U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

4.3 NUCLEAR DESIGN

REVIEW RESPONSIBILITIES

Primary - ~~Core Performance Branch (CPB)~~ Reactor Systems Branch (SRXB)[1]

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 10, 11, 12, 13, 20, 25, 26, 27, and 28.

~~The review of the nuclear design under this SRP section, the review of the fuel system design under SRP Section 4.2, the review of the thermal and hydraulic design under SRP Section 4.4, and the review of the transient and accident analyses under the SRP section for Chapter 15 of the applicant's safety analysis report (SAR), are all necessary in order to confirm that the requirements defined above are met.~~[2]

The specific areas of interest in the nuclear design include:

1. Confirmation that design bases are established as required by the appropriate General Design Criteria.

2. The areas concerning core power distribution. These are:

   a. The presentation of expected or possible 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 possible fuel burnup distributions.

   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. 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 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.

   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 (SAR Section 4.3.3), and comparison where possible with reactor experiments.

   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 4.3.2.8), 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 transuranium 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 control requirements and provisions for control needed to compensate for the effects on the power distribution of the high cross-section Xe-135 isotope.

   e. 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.

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

   g. 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 fuel assemblies. 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.

7. 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 data base used for neutron cross-sections and other nuclear parameters.

   c. Verification of the analytical methods by comparison with measured data.

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

   a. Neutron flux spectrum above 1 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 data base for fast neutron cross sections.

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

   f. Uncertainties in the calculation.

~~9. The adequacy of limits on power distribution during normal operation is reviewed in connection with review of the thermal-hydraulic design under SRP Section 4.4.~~[3]

~~10. The adequacy of proposed instrumentation to meet the requirements for maintaining the reactor operating state within defined limits is reviewed under SRP Sections 7.1 through 7.6.~~[4]

Review Interfaces:[5]

The SRXB also performs the following reviews under the SRP sections indicated:

1. The SRXB reviews the thermal, mechanical, and materials design of the fuel system as part of its review responsibility for SRP Section 4.2.[6]

2. The SRXB 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.[7]

3. The SRXB 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.[8]

4. The SRXB 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 review responsibilities in Chapter 15.[9]

5. The SRXB 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.[10]

In addition, the SRXB will coordinate with other branches' evaluations that interface with the overall review of the system as follows:

1. The Materials and Chemical Engineering Branch (EMCB) 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.[11]

2. The Instrumentation and Controls Branch (HICB) 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.[12]

3. The Plant Systems Branch (SPLB) verifies that the new fuel and spent fuel storage facilities will maintain the fuel in a subcritical array during all credible storage conditions as part of its review responsibility for SRP Sections 9.1.1 and 9.1.2.[13]

II. ACCEPTANCE CRITERIA

The acceptance criteria in the area of nuclear design are based on meeting the relevant requirements of ~~the~~ General Design Criteria (GDC) 10-13, 20, and 25-28 which relate~~d~~ to the reactor core and reactivity control systems ~~(Ref. 1)~~[14]. The relevant requirements are as follows:

A. 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.

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

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

D. GDC 13 requires provision of instrumentation and controls 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.

E. GDC 20 requires automatic initiation of the reactivity control systems 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.

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

G. GDC 26 requires that two independent reactivity control systems 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.

H. GDC 27 requires that the reactivity control systems 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.

I. 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.

The following discussions present 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)[15] peak cladding temperature allowed for loss-of-coolant accidents (LOCA), are not exceeded during design basis accidents. 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 excore and incore 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 ~~CPB~~ SRXB[16] has enunciated Branch Technical Position CPB 4.3-1 for Westinghouse reactors which employ constant axial offset control (Reference ~~2~~[17] 12).

Acceptance criteria for power spike models can be found in a staff technical report on fuel densification (Reference ~~3~~[18] 11).

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 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 such as may exist in pressurized water reactors 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 judgment 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 ~~CPB~~ SRXB[19] position 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 data base should be representative of the state of the art, and the experiments used to validate the analytical methods should be adequate and encompass a sufficient range.

