NUREG-75/087

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 4.3 NUCLEAR DESIGN

REVIEW RESPONSIBILITIES

Primary - Core Performance Branch (CPB)

Secondary - Reactor Systems Branch (RSB)
Electrical, Instrumentation and Control Systems Branch (EICSB)

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.

The review of the nuclear design under this plan, the review of the fuel system design under Standard Review Plan (SRP) 4.2, the review of the thermal and hydraulic design under SRP 4.4, and the review of the accident analyses under the SRP 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.

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 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. The power distributions should include power spikes from fuel densification.

    b. The presentation of the core power distributions as axial, radial, and local distributions and peaking factors to be used in accident analyses.

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.

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 Xe135 isotope.

e. The adequacy of the control systems to insure 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 experiments.

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.

The RSB reviews the adequacy of limits on power distribution during normal operation in connection with their analyses of the thermal-hydraulic design, anticipated operational occurrences, and accidents, under SRP 4.4 and the plans for Chapter 15 of the SAR.

The EICSB, under the plans for SAR Chapter 7, reviews the adequacy of proposed instrumentation to meet the requirements for maintaining the reactor operating state within defined limits.

II. ACCEPTANCE CRITERIA

1. The basic acceptance criteria in the area of nuclear design are the general design criteria (GDC) related to the reactor core and reactivity control systems.

   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 and accident conditions, and to maintain them 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) cause 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.

2. The following discussions present less formal criteria and guidelines used in the review of the nuclear design.

   a. 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 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 2200°F 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.

It is a branch position that 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:

(1) 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 misa.igned rods.

(2) 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 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).

Branch positions on 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 (Ref. 2). The CPB has enunciated a branch technical position for Westinghouse reactors which employ constant axial offset control (Ref. 7).

Acceptance criteria for power spike models can be found in staff technical reports on fuel densification (Ref. 3).

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.

b. 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 or branch positions 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 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 plan 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.

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

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

(2) 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 position to require, where feasible, an alarm when any limit or restriction is violated or is about to be violated.

d. 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.

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 his contractors and consultants. This review in some cases will be supplemented by independent calculations performed by the staff or staff consultants.

1. The reviewer confirms, as part of the reviews of specific areas of the nuclear design outlined below, that the design bases, design features, and design limits specified by the GDC listed in Section II are established in conformance with those GDC.

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.

   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 (Ref. 4). Multigroup computer calculations by the reviewer or staff consultants are not done as a part of the normal review.

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 accidents. The reviewer examines:

   a. The applicability and accuracy of methods used for calculations including the use of more accurate check calculations such as the use of Monte Carlo techniques for Doppler models.

   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 life 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 PWR's, and results of measurements of transients during startup in the case of boiling water reactors (BWR's), 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. In special cases, audit calculations may be performed by the reviewer or staff consultants in specific areas to confirm the applicant's analyses. CPB maintains files of generic audit calculations made by staff consultants, for reference by the reviewer (Ref. 5).

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 BWR's) 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 strongest control rod being stuck in the full-out position must be made. In PWR's, 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 Section I.5 of this plan 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 II.2.c.(2) 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 accepta e.

c. The reviewer will employ the same procedures as in a, above, to evaluate the scram reactivity information described in I.5. The scram reactivity is a property of the reactor design and is not easily changed, but if restrictions are necessary the procedures in b, above, can be followed as applicable.

d. The reviewer or staff consultants may perform check calculations in this area as necessary to complete the review.

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 by the Auxiliary and Power Conversion Systems Branch (APCSB). The APCSB reviewer assumes that the applicant's criticality calculations have been reviewed by CPB and are acceptable. Independent criticality audit calculations may be done by the reviewer or staff consultants as necessary to complete the review.

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 (Ref. 6). 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 fission product buildup.

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., modal 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 reasonable.

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 spectrum for various test specimens may be estimated. This information is used by the Materials Engineering Branch in determining the reactor vessel material surveillance program requirements and pressure-temperature limits for operation. CPB reviews the calculational method, the geometric modeling, and the uncertainties in the calculations under this plan. 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 neutron spectrum 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^2 for neutrons of energy greater than one MeV. If the peak fluence is found to be greater than this value, the Materials Engineering Branch is notified.

IV. EVALUATION FINDING

The reviewer verifies that sufficient information has been provided and his 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 balances 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 _____%k/k subcritical margin in the hot condition at any time during the cycle with the most reactive 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. We also conclude that nuclear design bases, features, and limits have been established in conformance with the requirements of General Design Criteria 10, 11, 12, 13, 20, 25, 26, 27, and 28."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 10, "Reactor Design;" Criterion 11, "Reactor Inherent Protection;" Criterion 12, "Suppression of Reactor Power Oscillations;" Criterion 13, "Instrumentation and Control;" Criterion 20, "Protection System Functions;" Criterion 25, "Protection System Requirements for Reactivity Control Malfunctions;" Criterion 26, "Reactivity Control System Redundancy and Capability;" Criterion 27, "Combined Reactivity Control Systems Capability;" and Criterion 28, "Reactivity Limits."

