U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

4.3 NUCLEAR DESIGN

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of the assessment of reactor physics, neutronics, and nuclear design

Secondary - None

I. AREAS OF REVIEW

The review of the nuclear design of the fuel assemblies, control systems, and reactor core is carried out to aid in confirming that fuel design limits will not be exceeded during normal operation or anticipated operational transients and that the effects of postulated reactivity accidents will not cause significant damage to the reactor coolant pressure boundary or impair the capability to cool the core and to assure conformance with the requirements of General Design Criteria (GDC) 10, 11, 12, 13, 20, 25, 26, 27, and 28.

The specific areas of review are as follows:

1. Confirmation that design bases are established as required by the appropriate GDC.

2. The areas concerning core power distribution, including the following:

   A. The presentation of expected power distributions including normal and extreme cases for steady-state and allowed load-follow transients and covering a full range of reactor conditions of time in cycle, allowed control rod positions, and

USNRC STANDARD REVIEW PLAN

This Standard Review Plan, NUREG-0800, has been prepared to establish criteria that the U.S. Nuclear Regulatory Commission staff responsible for the review of applications to construct and operate nuclear power plants intends to use in evaluating whether an applicant/licensee meets the NRC's regulations. The Standard Review Plan is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide an acceptable method of complying with the NRC regulations.

The standard review plan sections are numbered in accordance with corresponding sections in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)." Not all sections of Regulatory Guide 1.70 have a corresponding review plan section. The SRP sections applicable to a combined license application for a new light-water reactor (LWR) are based on Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

These documents are made available to the public as part of the NRC's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Individual sections of NUREG-0800 will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience. Comments may be submitted electronically by email to NRR_SRP@nrc.gov.

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possible fuel burnup distributions. Predictions and calculations, by the applicant, are needed and required by the staff.

B. The presentation of the core power distributions as axial, radial, and local distributions and peaking factors to be used in the transient and accident analyses. As discussed in Draft Regulatory Guide (DG-1145), power distributions within fuel pins is also required. These within-pin power distributions are important for pressurized-water reactor (PWR) and boiling-water reactor (BWR) applications as they affect isotopic buildup/burnup. The effects of phenomena such as fuel densification should be included in these distributions and factors.

C. The translation of the design power distributions into operating power distributions, including instrument-calculation correlations; operating procedures and measurements; and necessary limits on these operations.

D. The requirements for instruments, the calibration and calculations involved in their use, and the uncertainties involved in translation of instrument readings into power distributions.

E. Limits and setpoints for actions, alarms, or scram for the instrument systems and demonstration that these systems can maintain the reactor within design power distribution limits.

F. Measurements in previous reactors and critical experiments and their use in the uncertainty analyses and the measurements to be made on the reactor under review, including startup confirmatory tests and periodically required measurements.

G. The translation of design limits, uncertainties, operating limits, instrument requirements, and setpoints into technical specifications.

3. The areas concerning reactivity coefficients. These are:

A. The applicant's presentation of calculated nominal values for the reactivity coefficients, such as the moderator coefficient, which involves primarily effects from density changes and takes the form of temperature, void, or density coefficients; the Doppler coefficient; and power coefficients. The range of reactor states to be covered includes the entire operating range from cold shutdown through full power and the extremes reached in transient and accident analyses. It includes the extremes of time in cycle and an appropriate range of control rod insertions for the reactor states. The applicant needs to demonstrate that the coefficients used are conservative. The applicant should provide information on reactivity coefficients in the form of curves covering the full applicable range of the variables. The difference between intra- and inter-assembly moderator coefficients, as discussed in DG-1145, needs to be discussed.

B. The applicant's presentation of uncertainty analyses for nominal values, including the magnitude of the uncertainty and the justification of the magnitude by examination of the accuracy of the methods used in calculations (safety analysis report (SAR) Section 4.3.3), and comparison where possible with reactor

experiments. For comparisons to experiments, it is important for the applicant to show that the experiments are applicable and relevant.

C. The applicant's combination of nominal values and uncertainties to provide suitably conservative values for use in reactor steady-state analysis (primarily control requirements SAR Section 4.3.2.4), stability analyses (SAR Section 4.3.2.7), and the transient and accident analyses presented in SAR Chapter 15.

4. The areas concerning reactivity control requirements and control provisions. These are:

A. The control requirements and provisions for control necessary to compensate for long-term reactivity changes of the core. These reactivity changes occur because of depletion of the fissile material in the fuel, depletion of burnable poison in some of the fuel rods, and buildup of fission products and transuranic isotopes.

B. The control requirements and provisions for control needed to compensate for the reactivity change caused by changing the temperature of the reactor from the hot zero power condition to the cold shutdown condition.

C. The control requirements and provisions for control needed to compensate for the reactivity effects caused by changing the reactor power level from full power to zero power.

D. The applicant needs to provide tables and discussions on control requirements and provisions at beginning of life (BOL), end of life (EOL), and intermediate times during the fuel cycle.

E. The control requirements and provisions for control needed to compensate for the effects on the power distribution and stability of the high cross-section neutron capture of the fission product nuclide xenon-135.

F. The adequacy of the control systems to assure that the reactor can be returned to and maintained in the cold shutdown condition at any time during operation. The applicant shall discuss shutdown margins (SDM). Shutdown margins need to be demonstrated by the applicant throughout the fuel cycle.

G. The applicant's analysis and experimental basis for determining the reactivity worth of a "stuck" control rod of highest worth and margins for stuck rods.

H. Uncertainties associated with the control rods needs to be considered, including:

i. Manufacturing tolerances
ii. Methods errors
iii. Operation other than planned
iv. Control element absorber depletion
v. Measurement uncertainty in shutdown margin demonstration

I. The provision of two independent control systems.

5. The areas of control rod patterns and reactivity worths. These are:

   A. Descriptions and figures indicating the control rod patterns expected to be used throughout a fuel cycle. This includes operation of single rods or of groups, or banks of rods, rod withdrawal order, and insertion limits as a function of power and core life.

   B. Descriptions of allowable deviations from the patterns indicated above, such as for misaligned rods, stuck rods, or rod positions used for spatial power shaping.

