C.I.4 <u>Reactor</u>

Chapter 4 of the final safety analysis report (FSAR) should provide an evaluation and supporting information to establish the capability of the reactor to perform its safety functions throughout its design lifetime under all normal operational modes, including transient, steady-state, and accident conditions. This chapter should also include information to support the accident analyses provided in Chapter 15.

The agency considers the information required to be submitted in this chapter to be reference, typical, or bounding fuel design information. Before Cycle 1 begins, the applicant can request an amendment to the Tier 2* information related to the fuel system design. The applicant must submit Cycle 1-related core fuel assembly design, control rod assembly design, core loading pattern, and related core parameters (related to Sections C.I.4.3 and C.I.4.4) for approval.

C.I.4.1 *Summary Description*

In this section, the applicant should provide a summary description of the mechanical, nuclear, and thermal-hydraulic designs of the various reactor components, including the fuel, reactor vessel internals, and reactivity control systems. This summary description should indicate the independent and interrelated performance and safety functions of each component. (Information on control rod drive systems (CRDSs) and reactor vessel internals provided in Sections 3.9.4 and 3.9.5 of the FSAR may be incorporated by reference.) In addition, this description should include a summary table of the important design and performance characteristics as well as a tabulation of analysis techniques used and load conditions considered (including computer code names).

C.I.4.2 *Fuel System Design*

The fuel system is defined as consisting of guide tubes or thimbles; fuel rods with fuel pellets, insulator pellets, cladding, springs, end closures, fill gas, and getters; water rods; burnable poison rods; spacer grids and springs; assembly end fittings and springs; channel boxes; and the reactivity control assembly. For the control rods, this section should include the reactivity control elements that extend from the coupling interface of the control rod drive mechanism (CRDM). In addition, applicants should present the design bases for the mechanical, chemical, and thermal designs of the fuel system, which can affect or limit the safe, reliable operation of the plant.

The description of the fuel system mechanical design should include, as a minimum, the following four aspects:

(1) mechanical design limits, such as those for allowable stresses, deflection, cycling, and fatigue

(2) capacity for fuel fission gas inventory and pressure

(3) listing of material properties

(4) considerations for radiation damage, cladding collapse time, materials selection, and normal operational vibration

Section 3.7.3 of the FSAR should provide details for seismic loadings; however, this section should provide shock loadings associated with a loss-of-coolant accident (LOCA) and the effects of combined shock and seismic loads.

The chemical design should consider all possible fuel cladding-coolant interactions.
The description of the thermal design should include such items as maximum fuel and cladding

temperatures, clad-to-fuel gap conductance as a function of burnup and operating conditions, and fuel cladding integrity criteria.

C.I.4.2.1 Design Bases

Applicants should explain and substantiate the selection of design bases from the perspective of safety considerations. Where the limits selected are consistent with proven practice, a referenced statement to that effect will suffice; however, where the limits exceed present practice, this section should provide an evaluation and explanation based on developmental work or analysis. These design bases may be expressed as either explicit numbers or general conditions. In addition, the discussion of design bases should include a description of the functional characteristics in terms of desired performance under stated conditions. This should relate systems, components, and materials performance under normal operating, anticipated transient, and accident conditions. The discussion should consider the following seven aspects with respect to performance:

(1) cladding

- (a) mechanical properties of the cladding (e.g., Young's modulus, Poisson's ratio, design dimensions, strength, ductility, and creep rupture limits) and effects of design temperature and irradiation on those properties
- (b) stress-strain limits
- (c) vibration and fatigue
- (d) chemical properties of the cladding

(2) fuel material

- (a) thermal-physical properties of the fuel (e.g., melting point, thermal conductivity, density, and specific heat) and effects of design temperature and irradiation on those properties
- (b) effects of fuel densification and fission product swelling
- (c) chemical properties of the fuel

(3) fuel rod performance

- (a) analytical models and conservatism in the input data
- (b) ability of the models to predict experimental or operating characteristics
- (c) standard deviation or statistical uncertainty associated with the correlations or analytical models

(4) spacer grid and channel boxes

- (a) mechanical, chemical, thermal, and irradiation properties of the materials
- (b) vibration and fatigue
- (c) chemical compatibility with other core components, including coolant

