculations, these parameters would combine as follows:

i. Average Power, $\overline{P} = \text{Initial Power} + \dfrac{\Delta P}{2}$

where $\Delta P = \text{Final Power} - \text{Initial Power}$

ii. $\text{Power Defect} = -\left(\Sigma\Delta\rho_{rods} + \Delta\rho_{xenon}\right)$

where: $\Sigma\Delta\rho_{rods}$ = the summation of all reactivity changes associated with changing rod bank configurations from one position to another.

$\Delta\rho_{xenon}$ = final xenon reactivity minus initial xenon reactivity.

iii. $$\text{Power Coefficient*} = \frac{\text{Power Defect}}{|\Delta P|}$$

$|\Delta P|$ = absolute value of power change. Actual power change, ΔP, may be + or - depending on whether final power is > or < than initial power. However, $|\Delta P|$ is always +.

*The value obtained is unique to the specific average power, $\overline{P}$ and RCS boron concentration at the time the measurements were made.

Since induced changes in power level were achieved by [illegible] changes in control bank positions, it should be noted that by plotting the change in xenon corrected reactivity (caused by rod bank position change) as a function of time, together with plotting the change in power level as a function of time, one can determine the power coefficient from ratios of these two plots for corresponding time intervals, i.e.

$$\text{Power Coefficient} = \frac{\Delta\rho/\Delta t}{\Delta P/\Delta t}$$

The numerator, $\Delta\rho/\Delta t$, would be obtained from a reactivity computer trace for a given Δt interval, and the denominator $\Delta P/\Delta t$ is obtained by determining, through calorimetric calculations, how the core power varies, as a function of time, during the corresponding time interval, Δt, selected on the reactivity computer trace. (Ideally, points on the reactivity versus time trace used for the numerator should be beyond the initial prompt response portion of the curve.)

f. Discrepancies between predictions and actual measurements should be satisfactorily accounted for by the licensee.

2. The test procedure should clearly identify relevant TS relaxation(s) if any, and highlight those requirements that are pertinent to the expected plant configurations (e.g., limitations on hot channel factors and allowable power distributions should be observed at all times).

SECTION II

INSPECTION REQUIREMENTS

NOTE: The specific inspection requirements that follow are applicable for test conditions involving both boron addition and boron dilution measurements of control rod worth. Differences, if any, are indicated in Section III of this procedure under the corresponding guidance for the specific line item requirement of this section.

1. Examine the adequacy of the licensee's procedure for measuring the differential and integral control rod worths during boron addition and/or dilution, and review the results for the most recent measurements.

   a. Verify that the prerequisites and initial conditions for the measurements as delineated in the procedure were met.

   b. Verify that during the measurements, precautions as indicated in the procedure, were observed.

   c. Verify that plant conditions during actual measurement correspond to those plant conditions required by the procedure and assumed in the analytical predictions.

   d. If these conditions are different from procedural requirements, verify that any relaxations were approved by the responsible personnel and that TS limitations were observed as appropriate.

   e. Verify that the values obtained for the control rod worths are within the acceptance criteria. Verify correctness of the calculations.

   f. Verify that the Reactor Coolant System and pressurizer were sampled for boron concentration as required to determine boron worth during control bank movement. (Typical sampling frequencies are at 15 minute intervals).

   g. Verify that the licensee has properly accounted for any discrepancies between actual measurements and expected results.

2. Verify that Technical Specification limits were observed during the measurements.

SECTION III

INSPECTION GUIDANCE

GENERAL

The reactivity worth of each bank is typically measured with the reactor critical at hot zero power. Rods may be "diluted into or borated out of" the core while their worth is recorded by a reactivity computer. The computer will solve the "INHOUR" equation using a power range nuclear instrumentation system channel as input. Control rod worth measurements can also be performed without a reactivity computer by correlating the reactivity associated with a change in boron concentration between two rod configurations, and the reactivity inserted or withdrawn according to the difference in rod configurations between the two states. In the case of dilution, primary grade water is injected into the Reactor Coolant System and the reactivity insertion caused by boron dilution is compensated for by insertion of the controlling bank until the reactor is again critical. The reactor coolant temperature and pressure are maintained constant througnout the test. The procedure is typically performed with each control bank as the controlling bank, thus obtaining an integral reactivity worth for each of the control banks.

In the case of boron addition, with the reactor critical at hot zero power, borated water is injected into the Reactor Coolant System and the negative reactivity caused by the boron injection is compensated for by withdrawal of the controlling bank. As for the dilution case, reactor coolant temperature and pressure are maintained constant throughout the test. The typical range of allowable values for RCS temperature is 542-549°F, maintaining the actual temperature within ±1°F of the selected value. For RCS pressure, the typical value is 2235 ± 25 psig. Typically, integral reactivity worths are obtained for each bank separately (operation without normal overlap) and for the control banks utilizing normal bank withdrawal sequences (with overlap).