   C. Descriptions, tables, and figures of the maximum worths of individual rods or banks as a function of position for power and cycle life conditions appropriate to rod withdrawal transients and rod ejection or drop accidents. Descriptions and curves of maximum rates of reactivity increase associated with rod withdrawals, experimental confirmation of rod worths or other factors justifying the reactivity increase rates used in control rod accident analyses, and equipment, administrative procedures, and alarms which may be employed to restrict potential rod worths should be included.

   D. Descriptions and graphs of scram reactivity as a function of time after scram initiation and other pertinent parameters, including methods for calculating the scram reactivity.

6. The area of criticality of the reactor during refueling. Discussions and tables giving values of k_{eff} for single assemblies and groups of adjacent fuel assemblies up to the number required for criticality, assuming the assemblies are dry and also immersed in water, are reviewed. The applicant needs to describe the basis for assuming that the maximum stated k_{eff} will not be exceeded.

7. The areas concerning stability. These are:

   A. As per Section C.1.4.3.2.7 in DG-1145, phenomena and reactor aspects that influence the stability of the nuclear reactor will be discussed by the applicant.

   B. Calculations and considerations given to xenon-induced spatial oscillations.

   C. Potential stability issues due to other phenomena or conditions, as presented by the applicant.

   D. Verification of the analytical methods for comparison with measured data.

8. The areas concerning analytical methods. These are:

   A. Descriptions of the analytical methods used in the nuclear design, including those for predicting criticality, reactivity coefficients, burnup, and stability.

   B. The database and/or nuclear data libraries used for neutron cross-section data and other nuclear parameters, including delayed neutron and photoneutron data and other relevant data.

   C. Verification of the analytical methods for comparison with measured data.

9. The areas concerning pressure vessel irradiation. These are:

   A. Neutron flux spectrum above 1 million electron volts (MeV) in the core, at the core boundaries, and at the inside pressure vessel wall.

   B. Assumptions used in the calculations, these include the power level, the use factor, the type of fuel cycle considered, and the design life of the vessel.

   C. Computer codes used in the analysis.

   D. The database for fast neutron cross-sections.

   E. The geometric modeling of the reactor, support barrel, water annulus, and pressure vessel.

   F. Uncertainties in the calculation.

10. Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). For design certification (DC) and combined license (COL) reviews, the staff reviews the applicant's proposed ITAAC associated with the structures, systems, and components (SSCs) related to this SRP section in accordance with SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria." The staff recognizes that the review of ITAAC cannot be completed until after the rest of this portion of the application has been reviewed against acceptance criteria contained in this SRP section. Furthermore, the staff reviews the ITAAC to ensure that all SSCs in this area of review are identified and addressed as appropriate in accordance with SRP Section 14.3.

11. COL Action Items and Certification Requirements and Restrictions. For a DC application, the review will also address COL action items and requirements and restrictions (e.g., interface requirements and site parameters).

    For a COL application referencing a DC, a COL applicant must address COL action items (referred to as COL license information in certain DCs) included in the referenced DC. Additionally, a COL applicant must address requirements and restrictions (e.g., interface requirements and site parameters) included in the referenced DC.

Review Interfaces

Other SRP sections interface with this section as follows:

1. The organization responsible for the review/assessment of nuclear design reviews the thermal, mechanical, and materials design of the fuel system as part of its review responsibility for Standard Review Plan (SRP) Section 4.2.

2. The organization responsible for the review/assessment of nuclear design reviews thermal margins, adequacies of power distribution limits, the effects of corrosion products (crud), and the acceptability of hydraulic loads as part of its review responsibility for SRP Section 4.4.

3. The organization responsible for the review/assessment of nuclear design verifies that BWR standby liquid control systems meet reactivity control requirements for anticipated

transients without scram (ATWS), including the minimum required boron concentration and system flow capability relative to the size of the reactor vessel, as part of its primary review responsibility for SRP Sections 9.3.5 and 15.8.

4. The organization responsible for the review/assessment of nuclear design reviews postulated fuel failures resulting from overheating of cladding, overheating of fuel pellets, excessive fuel enthalpy, pellet/cladding interaction, and bursting as part of its responsibilities in SRP Chapter 15.

5. The organization responsible for the review/assessment of nuclear design verifies compliance with requirements applicable to reactivity accidents (GDC 28) as part of its review responsibility for SRP Sections 15.4.8 and 15.4.9.

In addition, the organization responsible for the review/assessment of nuclear design will coordinate with other organization's evaluations that interface with the overall review of the system as follows:

1. The organization responsible for materials and chemical engineering reviews the neutron-induced embrittlement of the reactor vessel materials as part of its review responsibility for SRP Sections 5.3.1 and 5.3.2.

2. The organization responsible for instrumentation and control (I&C) reviews the adequacy of proposed instrumentation to meet the requirements for maintaining the reactor operating within defined limits as part of its review responsibility for SRP Sections 7.1 through 7.6.

3. The organization responsible for nuclear plant systems verifies that the new fuel will be maintained in a subcritical status during all credible conditions as part of its review responsibility.

The specific acceptance criteria and review procedures are contained in the referenced SRP sections.

II. ACCEPTANCE CRITERIA

Requirements

Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:

1. GDC 10 requires that acceptable fuel design limits be specified that are not to be exceeded during normal operation, including the effects of anticipated operational occurrences.

2. GDC 11 requires that, in the power operating range, the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity.

3. GDC 12 requires that power oscillations that could result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

4. GDC 13 requires provision of instrumentation and controls (I&C) to monitor variables and systems that can affect the fission process over anticipated ranges for normal operation, anticipated operational occurrences and accident conditions, and to maintain the variables and systems within prescribed operating ranges.

5. GDC 20 requires automatic initiation of the reactivity control systems (RCSs) to assure that acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and to assure automatic operation of systems and components important to safety occurs under accident conditions. There are usually primary and secondary independent RCSs.

6. GDC 25 requires that no single malfunction of the RCSs (this does not include rod ejection or dropout) causes violation of the acceptable fuel design limits.

7. GDC 26 requires that two independent RCSs of different design be provided, and that each system have the capability to control the rate of reactivity changes resulting from planned, normal power changes. One of the systems must be capable of reliably controlling anticipated operational occurrences. In addition, one of the systems must be capable of holding the reactor core subcritical under cold conditions.