(5) fuel assembly

- (a) structural design
- (b) thermal-hydraulic design

(6) reactivity control assembly and burnable poison rods

  (a) thermal-physical properties of the absorber material

  (b) compatibility of the absorber and cladding materials

  (c) cladding stress-strain limits

  (d) irradiation behavior of absorber material

(7) surveillance program

  (a) requirements for surveillance and testing of irradiated fuel rods, burnable poison rods, control rods, channel boxes, and instrument tubes/thimbles

C.I.4.2.2 Description and Design Drawings

Applicants should provide a description and final (FSAR) design drawing of the fuel rod components, burnable poison rods, fuel assemblies, and reactivity control assemblies showing arrangements, dimensions, critical tolerances, sealing and handling features, methods of support, internal pressurization, fission gas spaces, burnable poison content, and internal components. In addition, this section should include a discussion of design features that prevent improper orientation or placement of fuel rods or assemblies within the core.

This section should provide the following fuel system information and associated tolerances:

- type and metallurgical state of the cladding
- cladding outside diameter
- cladding inside diameter
- cladding inside roughness
- pellet outside diameter
- pellet roughness
- pellet density
- pellet resintering data
- pellet length
- pellet dish dimensions
- burnable poison content
- insulator pellet parameters
- fuel column length
- overall rod length
- rod internal void volume
- fill gas type and pressure
- sorbed gas composition and content
- spring and plug dimensions
- fissile enrichment
- equivalent hydraulic diameter
- coolant pressure
- design-specific burnup limit

Applicants should also provide the following design drawings:

- fuel assembly cross-section
- fuel assembly outline
- fuel rod schematic

- spacer grid cross-section
- guide tube and nozzle joint
- control rod assembly cross-section
- control rod assembly outline
- control rod schematic
- burnable poison rod assembly cross-section
- burnable poison rod assembly outline
- burnable poison rod schematic
- orifice and source assembly outline

C.I.4.2.3 Design Evaluation

Applicants should provide an evaluation of the fuel system design for the physically feasible combinations of chemical, thermal, irradiation, mechanical, and hydraulic interactions. The evaluation of these interactions should include the effects of normal reactor operations, anticipated operational occurrences, anticipated transients without scram, and postulated accidents. In particular, the fuel system design evaluation should include the following six considerations:

(1) cladding

(a) vibration analysis

(b) fuel element internal and external pressure and cladding stresses during normal and accident conditions, with particular emphasis on temperature transients or depressurization accidents

(c) potential for chemical reaction, including hydriding, fission product attack, and crud deposition

(d) fretting and crevice corrosion

(e) stress-accelerated corrosion

(f) cycling and fatigue

(g) material wastage attributable to mass transfer

(h) rod bowing attributable to thermal, irradiation, and creep dimensional changes

(i) consequences of power-coolant mismatch

(j) irradiation stability of the cladding

(k) creep collapse and creepdown

(2) fuel

(a) dimensional stability of the fuel

(b) potential for chemical interaction, including possible waterlogging rupture

(c) thermal stability of the fuel, including densification, phase changes, and thermal expansion

(d) irradiation stability of the fuel, including fission product swelling and fission gas release

(3) fuel rod performance

(a) fuel-cladding mechanical interaction

(b) failure and burnup experience, including the thermal conditions for which the experience was obtained for a given type of fuel and the results of long-term irradiation testing of production fuel and test specimens

(c) fuel and cladding temperatures, both local and gross, with an indication of the correlation used for thermal conductivity, gap conductance as a function of burnup and power level, and the method of employing peaking factors

(d) an analysis of the potential effect of sudden temperature transients on waterlogged elements or elements with high internal gas pressure

(e) an analysis of temperature effects during anticipated operational transients that may cause bowing or other damage to fuel, control rods, or structure

(f) an analysis of the energy release and potential for a chemical reaction in the event of a physical burnout of fuel elements[1]

(g) an analysis of the energy release and resulting pressure pulse should waterlogged elements rupture and spill fuel into the coolant[1]

(h) an analysis of fuel rod behavior in the event that coolant flow blockage is predicted[1]

(4) spacer grid and channel boxes

(a) dimensional stability considering thermal, chemical, and irradiation effects
(b) spring loads for grids

(5) fuel assembly

(a) loads applied by core restraint system
(b) analysis of combined shock (including LOCA) and seismic loading
(c) loads applied in fuel handling, including misaligned handling tools

(6) reactivity control assembly and burnable poison rods

(a) internal pressure and cladding stresses during normal, transient, and accident conditions

(b) thermal stability of the absorber material, including phase changes and thermal expansion

(c) irradiation stability of the absorber material, taking into consideration gas release and swelling

(d) potential for chemical interaction, including possible waterlogging rupture

When conclusive operating experience is not available, applicants should discuss any prototype testing associated with the fuel design, with a particular focus on any of the following prototype tests that have been performed:

- spacer grid structural tests
- control rod structural and performance tests
- fuel assembly structural tests (lateral, axial and torsional stiffness, frequency, and damping)

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[1] If Chapter 15 of the FSAR includes this information, applicants may incorporate it into this section by reference.