Technical Rationale:[20]

The technical rationale for application of the above acceptance criteria to the fuel system design 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 tend 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 which 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 which 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 provides 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 instrumentation and controls 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 instrumentation and controls systems can adequately monitor changes in core reactivity and maintain variables which 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 reactivity control systems 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 shutdown 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 reactivity control system can cause violation of acceptable fuel design limits. The nuclear design review includes verification that no single malfunction of the reactivity control system causes the fuel design limits to be exceeded. Meeting the requirements of GDC 25 significantly reduces the possibility that a malfunction in the reactivity control system would result in nuclear fuel damage.

7. GDC 26 requires that two independent reactivity control systems of different design be provided. Review of the nuclear design verifies that two independent reactivity control systems 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 reactivity control systems 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 reactivity control systems 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 reactivity control system 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 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 review procedures below apply in general to both the 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 analyses presented in the SAR. The review of the nuclear design of a plant is based on the information provided by the applicant in the safety analysis report, as amended, and in meetings and discussions with the applicant and the applicant's~~his~~[21] 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 ~~CPB~~ SRXB[22] 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 reviewer examines the calculation of effective delayed neutron fraction (β_{eff}) and prompt neutron lifetime (l*) 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.[23]

   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, at the OL stage, 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. 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 pressurized water reactor (PWR) moderator temperature coefficient to be used in the steam line 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 moderator temperature coefficients and full power Doppler coefficients in the case of PWRs, and results of measurements of transients during startup in the case of boiling water reactors (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 moderator temperature coefficient (MTC) values. The MTC should be non-positive over the entire fuel cycle when the reactor is critical.[24]

f. The appropriateness of reactivity coefficients used in evaluating reactivity accidents reviewed under SRP Sections 15.4.8 and 15.4.9.[25]

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

   a. The reviewer determines that two independent reactivity control systems 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 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 light water reactors. 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 reactivity control systems 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 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 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.[26]

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. This information is related to information on fuel storage presented in SAR Section 9.1 and reviewed under SRP Sections 9.1.1 and 9.1.2. The reviewer of SRP Section 9.1 assumes that the applicant's criticality calculations have been reviewed by ~~CPB~~SRXB[27] and are acceptable.

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 (Reference ~~4~~13)[28]. 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 U-238 resonance integral and correlation with experimental data; the computation of resonance integrals for other isotopes as appropriate (for example, Pu-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 data base, 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 fluence, nvt. 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. ~~CPB~~SRXB[29] reviews the calculational method, the geometric modeling, and the uncertainties in the calculations under this SRP section. The review procedures for pressure vessel irradiation include determinations that:

a. The calculations were performed by higher order theory than diffusion theory.

b. The geometric modeling 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^{20} n/cm^{230} 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.

For standard design certification reviews under 10 CFR Part 52, the procedures above should be followed, as modified by the procedures in SRP Section 14.3 (proposed), to verify that the

design set forth in the standard safety analysis report, including inspections, tests, analysis, and acceptance criteria (ITAAC), site interface requirements and combined license action items, meet the acceptance criteria given in subsection II. SRP Section 14.3 (proposed) contains procedures for the review of certified design material (CDM) for the standard design, including the site parameters, interface criteria, and ITAAC.[31]

IV. EVALUATION FINDING

The reviewer verifies that sufficient information has been provided and ~~his~~the[32] review supports the following type of evaluation finding, which is to be included in the staff's safety evaluation report:

> 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 that 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 a _____ %~~W~~Δk/k[33] 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, we conclude 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 Doppler coefficient of reactivity, and
>
>   b. Using calculational methods that have been found acceptable.
>
>   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 which 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, and

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 instrumentation and controls 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, and

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

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

   a. Having a system tha~~nt~~[34] can reliably control anticipated operational occurrences,

   b. Having a system that can hold the core subcritical under cold conditions, and

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

5. The applicant has met the requirements of GDC 27 with respect to reactivity control systems 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, and

   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 ~~CPB~~SRXB[35] 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 KJ/g (280 cal/g~~m~~)[36] for BWRs (same as for PWRs),

   c. Meeting the criteria on the capability to cool the core, and

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 reactivity control system 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, and

c. That no single malfunction of the reactivity control system causes violation of the fuel design limits.