8. GDC 27 requires that the RCSs have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods.

9. GDC 28 requires that the effects of postulated reactivity accidents neither result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor cause sufficient damage to impair significantly the capability to cool the core.

10. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;

11. 10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.

SRP Acceptance Criteria

Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for the review described in this SRP section. The SRP is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide acceptable

methods of compliance with the NRC regulations.

The following discussion presents less formal criteria and guidelines used in the review of the nuclear design for meeting the relevant requirements of the GDCs identified above.

1. There are no direct or explicit criteria for the power densities and power distributions allowed during (and at the limits of) normal operation, either steady-state or load-following. These limits are determined from an integrated consideration of fuel limits (SAR Section 4.2), thermal limits (SAR Section 4.4), scram limits (SAR Chapter 7), and transient and accident analyses (SAR Chapter 15). The design limits for power densities (and thus for peaking factors) during normal operation should be such that acceptable fuel design limits are not exceeded during anticipated transients and that other limits, such as the 1204°C (2200°F) peak cladding temperature allowed for loss-of-coolant accidents (LOCAs), are not exceeded during design-basis accidents. Consideration must also be made to the effect of coolant temperatures and enthalpy on the fuel and cladding temperatures. The limiting power distributions are then determined such that the limits on power densities and peaking factors can be maintained in operation. These limiting power distributions may be maintained (i.e., not exceeded) administratively (i.e., not by automatic scrams), provided a suitable demonstration is made that sufficient, properly translated information and alarms are available from the reactor instrumentation to keep the operator informed.

   The acceptance criteria in the area of power distribution are that the information presented should satisfactorily demonstrate that:

   A. A reasonable probability exists that the proposed design limits can be met within the expected operational range of the reactor, taking into account the analytical methods and data for the design calculations; uncertainty analyses and experimental comparisons presented for the design calculations; the sufficiency of design cases calculated covering times in cycle, rod positions, load-follow transients, etc.; and special problems such as power spikes due to densification, possible asymmetries, and misaligned rods.

   B. A reasonable probability exists that in normal operation the design limits will not be exceeded, based on consideration of information received from the power distribution monitoring instrumentation; the processing of that information, including calculations involved in the processing; the requirements for periodic check measurements; the accuracy of design calculations used in developing correlations when primary variables are not directly measured; the uncertainty analyses for the information and processing system; and the instrumentation alarms for the limits of normal operation (e.g., offset limits, control bank limits) and for abnormal situations (e.g., tilt alarms for control rod misalignment).

   Criteria for acceptable values and uses of uncertainties in operation, instrumentation numerical requirements, limit settings for alarms or scram frequency and extent of power distribution measurements, and use of ex-core and in-core instruments and related correlations and limits for offsets and tilts, all vary with reactor type. They can be found in staff safety evaluation reports and in appropriate sections of the technical specifications and accompanying bases for reactors similar to the reactor under review. The organization responsible for the review/assessment of nuclear design has enunciated Branch Technical Position CPB 4.3-1 for Westinghouse reactors that employ

constant axial offset control.

Acceptance criteria for power spike models can be found in a NUREG report on fuel densification, and are discussed in Regulatory Guide (RG) 1.126.

Generally, special or newly emphasized problems related to core power distributions will not be a direct part of normal reviews but will be handled in special generic reviews. Fuel densification effects and the related power spiking and the use of uncertainties in design limits are examples of these areas.

2. The only directly applicable GDC in the area of reactivity coefficients is GDC 11, which states "...the net effect of the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity," and is considered to be satisfied in light water reactors (LWRs) by the existence of the Doppler and negative power coefficients. There are no criteria that explicitly establish acceptable ranges of coefficient values or preclude the acceptability of a positive moderator temperature coefficient (MTC) such as may exist in PWRs at beginning of core life.

   The acceptability of the coefficients in a particular case is determined in the reviews of the analyses in which they are used, e.g., control requirement analyses, stability analyses, and transient and accident analyses. The use of spatial effects such as weighting approximations as appropriate for individual transients are included in the analysis reviews. The judgement to be made under this SRP section is whether the reactivity coefficients have been assigned suitably conservative values by the applicant. The basis for that judgment includes the use to be made of a coefficient, i.e., the analyses in which it is important; the state of the art for calculation of the coefficient; the uncertainty associated with such calculations, experimental checks of the coefficient in operating reactors; and any required checks of the coefficient in the startup program of the reactor under review.

3. Acceptance criteria relative to control rod patterns and reactivity worths include:

   A. The predicted control rod worths and reactivity insertion rates must be reasonable bounds to values that may occur in the reactor. These values are used in the transient and accident analyses and judgment as to the adequacy of the uncertainty allowances are made in the review of the transient and accident analyses.

   B. Equipment, operating limits, and procedures necessary to restrict potential rod worths or reactivity insertion rates should be shown to be capable of performing these functions. It is a position of the organization responsible for the review/assessment of nuclear design to require, where feasible, an alarm when any limit or restriction is violated or is about to be violated.

4. There are no specific criteria that must be met by the analytical methods or data that are used by an applicant or reactor vendor. In general, the analytical methods and database should be representative of the state of the art, and the experiments used to validate the analytical methods should be adequate representations of fuel designs in the reactor and encompass a sufficient range of variables and operating conditions.

Technical Rationale

The technical rationale for application of these acceptance criteria to the areas of review addressed by this SRP section is discussed in the following paragraphs:

1. GDC 10 requires that acceptable fuel design limits be specified that are not to be exceeded during normal operation, including the effects of anticipated operational occurrences. The reactor core's nuclear design is one of several key design aspects that ensure fuel design limits will not be exceeded during normal operations. Compliance with GDC 10 significantly reduces the likelihood of fuel failures occurring during normal operations, including anticipated operational occurrences, thereby minimizing the possible release of fission products to the environment.

2. GDC 11 requires that the net effect of prompt inherent nuclear feedback characteristics in the core tends to compensate for rapid increases in reactivity when operating in the power range. The nuclear design of the reactor core establishes the various reactivity coefficient values that produce the desired feedback characteristics. Compliance with GDC 11 causes the reactor core to be inherently safe during power range operations, thus eliminating the possibility of an uncontrolled nuclear excursion.