- fuel assembly hydraulic flow tests (lift forces, control rod wear, vibration, and assembly wear and life)
- in-reactor testing of design features and lead assemblies of a new design, which may include one or more of the following:
    - fuel and burnable poison rod growth
    - fuel rod bowing
    - fuel assembly growth
    - fuel assembly bowing
    - channel box wear and distortion
    - fuel rod ridging
    - crud formation
    - fuel rod integrity
    - holddown spring relaxation
    - spacer grid spring relaxation
    - guide tube wear characteristics

The section should also discuss the following phenomenological models:

- radial power distribution
- fuel and cladding temperature distribution
- burnup distribution in the fuel
- thermal conductivity of the fuel, cladding, cladding crud, and oxidation layers
- densification of the fuel
- thermal expansion of the fuel and cladding
- fission gas production and release
- solid and gaseous fission product swelling
- fuel restructuring and relocation
- fuel and cladding dimensional changes
- fuel-to-cladding heat transfer coefficient
- thermal conductivity of the gas mixture
- thermal conductivity in the Knudsen domain
- fuel-to-cladding contact pressure
- heat capacity of the fuel and cladding
- growth and creep of the cladding
- rod internal gas pressure and composition
- sorption of helium and other fill gases
- cladding oxide and crud layer thickness
- cladding-to-coolant heat transfer coefficient

In addition, applicants should provide the following three main types of information:

(1) The design evaluation should include the following fuel system damage criteria for all mechanisms:

(a) stress, strain, or loading limits for spacer grids, guide tubes, thimbles, fuel rods, control rods, channel boxes, and other fuel system structural members

(b) cumulative number of strain fatigue cycles

(c) fretting wear at contact points on structural members

(d) oxidation, hydriding, and the buildup of corrosion products

(e) dimensional changes, such as rod bowing or irradiation growth on fuel rods and guide tubes (discuss associated analyses)

(f) fuel and burnable poison rod internal gas pressures

(g) worst-case hydraulic loads for normal operations

(h) keeping control rods watertight to maintain control rod reactivity

(2) Regarding fuel rod failure, the design evaluation should include the following:

(a) analysis of maximum linear heat generation rate (LHGR) anywhere in the core, including all hot spots and hot channel factors, and the effects of burnups and composition on the melting point

(b) calculation of the cladding swelling and rupture resulting from the temperature distribution in the cladding and pressure differences between the inside and outside of the cladding (should be included in the evaluation model for the emergency core cooling system (ECCS))

(3) Regarding fuel coolability, the design evaluation should include the following:

(a) how the analysis of the core flow distribution accounts for the burst strain and flow blockage caused by ballooning (swelling)

(b) whether the analyses of other accidents involving system depressurization include burst strain and flow blockage caused by ballooning (swelling)

C.I.4.2.4 Testing and Inspection Plan

This section should describe the testing and inspections to be performed to verify the design characteristics of the fuel system components, including cladding integrity; dimensions; fuel enrichment; burnable poison concentration; absorber composition; and characteristics of the fuel, absorber, and poison pellets. This section should also include descriptions of radiographic inspections, destructive tests, fuel assembly dimensional checks, and the inspection program for new fuel assemblies and new control rods to ensure mechanical integrity after shipment. Where testing and inspection programs are essentially the same for plants previously licensed (or designs previously certified) under Title 10, Part 50, “Domestic Licensing of Production and Utilization Facilities,” of the *Code of Federal Regulations* (10 CFR Part 50) or 10 CFR Part 52, “Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants,” applicants should provide a statement to that effect, along with an identification of the fabricator and a table summarizing the important design and performance characteristics.

In addition, this section should describe the online fuel rod failure monitoring methods and postirradiation surveillance package as well as surveillance of control rods containing boron carbide (B_4C).