For design certification reviews, the findings will also summarize, to the extent that the review is not discussed in other safety evaluation report sections, the staff's evaluation of inspections, tests, analyses, and acceptance criteria (ITAAC), including design acceptance criteria (DAC), site interface requirements, and combined license action items that are relevant to this SRP section.[37]

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

This SRP section will be used by the staff when performing safety evaluations of license applications submitted by applicants pursuant to 10 CFR 50 or 10 CFR 52.[38] Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of 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.[39]

Review procedure III.3.e applies to reviews for all new applications.[40]

VI. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 10, "Reactor Design."~~;~~[41]

2. 10 CFR Part 50, Appendix A, General Design Criterion 11, "Reactor Inherent Protection."~~;~~

3. 10 CFR Part 50, Appendix A, General Design Criterion 12, "Suppression of Reactor Power Oscillations."~~;~~

4. 10 CFR Part 50, Appendix A, General Design Criterion 13, "Instrumentation and Control.";

5. 10 CFR Part 50, Appendix A, General Design Criterion 20, "Protection System Functions.";

6. 10 CFR Part 50, Appendix A, General Design Criterion 25, "Protection System Requirements for Reactivity Control Malfunctions.";

7. 10 CFR Part 50, Appendix A, General Design Criterion 26, "Reactivity Control System Redundancy and Capability.";

8. 10 CFR Part 50, Appendix A, General Design Criterion 27, "Combined Reactivity Control Systems Capability.";

9. 10 CFR Part 50, Appendix A, General Design Criterion 28, "Reactivity Limits."

10. Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors."[42]

113. R. 0. MeyerNUREG-0085[43], "The Analysis of Fuel Densification," NUREG-0085, July 1976.

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

134. M. K. Drake, ed., "Data Formats and Procedures for the ENDF Neutron Cross Section Library," BNL-50274 (ENDF-102), National Neutron Cross Section Center, Brookhaven National Laboratory (1970).

BRANCH TECHNICAL POSITION CPB 4.3-1
(Currently the responsibility of Reactor Systems Branch (SRXB))[44]
WESTINGHOUSE CONSTANT AXIAL OFFSET CONTROL (CAOC)

A. BACKGROUND

In connection with the staff review of WCAP-8185 (17x17), we 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 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 Appendix K and 10 CFR 50.46 (i.e., 1204°C (2200°F)[45] peak clad temperature) if F_Q ~~F~~<[46] 2.32. Also, Westinghouse showed that CAOC procedures employing a ± 5% target band would limit peak F_Q for each of these reactors to less than 2.32.

We recognized at that time, however, that not all plants needed to maintain F_Q below 2.32 to meet fuel acceptance criteria (FAC)[47], or needed to operate within a ± 5% band to achieve F_Q ~~F~~<[48] 2.32. In fact, Point Beach was allowed to operate with a wider band because the Wisconsin Electric Power Company demonstrated to our satisfaction that the reactors could be maneuvered within a wider band (+6,-9%) and still hold F_Q below 2.32. We 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 we accepted CAOC it was not just F_Q = 2.32 and a ± 5% band width we were approving, but the CAOC methodology. This is analogous to our 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 excore detectors.

Westinghouse argues that point 3, above, 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 which has been shown to predict conservative power densities when compared to experiment. While we have accepted CAOC on the basis of these analyses, we have also required 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, we are sponsoring a series of calculations at BNL to check aspects of the Westinghouse analysis.

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

B. BRANCH TECHNICAL POSITION

An applicant or licensee proposing CAOC for other than F_Q = 2.32 and WI = ±5% is expected to provide:

1. Analyses of F_Q x power fraction showing the maximum $F_Q(z)$ at power levels up to 100% and DNB 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 Reference 2 for the cases which 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 incore maps during the transient. (NOTE: Westinghouse has outlined for both the NRC staff and the ACRS an augmented startup test program designed to confirm experimentally the predicted power shapes. This program is presented in a Westinghouse report (Reference[49] 3). The tests will be carried out at several representative--both 15x15 and 17x17--reactors. We have endorsed these tests as has the ACRS in its June 12, 1975 letter for the Diablo Canyon plant. In addition, for the near term we plan to require that those licensees who propose to depart from the previously approved peaking factor and target band width perform similar tests, precisely which ones to be determined on a case-by-case basis, to broaden our confidence in analytical methods by extending the comparison of prediction with measurement to include more and more burnup histories.