3. GDC 12 requires that the reactor core and the associated coolant, control, and protection systems be designed to ensure that power oscillations that result in conditions exceeding specified acceptable fuel design limits are not possible, or can be reliably and readily detected and suppressed. Power oscillations within the reactor core may result from conditions such as improper fuel design or loading or improper reactivity control including control rod positioning, coolant flow instabilities, moderator void formation, and instabilities associated with nonhomogeneous reactor coolant density distributions.

   The occurrence of power oscillations can lead to excessive localized power peaking or cyclic thermal fatigue, and may cause fuel design limits to be exceeded. Compliance with GDC 12 provided assurance that the nuclear design of the reactor core will prevent power oscillations that could challenge the integrity of the fuel and cause the possible release of fission products to the environment.

4. GDC 13 requires that I&C be provided to monitor variables and systems that can affect the fission process over normal operating ranges, anticipated operational occurrences, and accident conditions, and to maintain the variables and systems within the prescribed operating ranges. The nuclear design review includes verification that instrumentation and systems, along with the data processing systems and alarms, will reasonably assure maintenance of core power distributions within specified design limits. Compliance with GDC 13 provides assurance that I&C systems can adequately monitor changes in core reactivity and maintain variables that affect core reactivity within designed operating ranges, thus minimizing the possibility of an adverse transient affecting the integrity of the fuel cladding.

5. GDC 20 requires automatic initiation of the RCSs to assure that acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and to assure automatic operation of systems and components important to safety under accident conditions. Review of the nuclear design verifies the adequacy of control systems and setpoints necessary to shut down the reactor at any time during operation. The automatic initiation of control systems during a reactor transient prevents damage to the

nuclear fuel and, in the early stages of a reactor accident, will minimize the extent of damage to the fuel, thus reducing the release of fission products to the reactor coolant system and possibly the environment.

6. GDC 25 requires that no single malfunction of the RCS can cause violation acceptable fuel design limits. The nuclear design review includes verification that no single malfunction of the RCS can causes the fuel design limits to be exceeded. Meeting the requirements of GDC 25 significantly reduces the possibility that a malfunction in the RCS would result in nuclear fuel damage.

7. GDC 26 requires that two independent RCSs of different design be provided. Review of the nuclear design verifies that two independent RCSs exist, and that one system can reliably control core reactivity during normal power changes and anticipated operational occurrences. The review also verifies that one system can hold the core subcritical under cold conditions. Compliance with GDC 26 provides assurance that core reactivity can be safely controlled and that sufficient negative reactivity exists to maintain the core subcritical under cold conditions, thus minimizing the likelihood of fuel damage and the subsequent release of fission products.

8. GDC 27 requires that the RCSs have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods. The nuclear design review verifies that the RCSs provide a movable control rod system and a liquid poison system and that the core has sufficient shutdown margin assuming a stuck rod. Meeting the requirements of GDC 27 provides assurance that the RCS will be designed such that damage to the fuel in the event of an accident will be minimized.

9. GDC 28 requires that the effects of postulated reactivity insertion accidents not result in damage to the reactor coolant pressure boundary nor cause sufficient damage to impair the capability to cool the core. This SRP section reviews the reactivity coefficients and rod worths assumed in the analysis of reactivity insertion events in Chapter 15 of the SRP. Compliance with GDC 28 provides assurance that the second barrier (i.e., the reactor coolant pressure boundary) that prevents the release of fission products to the environment will not be damaged in the event of a reactivity insertion accident were to occur and that core cooling will not be prevented by the structural collapse of fuel in the core.

III. REVIEW PROCEDURES

The reviewer will select material from the procedures described below, as may be appropriate for a particular case.

These review procedures are based on the identified SRP acceptance criteria. For deviations from these acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives provide an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

The review procedures below apply in general to the COL which supersedes the earlier sequential construction permit (CP) and operating license (OL) stage reviews. At the CP stage, parameter values and certain design aspects may be preliminary and subject to change. At the OL stage, final values of parameters should be used in the analysis presented in the SAR. The

review of the nuclear design of a plant is based on the information provided by the applicant in the SAR, as amended, and in meetings and discussions with the applicant and the applicant's contractors and consultants. This review in some cases will be supplemented by independent calculations performed by the staff or staff consultants. Files of audit calculations are maintained by the organization responsible for the review/assessment of nuclear design for reference by the reviewer.

1. The reviewer confirms, as part of the review of specific areas of the nuclear design outlined below, that the design bases, design features, and design limits are established in conformance with the GDCs listed in Subsection II of this SRP section.

2. The reviewer examines the information presented in the SAR to determine that the core power distributions for the reactor can reasonably be expected to fall within the design limits throughout all normal (steady-state and load-follow) operations, and that the instrument systems employed, along with the information processing systems and alarms, will reasonably assure the maintenance of the distributions within these limits for normal operation.

   The review examines the calculation of effective delayed neutron fraction (β_{eff}) and prompt neutron lifetime (1*) and verifies that appropriate values are used in the reactivity accidents reviewed under SRP Sections 15.4.8 and 15.4.9. Regulatory Guide 1.77 provides guidance for calculating effective delayed neutron fraction and prompt neutron lifetime values. The applicant should use spatially dependent methods. It is necessary to ensure that the methods are appropriate and that the cross-section date and other parameters used as input are appropriately parameterized. DG-1145 provides further guidance and additional relevant information to the applicant.

   For a normal review, many areas related to core power distribution will have been examined in generic reviews or earlier reviews of reactors with generally similar core characteristics and instrument systems. A large part of the review on a particular case may then involve comparisons with information from previous application reviews. The comparisons may involve the shapes and peaking factors of normal and limiting distributions over the range of operating states of the reactor, the effects of power spikes from densification, assigned uncertainties and their use, calculation methods and data used, correlations used in control processes, instrumentation requirements, information processing methods, including computer use setpoints for operational limits and alarm limits, and alarm limits for abnormalities such as flux asymmetries.

   An important part of this review, focusing on considerations of operations, covers the relevant sections of the proposed technical specifications where power distributions and related controls such as control rod limits are discussed. Here the instrument requirements, limit settings, and measurement frequencies and requirements are set forth in full detail. The comparison of technical specifications should reveal any differences between essentially identical reactors or any lack of difference between reactors with changed core characteristics. Where these occur the reviewer must assess the significance and validity of the differences or lack of differences. This review and comparison may be supplemented with examinations of related topical reports from reactor vendors, generic studies by staff consultants, and startup reports from operating reactors which contain information on measured power distributions.