1. Experimentally, rod reactivity worth curves are obtained by plotting some appropriate form of the output of the reactivity computer versus rod bank height. For differential worth curves, differential bank worth, $\Delta\rho/\Delta h$, is plotted versus bank height, h. For integral rod worth curves, the integral bank worth, $\Sigma\Delta\rho$, is plotted versus bank height, h. The procedure for these calculations should provide for periodic recording of parameters such as reactor thermal power, reactor coolant system temperature and pressure and boron concentration, and pressurizer boron concentration. Consideration of these parameters is essential in the determination of rod worths.

   a. Typical prerequisites and initial conditions for these measurements are as follows:

      - Operational alignment of the neutron monitoring system has been satisfactorily completed.

- The reactivity computer, if required, is installed and operational.

- Chemistry support is available to sample the reactor coolant system and pressurizer for boron concentration as required.

- The reactor is critical and stable, at the preestablished temperature and at zero power, with the neutron flux in the range established for zero power physics tests.

- The rod banks are in their required configuration, and the rod control switch is in its predetermined position as indicated by procedure (e.g., use of the MANUAL mode to move rods during the separate bank, no overlap measurement of rod bank worths should invalidate the results).

b. The following represent typical precautions to be observed during these measurements:

- A limitation on the maximum start-up rate allowed during the test.

- Close adherence to the prescribed values of temperature and pressure of the reactor coolant system throughout the test. (Refer to last paragraph of GENERAL GUIDANCE section for typical values of RCS temperature and pressure).

- Adherence to the neutron flux and reactor power limits established for zero power physics tests.

- Separate, no overlap bank movement for the portion of measurement yielding independent bank reactivity worths.

- Clear awareness and understanding of special TS requirements during the test (e.g., Group Height and Rod Insertion Limits) and positive adherence to unrelaxed requirements such as hot channel factor and thermal power limitations.

c. The total rod bank configuration for each of these measurements should correspond to the analytical configurations used for the predicted reactivity worth of the rod banks. For independent bank worth measurements, it is important that the bank overlap mode of operation not be used if the results are to be valid.

d. Typically, changes and/or deviations from the test procedure require, as a minimum, agreement of the head test engineer and the shift supervisor.

e. The calculations involved in rod worth measurements deal primarily with two parameters, namely, reactivity and rod position. By maintaining a record of the reactivities calculated by the reactivity

SECTION I
INSPECTION OBJECTIVE

1. Verify that testing is conducted in accordance with approved procedures.

2. Verify that facility is being operated in conformance with NRC requirements and licensee procedures.

computer as a function of initial and final rod positions, plots of differential bank worth ($\Delta\rho/\Delta h$) versus bank position (h) and integral bank worth ($\Sigma\Delta\rho$) versus bank position (h) may be obtained. These results are then compared to the acceptance criteria which reflect the analytical predictions.

f. Boron concentration analysis of the reactor coolant system and pressurizer is necessary in order to determine RCS boron worth during control bank movement. Boron samples should be marked with the time they were taken and the sample points. Results of the analyses should be logged.

g. Apparent discrepancies between predictions and actual measurements should be satisfactorily accounted for by the licensee.

2. The test procedure should clearly highlight those TS requirements that are pertinent to the expected plant configurations.

SECTION II
INSPECTION REQUIREMENTS

1. Observe at least three of the following tests for BWR's or five for PWR's and verify that they were performed in accordance with technically adequate and approved procedures and Technical Specification requirements. Verify by record review that the remainder of the tests were conducted.

   a. Boiling Water Reactors

      (1) Control Rod Drive Scram Time Tests
      (2) NI response to Rod Movement and any reactivity coefficients measured
      (3) Core Power Distriburion Limits (Procedure 61702)
      (4) Calibration of Local Power Range Monitors (Procedure 61703)
      (5) APRM Calibration (Procedure 61704)
      (6) Core Thermal Power Evaluation (Procedure 61706)
      (7) Determination of Reactor Shutdown Margin (Procedure 61707)

   b. Pressurized Water Reactor Prior to Criticality

      (1) Rod drive and rod position indication checks
      (2) Reactor themocouple/RTD Cross Calibration

   c. PWR's After Criticality

      (1) Core Power Distribution Limits (Procedure 61702)
      (2) Incore/Excore Calibration (Procedure 61705)
      (3) Core Thermal Power Evaluation (Procedure 61706)
      (4) Determination of Reactor Shutdown Margin (Procedure 61707)
      (5) Isothermal Temperature Coefficient (Procedure 61708)
      (6) Power Coefficient of Reactivity Measurement (Procedure 61709)
      (7) Control Rod Worth Measurement (Procedure 61710)
      (8) Target Axial Flux Difference Calculation, W-NSSS (Procedure 61711)

2. Review the test data for all tests identified in Item 1 and verify the results meet acceptance criteria and that all deficiencies are resolved in a timely manner.

SECTION III
INSPECTION GUIDANCE

General

The time required to complete inspection effort associated with the referenced procedures for Items 1 and 2 will be recorded on the 766 Form with the referenced procedure number identified as the module number inspected. Inspection items which do not have a referenced procedure will also be recorded on the 766 Form with Procedure 72700 identified as the module number inspected.

1. The licensees master outage check list normally identifies the startup tests to be accomplished in connjection with the refueling outage. The verification should include a determination that test procedures are available for each test and that any changes thereto since the previous test have been reviewed and approved by Licensee Management.

2. Within two refuelings, all tests shall be witnessed.