C.I.4.3 *Nuclear Design*

C.I.4.3.1 Design Bases

This section should provide and discuss the design bases for the nuclear design of the fuel and reactivity control systems, including nuclear and reactivity control limits such as excess reactivity, fuel burnup, negative reactivity feedback, core design lifetime, fuel replacement program, reactivity coefficients, stability criteria, maximum controlled reactivity insertion rates, control of power distribution, shutdown margins, stuck rod criteria, rod speeds, chemical and mechanical shim control, burnable poison requirements, and backup and emergency shutdown provisions.

C.I.4.3.2 <u>Description</u>

Applicants should describe the nuclear characteristics of the design, including the information indicated in the following subsections.

C.I.4.3.2.1 Nuclear Design Description

Applicants should list, describe, or illustrate features of the nuclear design that are not discussed in specific subsections for appropriate times in the fuel cycle. The description should include such areas as fuel enrichment distributions, burnable poison distributions, other physical features of the lattice or assemblies relevant to nuclear design parameters, delayed neutron fraction and neutron lifetimes, core lifetime and burnup, plutonium buildup, soluble poison insertion rates, and the relationship to cooldown, xenon burnout, or other transient requirements.

C.I.4.3.2.2 Power Distribution

This section should provide full quantitative information on calculated "normal" power distributions, including distributions within typical assemblies, axial distributions, gross radial distributions (XY assembly patterns), and nonseparable aspects of radial and axial distributions. This should include a full range of both representative and limiting power density patterns related to representative and limiting conditions of such relevant parameters as power, flow, flow distribution, rod patterns, time in cycle (burnup and possible burnup distributions), cycle, burnable poison, and xenon. The information should cover these patterns in sufficient detail to ensure that normally anticipated distributions are fully described and the effects of all parameters important in affecting distributions are displayed. This should include details of transient power shapes and magnitudes accompanying normal transients, such as load following, xenon buildup, decay or redistribution, and xenon oscillation control. Applicants should describe the radial power distribution within a fuel pin and its variation with burnup if this is used in thermal calculations.

This section should discuss and assign specific magnitudes to errors or uncertainties that may be associated with these calculated distributions and provide the experimental data, including results from both critical experiments and operating reactors that support the analysis, likely distribution limits, and assigned uncertainty magnitudes. It should also discuss experimental checks to be performed on this reactor as well as the criteria for satisfactory results.

Applicants should provide detailed descriptions of the design power distributions (shapes and magnitudes) and design peaking factors to be used in steady-state limit statements and transient analysis initial conditions. The description should include all relevant components and such variables as maximum allowable peaking factors versus axial position or changes over the fuel cycle. Applicants should justify the selections by discussing the relationships of these design assumptions to the previously provided expected and limiting distributions and uncertainty analysis.

This section should describe the relationship of these distributions to the monitoring instrumentation, discussing in detail the adequacy of the number of instruments and their spatial deployment (including allowed failures); required correlations between readings and peaking factors, calibrations and errors, and operational procedures and specific operational limits; axial and azimuthal asymmetry limits; limits for alarms, and rod blocks, scrams, and other items to demonstrate that sufficient information is available to determine, monitor, and limit distributions associated with normal operation to within proper limits. Applicants should describe in detail all calculations, computer codes, and computers used in the course of operations that are involved in translating power distribution-related measurements into calculated power distribution information. This section should provide the frequency with which the calculations are normally performed and execution times of the calculations. It should also describe the input data required for the codes. In addition, applicants should provide a full quantitative analysis of the uncertainties associated with the sources and processing of information used to produce operational power distribution results. This should include consideration of allowed instrumentation failures.

C.I.4.3.2.3 *Reactivity Coefficients*

This section should provide full quantitative information on calculated reactivity coefficients, including the fuel Doppler coefficient, moderator coefficients (density, temperature, pressure, and void), and power coefficient. It should state the precise definitions or assumptions related to parameters involved (e.g., effective fuel temperature for Doppler, distinction between intra- and interassembly moderator coefficients, parameters held constant in the power coefficient, spatial variation of parameters, and flux weighting used). The information should primarily take the form of curves covering the full applicable range of parameters (density, temperature, pressure, void, and power) from cold startup through limiting values used in accident analyses. It should include quantitative discussions of both spatially uniform parameter changes and those nonuniform parameter and flux weighting changes appropriate to operational and accident analyses as well as the methods used to treat nonuniform changes in transient analyses.