3. Some vendor codes do not use reactivity coefficients. When they are used, the reviewer

determines from the applicant's presentations that suitably conservative reactivity coefficients have been developed for use in reactor analyses such as those for control requirements, stability, and transients and accidents. The reviewer examines:

A. The applicability and accuracy of methods used for calculations including the use of more accurate check calculations.

B. The models involved in the calculations, such as the model used for effective fuel temperature in Doppler coefficient analyses.

C. The reactor state conditions assumed in determining values of the coefficients. For example, the PWR MTC to be used in the steamline break analysis is usually based on the reactor condition at end of cycle, with all control rods inserted except the most reactive rod, and the moderator temperature in the hot standby range.

D. The applicability and accuracy of experimental data from critical experiments and operating reactors used to determine or justify uncertainty allowances. Measurements during startup and during the cycle of MTCs and full power Doppler coefficients in the case of PWRs, and results of measurements of transients during startup in the case of BWRs should be examined. As part of the review, comparisons are made between the values and uncertainty allowances for reactivity coefficients for the reactor under review and those for similar reactors previously reviewed and approved. Generally, many essential areas will have been covered during earlier reviews of similar reactors. The reviewer notes any differences in results for essentially identical reactors and any lack of differences for reactors with changed core characteristics, and judges the significance and validity of any differences or lack of differences.

E. The range of MTC values. The MTC should be non-positive over the entire fuel cycle when the reactor is at a significant power level.

F. The appropriateness of reactivity coefficients used in evaluating reactivity accidents reviewed under SRP Sections 15.4.8 and 15.4.9.

4. The review procedures in the area of reactivity control requirements and control provisions are as follows:

A. The reviewer determines that two independent RCSs of different design are provided.

B. The reviewer examines the tabulation of control requirements, the associated uncertainties, and the capability of the control systems, and determines by inspection and study of the analyses and experimental data that the values are realistic and conservative.

C. The reviewer determines that one of the control systems is capable of returning the reactor to the cold shutdown condition and maintaining it in this condition at any time in the cycle. It is necessary that proper allowance must be made for all of the mechanisms that change the reactivity of the core as the reactor is taken from the cold shutdown state to the hot full-power operating state. The reviewer

should determine that proper allowance is made for the decrease in fuel temperature, moderator temperature, and the loss of voids (in BWRs) as the reactor goes from the power operating range to cold shutdown.

D. The reviewer determines that one of the control systems is capable of rapidly returning the reactor to the hot standby (shutdown) condition from any power level at any time in the cycle. This requirement is met by rapid insertion of control rods in all current LWRs. Proper allowance for the highest worth control rod being stuck in the full-out position must be made. In PWRs, operational reactivity control is carried out by movement of control rods and by adjustments of the concentration of soluble poison in the coolant. The reviewer must pay particular attention to the proposed rod insertion limits in the power operating range, to assure that the control rods are capable of rapidly reducing the power and maintaining the reactor in the hot standby condition. This is an important point because the soluble poison concentration in the coolant could be decreased in order to raise reactor power, while the control rods were left inserted so far that in the event of a scram (rapid insertion of control rods), the available reactivity worth of the control rods on full insertion would not be enough to shut the reactor down to the hot standby condition.

E. The reviewer determines that each of the independent RCSs is capable of controlling the reactivity changes resulting from planned, normal power operation. This determination is made by comparing the rate of reactivity change resulting from planned, normal operation to the capabilities of each of the two control systems. Sufficient margin must exist to allow for the uncertainties in the rate.

5. The review procedures in the area of control rod patterns and reactivity worths are:

A. The reviewer determines by inspection and study of the information described in Subsection I.5 of this SRP section that the control rod and bank worths are reasonable. This determination involves evaluation of the appropriateness of the analytical models used, the applicability of experimental data used to validate the models, and the applicability of generic positions or those established in previous reviews of similar reactors.

B. The reviewer determines the equipment, operating restrictions, and administrative procedures that are required to restrict possible control rod and bank reactivity worths, and the extent to which the alarm criterion in Subsection II.3.B of this SRP section is satisfied. If the equipment involved is subject to frequent downtime, the reviewer must determine if alternative measures should be provided or the extent of proposed outage time is acceptable.

C. The reviewer will employ the same procedures as in item 5.A, above, to evaluate the scram reactivity information described in Subsection I.5 of this SRP section. The scram reactivity is a property of the reactor design and is not easily changed, but if restrictions are necessary the procedures in item 5.B, above, can be followed as applicable.

D. The reviewer confirms the appropriateness of control rod reactivity worths used

in the reactivity accident analyses reviewed under SRP Sections 15.4.8 and 15.4.9. Regulatory Guide 1.77 provides guidance for calculating maximum rod worths to be used in evaluating control rod ejection accidents for PWRs. DG-1145 should also be consulted. Relevant experience and information from BWR situations and scenarios must be considered where applicable.

6. The information presented on criticality of fuel assemblies is reviewed in the context of the applicant's physics calculations and the ability to calculate criticality of a small number of fuel assemblies.

7. The reviewer exercises professional judgment and experience to ascertain the following about the applicant's analytical methods:

   A. The computer codes used in the nuclear design are described in sufficient detail to enable the reviewer to establish that the theoretical bases, assumptions, and numerical approximations for a given code reflect the current state of the art.

   B. The source of the neutron cross-sections used in fast and thermal spectrum calculations is described in sufficient detail so that the reviewer can confirm that the cross-sections are comparable to those in the current ENDF/B data files (i.e., ENDF/B-VII) and other sources of nuclear data, such as JENDL and JEFF3, etc. If modifications and normalization of the cross-section data have been made, the bases used must be determined to be acceptable.

   C. The procedures used to generate problem-dependent cross-section sets are given in sufficient detail so that the reviewer can establish that they reflect the state of the art. The reviewer confirms that the methods used for the following calculations are of acceptable accuracy: the fast neutron spectrum calculation; the computation of the uranium-238 resonance integral and correlation with experimental data; the computation of resonance integrals for other isotopes as appropriate (for example, plutonium-240); calculation of the Dancoff correction factor for a given fuel lattice; the thermal neutron spectrum calculation; the lattice cell calculations, including fuel rods, control assemblies, lumped burnable poison rods, fuel assemblies, and groups of fuel assemblies, and calculations of fuel and burnable poison depletion and buildup of fission products and transuranium isotopes.