C. 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.[50]

2. C. Eicheldinger, Westinghouse Electric Corporation, letter to D. B. Vassallo, U.S. Nuclear Regulatory Commission, July 16, 1975.[51]

3. K. A. Jones, et al., "Augmented Startup and Cycle 1 Physics Program," WCAP-8575, Westinghouse Electric Corporation, August 1975.[52]

SRP Draft Section 4.3
Attachment A - Proposed Changes in Order of Occurrence

Item numbers in the following table correspond to superscript numbers in the redline/strikeout copy of the draft SRP section.

Item	Source	Description
1.	SRP-UDP Format Item, Update PRB names.	Changed PRB name to reflect latest responsibility assignments for SRP section 4.3.
2.	Editorial	This information was relocated to the Review Interface section.
3.	Editorial	Relocated item 9 to the review interface section.
4.	Editorial	Relocated item 10 to the review interface section.
5.	SRP-UDP format item, Reformat Areas of Review	Added Review Interface heading to Areas of Review. Review interfaces did not exist in SRP 4.3, so appropriate interfaces were developed from statements in the areas of review, acceptance criteria, and review procedures subsections.
6.	SRP-UDP format item, Reformat Areas of Review	A review interface with SRP Section 4.2 was relocated from the second paragraph of the existing Areas of Review.
7.	SRP-UDP format item, Reformat Areas of Review	A review interface with SRP Section 4.4 was relocated from the second paragraph of the existing Areas of Review and from paragraph I.9.
8.	SRP-UDP Integration, implementation of ATWS issues in the SRP	Added a review interface describing reviews of BWR standby liquid control system capabilities with respect to requirements of the ATWS rule which have been added in other SRP sections based upon ROCs 119 and 298.
9.	SRP-UDP format item, Reformat Areas of Review	A review interface with SRP Chapter 15 was relocated from the second paragraph of the existing Areas of Review.
10.	SRP-UDP format item, Reformat Areas of Review; II 1327	A review interface with SRP Sections 15.4.8 and 15.4.9 was adapted from paragraph 6 of the Evaluation Findings.

Item	Source	Description
11.	PI 21215, PI 21216	Added SRP Section 5.3.1 to the list of interfacing sections under the primary review of the EMCB. SRP Section 5.3.1 is the primary implementing section for 10 CFR 50, Appendix G and Appendix H requirements which utilize the neutron fluence reviewed in SRP Section 4.3 to determine fracture toughness properties. SRP Section 5.3.3 was deleted since this section provides an integrated review of SRP Sections 5.2.3, 5.2.4, 5.3.1, and 5.3.2, and as such does not directly interface with SRP Section 4.3.
12.	SRP-UDP format item, Reformat Areas of Review	A review interface with SRP Sections 7.1 through 7.6 was adapted from existing specific areas of review in subsection I item 10.
13.	SRP-UDP format item, Reformat Areas of Review	A review interface with SRP Sections 9.1.1 and 9.1.2 was adapted from paragraph 6 of the Review Procedures.
14.	SRP-UDP Format Item, update reference citation.	Revised reference citation to be consistent with SRP-UDP required format that excludes parenthetical notation for CFR and GDC citations. Added the specific GDCs listed as Reference 1 to the text and made minor grammatical corrections.
15.	SRP-UDP format item - NRC Metrication policy implementation.	Converted 2200°F to 1204°C and placed 2200°F in parentheses.
16.	SRP-UDP Format Item, Update PRB names.	Changed PRB name to reflect latest responsibility assignments for SRP section 4.3.
17.	SRP-UDP Format Item, update reference citations.	Revised reference citation to be consistent with SRP-UDP required format that specifies spelling out the word Reference. Changed reference number to agree with changes in the Reference subsection numbering.
18.	SRP-UDP Format Item, update reference citations.	Revised reference citation to be consistent with SRP-UDP required format that specifies spelling out the word Reference. Changed reference number to agree with changes in the Reference subsection numbering.
19.	SRP-UDP Format Item, Update PRB names.	Changed PRB name to reflect latest responsibility assignments for SRP section 4.3.
20.	SRP-UDP format item, Develop Technical Rationale.	Added Technical Rationale heading to Acceptance Criteria subsection and developed Technical Rationale for GDC 10, 11, 12, 13, 20, 25, 26, 27, and 28.