2. Staff safety evaluation reports and plant technical specifications. Examples of these are:

   a. Safety Evaluation Report, General Electric Standard Safety Analysis Report (GESSAR), Section 4.3, Docket No. STN 50-447, U. S. Atomic Energy Commission, November 1974.

b. Safety Evaluation Report, Combustion Engineering Standard Safety Analysis Report (CESSAR), Section 4.3, Docket No. STN 50-470, U. S. Nuclear Regulatory Commission, to be published.

c. Safety Evaluation Report, Jamesport Nuclear Power Station Units 1 and 2, Section 4.3, Docket No. STN 50-516/517, U. S. Nuclear Regulatory Commission, to be published.

d. Safety Evaluation Report, Greenwood Energy Center Units 2 and 3, Section 4.3, Docket Nos. 50-452/453, U. S. Atomic Energy Commission, July 17, 1974.

e. Technical Specifications, Browns Ferry Nuclear Plant Unit 1 and Unit 2, Sections 2.1 and 3.2 through 3.5, License No. DPR-33 and 52, [illegible] 28, 1974.

f. Technical Specifications, Millstone Point Nuclear Power Station Unit No. 2, Sections 2.1, 3.1, and 3.2, Docket No. 50-336, to be published.

g. Technical Specifications, D. C. Cook Nuclear Plant Unit 1, Sections 2.1 and 3.1 through 3.3, License No. DPR-58, October 25, 1974.

h. Technical Specifications, Arkansas Nuclear One Unit 1, Sections 2.1, 3.1, and 3.5, License No. DPR-51, May 21, 1974.

3. Staff technical reports on fuel densification:

   a. Regulatory Staff, "Technical Report on Densification of Light Water Reactor Fuels," U. S. Atomic Energy Commission, November 14, 1972.

   b. Regulatory Staff, "Technical Report on Densification of Babcock and Wilcox Reactor Fuels," U. S. Atomic Energy Commission, July 6, 1973.

   c. Regulatory Staff, "Technical Report on Densification of Exxon Nuclear BWR Fuels," U. S. Atomic Energy Commission, September 3, 1973.

   d. Regulatory Staff, "Technical Report on Densification of Gulf United Nuclear Fuels Corporation Fuels for Light Water Reactors," U. S. Atomic Energy Commission, November 21, 1973.

   e. Regulatory Staff, "Technical Report on Densification of Westinghouse PWR Fuel," U. S. Atomic Energy Commission, May 14, 1974.

4. Topical and startup test reports which are current and applicable to the reactor under review. Examples of these are:

   a. G. N. Kear and M. J. Ruderman, "An analysis of Methods in Control Rod Theory and Comparison with Experiment," GEAP-3937, General Electric Company, May 1962.

b. J. S. Moore, "Power Distribution Control of Westinghouse PWR's," WCAP-7811, Westinghouse Electric Corporation, December 1971.

c. J. O. Cermak, et al, "Pressurized Water Reactors pH-Reactivity," WCAP-3696-8, Westinghouse Electric Corporation, October 1968.

d. "Surry Power Station - Unit 2, Startup Test Report," Virginia Electric Power Company, July 31, 1973.

e. J. E. Outz, "Plant Startup Test Report, H. B. Robinson Unit No. 2," WCAP-7844, Westinghouse Electric Corporation, January 1972.

f. R. H. Clark and T. G. Pitts, "Physics Verification Experiments, Core I," BAW-TM-455, Babcock and Wilcox Company, June 1966.

g. R. H. Clark, "Physics Verification Experiments, Cores II and III," BAW-TM-458, Babcock and Wilcox Company, July 1966.

h. D. R. Jones and J. G. Harsum, "Field Testing Requirements for Fuel Curtains and Control Rods," NEDO-10017, General Electric Company, June 1969.

i. R. Barry, et al, "Nuclear Design of Westinghouse PWR's with Burnable Poison Rods," WCAP-9000-L, Revision 1, Westinghouse Electric Corporation, June 1969.

j. G. V. Kumar, "Startup Test Results - Dresden NPS Unit 3," NEDC-10692, General Electric Company, December 1972.

k. E. J. Dean, "Quad Cities Units 1 and 2 - Startup Test Results," NEDC-10812, General Electric Company, March 1973.

l. J. D. LeBlanc, "Maine Yankee Atomic Power Station Startup Test Report," Maine Yankee Atomic Power Company, June 1973.

5. Brookhaven National Laboratory interim report files maintained by Core Performance Branch, Task 2, "Moderator Coefficients," and Task 3, "Control Rod Worths."

6. 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).

7. Branch Technical Position CPB 4.3-1, "Westinghouse Constant Axial Offset Control," July 1975, attached to Standard Review Plan 4.3.

BRANCH TECHNICAL POSITION CPB 4.3-1
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 in such a fashion 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 '74), the maximum allowable F_Q (for LOCA) was 2.5 or greater. Later (late '74), 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 CFR 50.46 (i.e., 2200°F peak clad temperature) if $F_Q \leq 2.32$. Also, Westinghouse showed that CAOC procedures employing a $\pm$ 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 FAC, or, needed to operate within a $\pm$ 5% band to achieve $F_Q \leq 2.32$. 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 $\pm$ 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 methodol , which is described in WCAP-8385 (Ref. 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 out-of-core 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

Whenever an applicant or licensee proposes CAOC for other than $F_Q = 2.32$ and $\Delta I = \pm 5\%$ he 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. The details of this program will be disclosed in a soon-to-be-issued WCAP report. 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 Diablo letter. 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. REFERENCE

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

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

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