   D. The gross spatial flux calculations that are used in the nuclear design are discussed in sufficient detail so that the reviewer can confirm that the following items are adequate to produce results of acceptable accuracy: the method of calculation (e.g., diffusion theory, S_n transport theory, Monte Carlo, synthesis); the number of energy groups used; the number of spatial dimensions (1, 2, or 3) used; the number of spatial mesh intervals, when applicable; and the type of boundary conditions used, when applicable.

   E. The calculation of power oscillations and stability indices for diametral xenon reactivity transients, axial xenon reactivity transients, other possible xenon reactivity transients, and non-xenon-induced reactivity transients are discussed in sufficient detail so that the reviewer can confirm for each item that the method of calculation (e.g., nodal analysis, diffusion theory, transport theory, synthesis) and the number of spatial dimensions used (1, 2, or 3) are acceptable.

F. Verification of the database, computer codes, and analysis procedures has been made by comparing calculated results with measurements obtained from critical experiments and operating reactors. The reviewer ascertains that the comparisons cover an adequate range for each item and that the conclusions of the applicant are acceptable.

8. The analysis of neutron irradiation of the reactor vessel may be used in two ways. It may provide the design basis for establishing the vessel material nil-ductility transition temperature as a function of the neutron fluence. Neutron fluence is the time integrated neutron fluence rate (i.e. neutron flux) as expressed in neutrons per square centimeter. Neutron fluence is often represented by the somewhat archaic term "nvt," where "n" is the neutron density, "v" is the velocity, and "t" is the time interval. Or, it may provide the relative flux spectra at various positions between the pressure vessel and the reactor core so that the flux spectra for various test specimens may be estimated. This information is used in determining the reactor vessel material surveillance program requirements and pressure-temperature limits for operation under SRP Sections 5.3.2 and 5.3.3. The organization responsible for the review/assessment of nuclear design reviews the calculational method, the geometric modeling, and the uncertainties in the calculations under SRP Section. The review procedures for pressure vessel irradiation include determinations that:

A. The calculations were performed by suitable radiation transport methods, that are fundamentally more correct and accurate than diffusion methods.

B. The geometric modeling and source distribution is detailed enough to properly estimate the relative flux spectra at various positions from the reactor core boundary to the pressure vessel wall.

C. The peak vessel wall fluence for the design life of the plant is less than 10 neutrons per square centimeter (n/cm^2) for neutrons of energy greater than 1 MeV. If the peak fluence is found to be greater than this value, the reviewers of SRP Sections 5.3.2 and 5.3.3 are notified.

9. For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the final safety analysis report (FSAR) meets the acceptance criteria. DCs have referred to the FSAR as the design control document (DCD). The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC FSAR.

For review of a COL application, the scope of the review is dependent on whether the COL applicant references a DC, an early site permit (ESP) or other NRC approvals (e.g., manufacturing license, site suitability report or topical report).

for the review of ITAAC. The review of ITAAC cannot be completed until after the completion of this section.

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the review and calculations (if applicable) support conclusions of the following type to be included in the staff's safety evaluation report. The reviewer also states the bases for those conclusions.

These areas include codes and calculational methodology used by the applicant in assessing the nuclear design and operational behavior. In particular, reactivity effects and coefficients will be closely assessed.

The applicant has described the computer programs and calculational techniques used to predict the nuclear characteristics of the reactor design and has provided examples to demonstrate the ability of these methods to predict experimental results. The staff concludes that the information presented adequately demonstrates the ability of these analyses to predict reactivity and physics characteristics of the plant.

To allow for changes of reactivity due to reactor heatup, changes in operating conditions, fuel burnup, and fission product buildup, a significant amount of excess reactivity is designed into the core. The applicant has provided substantial information relating to core reactivity requirements for the first cycle and has shown means have been incorporated into the design to control excess reactivity at all times. The applicant has shown that sufficient control rod worth is available to shut down the reactor with at least the currently accepted (_____ %Δk/k) subcritical margin in the hot condition at any time during the cycle with the highest worth control rod stuck in the fully withdrawn position.

On the basis of our review, the staff concludes that the applicant's assessment of reactivity control requirements over the first core cycle is suitably conservative, and that adequate negative worth has been provided by the control system to assure shutdown capability. Reactivity control requirements will be reviewed for additional cycles as this information becomes available.

The staff concludes that the nuclear design is acceptable and meets the requirements of GDC 10, 11, 12, 13, 20, 25, 26, 27, and 28. This conclusion is based on the following:

1. The applicant has met the requirements of GDC 11 with respect to prompt inherent nuclear feedback characteristics in the power operating range by:

   A. Calculating a negative power coefficient of reactivity.

   B. Using calculational methods that have been found acceptable.

   The applicant needs to present reactivity coefficients actually used in transient analyses and safety assessments and show that suitably conservative values are used. The staff has reviewed the Doppler reactivity coefficients in this case and found them to be suitably conservative.

2. The applicant has met the requirements of GDC 12 with respect to power oscillations that could result in conditions exceeding specified acceptable fuel design limits by:

   A. Showing that such power oscillations are not possible and/or can be easily detected and thereby remedied.

   B. Using calculational methods that have been found acceptable.

The staff has reviewed the analysis of these power oscillations in this case and found them to be suitably conservative.

3. The applicant has met the requirements of GDC 13 with respect to provision of I&C to monitor variables and systems that can affect the fission process by:

   A. Providing instrumentation and systems to monitor the core power distribution, control rod positions and patterns, and other process variables such as temperature and pressure.