Item	Source	Description
21.	Editorial	Made minor revision to make the SRP gender neutral.
22.	SRP-UDP Format Item, Update PRB names.	Changed PRB name to reflect latest responsibility assignments for SRP section 4.3.
23.	**Integrated Impact 1327**	Added a paragraph to state that the reviewer should review the effective delayed neutron fraction and prompt neutron lifetime values used in the reactivity accident analyses.
24.	**Integrated Impact 512**	Added a paragraph to the Review Procedures to ensure that the moderator temperature coefficient is non-positive.
25.	**Integrated Impact 1327**	Added a paragraph to state that the reviewer should review the appropriateness of reactivity coefficients used in the reactivity accident analyses.
26.	**Integrated Impact 1327**	Added a paragraph to state that the reviewer should review the appropriateness of control rod worths used in the analysis of reactivity accidents.
27.	SRP-UDP Format Item, Update PRB names.	Changed PRB name to reflect latest responsibility assignments for SRP section 4.3.
28.	SRP-UDP Format Item, update reference citations.	Revised reference citation to be consistent with SRP-UDP required format that specifies spelling out the word Reference. Changed reference number to agree with changes in the Reference subsection numbering.
29.	SRP-UDP Format Item, Update PRB names.	Changed PRB name to reflect latest responsibility assignments for SRP section 4.3.
30.	Editorial	Revised the "20" and "2" to be superscripted so that the value would read 10^{20} n/cm^2 and not 1020 n/cm2 so as to correct existing typographical errors.
31.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard paragraph to address application of Review Procedures in design certification reviews.
32.	Editorial	Made minor editorial change to make the SRP gender neutral.
33.	Editorial	Replaced "W" with the delta symbol, "Δ" to indicate the correct units for reactivity.
34.	Editorial.	Replaced the letter "n" with "t" to correct a typographical error.

Item	Source	Description
35.	SRP-UDP Format Item, Update PRB names.	Changed PRB name to reflect latest responsibility assignments for SRP sections 15.4.8 and 15.4.9.
36.	SRP-UDP format item - NRC Metrication policy implementation.	Converted 280 cal/gm to 1.17 KJ/g and placed 280 cal/g in parentheses (Note: the "m" in gm was deleted to be consistent with the federal standard).
37.	SRP-UDP format item, 10 CFR 52 Applicability	Added statement to Evaluation Findings addressing the findings associated with design certification reviews.
38.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard sentence to address application of the SRP section to reviews of applications filed under 10 CFR Part 52, as well as Part 50.
39.	SRP-UDP Guidance	Added standard paragraph to indicate applicability of this section to reviews of future applications.
40.	**Integrated Impact 512**.	Added a statement to specify that review procedure III.3.e applies to new applications only.
41.	Editorial.	Divided the list of GDC into separate citations for each GDC to be consistent with the other sections and renumbered the other references in the list as a result of this change.
42.	**Integrated Impact 1327**	Added RG 1.77 to the list of references.
43.	Editorial.	Changed the citation format of this NUREG to be consistent with the way NUREGs are cited in the other SRP Sections.
44.	Editorial	Added a parenthetical statement that SRXB is currently responsible for this technical position.
45.	SRP-UDP format item - NRC Metrication policy implementation.	Converted 2200°F to 1204°C and placed 2200°F in parentheses.
46.	Editorial	Made revision to correct typographical error in original document.
47.	Editorial	Spelled out the acronym "FAC," and placed it in parenthesis.
48.	Editorial	Made revision to correct typographical error in original document.

Item	Source	Description
49.	SRP-UDP Format Item, update reference citations.	Revised reference citation to be consistent with SRP-UDP required format that specifies spelling out the word Reference.
50.	Unverified Reference.	The reference is a vendor report and could not be verified.
51.	Unverified Reference.	The reference is a vendor letter to the NRC and could not be verified.
52.	Unverified Reference.	The reference is a vendor report and could not be verified.

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Integrated Impact No.	Issue	SRP Subsections Affected
512	Modify Review Procedures to include explicit verification, for evolutionary PWRs, of a non-positive MTC over the entire fuel cycle when the reactor is critical.	III
1327	Modify Review Procedures to include verification that rod worths, reactivity coefficients, and reactor physics values used in the analysis of reactivity accidents are appropriate. Add RG 1.77 to the list of references.	I, III, VI