   B. Providing suitable alarms and/or control room indications for these monitored variables.

4. The applicant has met the requirements for GDC 26 with respect to provision of two independent RCSs of different designs by:

   A. Having a system that can reliably control anticipated operational occurrences.

   B. Having a system that can hold the core subcritical under cold conditions.

   C. Having a system that can control planned, normal power changes.

5. The applicant has met the requirements of GDC 27 with respect to RCSs that have a combined capability in conjunction with poison addition by the emergency core cooling system of reliably controlling reactivity changes under postulated accident conditions by:

   A. Providing a movable control rod system and a liquid poison system.

   B. Performing calculations to demonstrate that the core has sufficient shutdown margin with the highest worth stuck rod.

6. The applicant has met the requirements of GDC 28 with respect to postulated reactivity accidents by (reviewed by the organization responsible for the review/assessment of nuclear design under SRP Sections 15.4.8 or 15.4.9):

   A. Meeting the regulatory position in Regulatory Guide 1.77 for PWRs.

   B. Meeting the fuel enthalpy limit of 1.17 kilojoule per gram (280 calories per gram) for BWRs (same as for PWRs).

   C. Meeting the criteria on the capability to cool the core.

   D. Using calculational methods that have been found acceptable for reactivity insertion accidents.

7. The applicant has met the requirements of GDC 10, 20, and 25 with respect to specified acceptable fuel design limits by providing analyses demonstrating:

   A. That normal operation, including the effects of anticipated operational occurrences, have met fuel design criteria.

B. That the automatic initiation of the RCS assures that fuel design criteria are not exceeded as a result of anticipated operational occurrences and assures the automatic operation of systems and components important to safety under accident conditions.

C. That no single malfunction of the RCS causes violation of the fuel design limits.

For DC and COL reviews, the findings will also summarize the staff's evaluation of requirements and restrictions (e.g., interface requirements and site parameters) and COL action items relevant to this SRP section.

In addition, to the extent that the review is not discussed in other SER sections, the findings will summarize the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable.

V. IMPLEMENTATION

The staff will use this SRP section in performing safety evaluations of DC applications and license applications submitted by applicants pursuant to 10 CFR Part 50 or 10 CFR Part 52. Except when the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the staff will use the method described herein to evaluate conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications docketed six months or more after the date of issuance of this SRP section, unless superceded by a later revision.

Review procedure III.3.E applies to reviews for all new applications.

VI. REFERENCES

1. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

2. 10 CFR Part 50, Appendix A, General Design Criterion 10, "Reactor Design."

3. 10 CFR Part 50, Appendix A, General Design Criterion 11, "Reactor Inherent Protection."

4. 10 CFR Part 50, Appendix A, General Design Criterion 12, "Suppression of Reactor Power Oscillations."

5. 10 CFR Part 50, Appendix A, General Design Criterion 13, "Instrumentation and Control."

6. 10 CFR Part 50, Appendix A, General Design Criterion 20, "Protection System Functions."

7. 10 CFR Part 50, Appendix A, General Design Criterion 25, "Protection System Requirements for Reactivity Control Malfunctions."
8. 10 CFR Part 50, Appendix A, General Design Criterion 26, "Reactivity Control System Redundancy and Capability."

9. 10 CFR Part 50, Appendix A, General Design Criterion 27, “Combined Reactivity Control Systems Capability.”

10. 10 CFR Part 50, Appendix A, General Design Criterion 28, “Reactivity Limits.”

11. 10 CFR Part 52, “Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants.”

12. Draft Regulatory Guide DG-1145, “Combined License Applications for Nuclear Power Plants (LWR Edition),” September 2006.

13. Branch Technical Position CPB 4.3-1, “Westinghouse Constant Axial Offset Control,” July 1975, attached to SRP Section 4.3.

14. NUREG-0085, “The Analysis of Fuel Densification,” July 1976.

15. Regulatory Guide 1.126, “An Acceptable Model and Related Statistical Methods for Fuel Densification,” Rev. 1, March 1976.

15. Regulatory Guide 1.77, “Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors.”

16. M. B. Chadwick, I. P. Oblowzinsk y, M. Herman, N. M. Greene, R. D. McKnight, D. L. Smith, P. G. Young, R. E. MacFarlane, G. M. Hale, S. C. Frank.e, A. C. Kahler, T. Kawano, R. C. Little, D. G. Madland, P. Moller, R. D. Mosteller, P. Page, P. Talou, H. Trellue, M. C. White, W. B. Wilson, R. Arcilla, C. L. Dunford, S. F. Mughabghab, B. Pritychenko, D. Rochman, A. A. Sonzogni, C. R. Lubitz, T. H. Trumbull, J. P. Weinmann, D. A. Brown, D. E. Cullen, D. P. Heinrichs, D. P. McNabb, H. Derrien, M. E. Dunn, N. M. Larson, L. C. Leal, A. D. Carlson, R. C. Block, J. B. Briggs, E. T. Cheng, H. C. Huria, K. S. Kozier, A. Courcella, V. Pronyaev, S. C. van der Marck, “ENDF/B-VII.o: Next Generation Evaluated Nuclear Data Library for Nuclear Science and Technology,” *Nuclear Data Sheets*, Vol. 107, Issue 12, pp. 2931–3118, December 2006.

PAPERWORK REDUCTION ACT STATEMENT

The information collections contained in the Standard Review Plan are covered by the requirements of 10 CFR Part 50 and 10 CFR Part 52, and were approved by the Office of Management and Budget, approval number 3150-0011 and 3150-0151.

PUBLIC PROTECTION NOTIFICATION

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

BRANCH TECHNICAL POSITION CPB 4.3-1
(Currently the responsibility of the organization responsible
for the review/assessment of nuclear design)
WESTINGHOUSE CONSTANT AXIAL OFFSET CONTROL (CAOC)

I. BACKGROUND

In connection with the staff review of WCAP-8185 (17 × 17), the staff reviewed and accepted a scheme developed by Westinghouse for operating reactors that assures that throughout the core cycle, including during the most limiting power maneuvers the total peaking factor, F_Q, will not exceed the value consistent with the LOCA or other limiting accident analysis. This operating scheme, called constant axial offset control (CAOC), involves maintaining the axial flux difference within a narrow tolerance band around a burnup-dependent target in an attempt to minimize the variation of the axial distribution of xenon during plant maneuvers.

Originally (early 1974), the maximum allowable F_Q (for LOCA) was 2.5 or greater. Later (late 1974), when needed changes were made to the emergency core cooling system (ECCS) evaluation model, Westinghouse, in order to meet physics analysis commitments to all its customers at virtually the same time, did a generic analysis (one designed to suit a spectrum of operating and soon-to-be-operating reactors) and showed that most plants could meet the requirements of 10 CFR 50 Appendix K and 10 CFR 50.46 (i.e., 1204°C (2200°F)) peak clad temperature) if the value for F_Q were less than 2.32. Westinghouse also showed that CAOC procedures employing a ± 5-percent target band would limit peak F_Q for each of these reactors to less than 2.32.

The staff recognized at that time, however, that not all plants needed to maintain F_Q below 2.32 to meet fuel acceptance criteria (FAC) or needed to operate within a ± 5-percent band to achieve a value for F_Q less than 2.32. In fact, Point Beach was allowed to operate within a wider band because the Wisconsin Electric Power Company demonstrated to the NRC's satisfaction that the reactors could be maneuvered within a wider band (+6, -9 percent) and still hold F_Q below 2.32. The staff fully expected that in time most plants would have individual CAOC analyses and procedures tailored to the requirements of their plant-specific ECCS analyses.

Therefore, when the staff accepted CAOC, it was not just F_Q value equal to 2.32 and a ± 5-percent bandwidth the staff were approving, but the CAOC methodology. This is analogous to the staff's review and approval of ECCS and fuel performance evaluation models.

The CAOC methodology, which is described in Reference 1, entails (1) establishing an envelope of allowed power shapes and power densities, (2) devising an operating strategy for the cycle which maximizes plant flexibility (maneuvering) and minimizes axial power shape changes, (3) demonstrating that this strategy will not result in core conditions that violate the envelope of permissible core power characteristics, and (4) demonstrating that this power distribution control scheme can be effectively supervised with ex-core detectors.

Westinghouse argues that point 3, in the CAOC methodology above (i.e., demonstrating the core conditions will not violate the envelope of permissible core power characteristics) is achieved by calculating all of the load-follow maneuvers planned for the proposed cycle and showing that the maximum power densities expected are within limits. These calculations are performed with a radial/axial synthesis method that has been shown to predict conservative

power densities when compared to experimental data. While the staff has accepted CAOC on the basis of these analyses, it also requires that power distributions be measured throughout a number of representative (frequently limiting) maneuvers early in cycle life to confirm that peaking factors are no greater than predicted.

Additionally, the staff is sponsoring a series of calculations at Brookhaven National Laboratory to check aspects of the Westinghouse analysis.

The power distribution measurement tests described above will, of course, automatically relate in-core and ex-core detector responses and thereby validate that power distribution control can be managed with ex-core detectors.

II. BRANCH TECHNICAL POSITION

An applicant or licensee proposing CAOC for other than an F_Q value equal to 2.32 and a bandwidth equal to ± 5 percent is expected to provide the following:

1. Analyses of F_Q × power fraction showing the maximum $F_Q(z)$ at power levels up to 100 percent and departure from nucleate boiling performance with allowed axial shapes relative to the design bases for overpower and loss of flow transients. The envelope of these analyses must be shown to be valid for all normal operating modes and anticipated reactor conditions. (See Table 1 of the letter from Westinghouse Electric Corporation to U.S. Nuclear Regulatory Commission, July 16, 1975, for the cases that must be analyzed to form such an envelope.)

2. A description of the codes used, how cross-sections for cycle were determined, and what F_{xy} values were used.

3. A commitment to perform load-follow tests wherein F_Q is determined by taking in-core maps during the transient. (Note: Westinghouse has outlined for both the NRC staff and the Advisory Committee on Reactor Safeguards (ACRS) an augmented startup test program designed to confirm experimentally the predicted power shapes. This program is presented in the Westinghouse Electric Corporation publication on, "Augmented Startup and Cycle 1 Physics Program," WCAP-8575, August 1975 describes this program. The tests will be carried out at several representative - both 15 × 15 and 17 × 17 - reactors. The staff has endorsed these tests as has the ACRS in its June 12, 1975, letter for the Diablo Canyon plant. In addition, for the near term, the staff plans to require that those licensees who propose to depart from the previously approved peaking factor and target bandwidth perform similar tests (the precise ones will be determined on a case-by-case basis) to broaden the confidence in analytical methods by extending the comparison of prediction with measurement to include more and more burnup histories.

III. REFERENCES

1. T. Morita, et al., "Power Distribution Control and Load Following Procedures," WCAP-8385 (proprietary) and WCAP-8403 (nonproprietary), Westinghouse Electric Corporation, September 1974.

2. C. Eicheldinger, Westinghouse Electric Corporation, Letter to D.B. Vassallo, U.S. Nuclear Regulatory Commission, July 16, 1975.

3. K.A. Jones, et al., “Augmented Startup and Cycle 1 Physics Program,” WCAP-8575, Westinghouse Electric Corporation, August 1975.

SRP Section 4.3
Description of Changes

This SRP section affirms the technical accuracy and adequacy of the guidance previously provided in (Draft) Revision 3, dated June 1996 of this SRP. See ADAMS accession number ML052070410.

In addition this SRP section was administratively updated in accordance with NRR Office Instruction, LIC-200, Revision 1, "Standard Review Plan (SRP) Process." The revision also adds standard paragraphs to extend application of the updated SRP section to prospective submittals by applicants pursuant to 10 CFR Part 52.

The technical changes are incorporated in Revision 3, dated March 2007:

Review Responsibilities - Reflects changes in review branches resulting from reorganization and branch consolidation. Change is reflected throughout the SRP.

I. AREAS OF REVIEW

The Areas of Review section incorporates expanded content to update technical information and discuss modern methods and practices. New areas on uncertainties and reactor stability are added.

II. ACCEPTANCE CRITERIA

The Acceptance Criteria section expands the content to update technical information. Some clarifications were added to the attached BTP CPB- 4.3-1 referenced in this section.

III. REVIEW PROCEDURES

The Review Procedures section adds discussion of combined license applications. The text includes wording on reactivity effects and coefficients, neutron fluence, and additional clarifications are made as needed. DG-1145 referenced as relevant to the topics.

IV. EVALUATION FINDINGS

The Evaluation Findings section adds clarifications and discussion of reactivity issues.

V. IMPLEMENTATION

The Implementation section contains changes to the revised references.

VI. REFERENCES

The Reference section is updated and several relevant new references are included.