Applicants should provide sufficient information to illustrate the normal and limiting values of parameters appropriate to operational and accident states, considering factors such as cycle, time in cycle, control rod insertions, boron content, burnable poisons, power distribution, and moderator density. This section should discuss potential uncertainties in the calculations and experimental results that support the analysis and assigned uncertainty magnitudes and experimental checks to be made in this reactor. Where limits on coefficients are especially important (e.g., positive moderator coefficients in the power range), applicants should fully detail the experimental checks on these limits.

This section should provide the coefficients actually used in transient analyses and show (by reference to previous discussions and uncertainty analyses) that suitably conservative values are used (1) for both beginning-of-life (BOL) and end-of-life (EOL) analyses, (2) where most negative or most positive (or least negative) coefficients are appropriate, and (3) where spatially nonuniform changes are involved.

C.I.4.3.2.4 *Control Requirements*

This section should provide tables and discussions related to core reactivity balances for BOL, EOL, and (where appropriate) intermediate conditions. The discussions should consider such reactivity influences as control bank requirements and expected and minimum worths, burnable poison worths, soluble boron amounts and unit worths for various operating states, stuck rod allowances, moderator and fuel temperature and void defects, burnup and fission products, xenon and samarium poisoning, pH effects, permitted rod insertions at power, and error allowances. Applicants should also provide and discuss the required and expected shutdown margin as a function of time in cycle, along with uncertainties in the shutdown margin and experimental confirmations from operating reactors.

Applicants should fully describe all methods, paths, and limits for normal operational control involving such areas as soluble poison concentration and changes, control rod motion, power shaping rod (e.g., part length rod) motion, and flow change. Descriptions should consider cold, hot, and peak xenon startup, load following and xenon reactivity control, power shaping (e.g., xenon redistribution or oscillation control), and burnup.

C.I.4.3.2.5 ***Control Rod Patterns and Reactivity Worths***

This section should provide full information on control rod patterns expected to be used throughout a fuel cycle. It should include details concerning separation into groups or banks if applicable; order and extent of withdrawal of individual rods or banks; limits (with justification) to be imposed on rod or bank positions as a function of power level and/or time in cycle or for any other reason; and expected positions of rods or banks for cold critical, hot standby critical, and full power for both BOL and EOL. Applicants should describe allowable deviations from these patterns for misaligned or stuck rods or for any other reason (such as spatial power shaping). For allowable patterns (including allowable deviations), applicants should indicate for various power, EOL, and BOL conditions the maximum worth of rods that might be postulated to be removed from the core in an ejection or drop accident as well as rods or rod banks that could be removed in rod withdrawal accidents. Applicants should also give the worths of these rods as a function of position, describe any experimental confirmations of these worths, and provide maximum reactivity increase rates associated with these withdrawals. This section should describe fully and give the methods for calculating the scram reactivity as a function of time after scram signal, including consideration for technical specification scram times, stuck rods, power level and shape, time in cycle, and any other parameters important for bank reactivity worth and axial reactivity shape functions. In addition, for boiling-water reactors (BWRs), applicants should provide criteria for control rod velocity limiters and control rod worth minimizers.

C.I.4.3.2.6 ***Criticality of Reactor During Refueling***

This section should state the maximum value of K_{eff} for the reactor during refueling and describe the basis for assuming that this maximum value will not be exceeded.

C.I.4.3.2.7 ***Stability***

This section should define the degree of predicted stability with regard to xenon oscillations in both the axial direction and the horizontal plane. If any form of xenon instability is predicted, it should include evaluations of higher-mode oscillations. Applicants should describe in detail the analytical and experimental bases for the predictions and include an assessment of potential error in the predictions. Applicants should also show how unexpected oscillations would be detectable before safety limits are exceeded.

This section should provide unambiguous positions regarding stability or lack thereof. That is, where stability is claimed, it should provide corroborating data from sufficiently similar power plants or provide commitments to demonstrate stability. Applicants should indicate criteria for determining whether the reactor will be stable. Where instability or marginal stability is predicted, applicants should provide details regarding the detection and control of oscillations as well as provisions for protection against exceeding safety limits. In cases in which the applicant does not provide a means for detecting and suppressing instabilities, the application should include a methodology for predicting margins to instability, and it must show that the reactor meets adequate acceptance criteria in this regard. A stability analysis would need to be performed on a cycle-specific basis to determine the limits of operation where stability is assured. Applicants should incorporate the commitment to perform the analysis with an approved methodology through reporting requirements (Section 5.6.5) in the technical specifications.

In addition, applicants should provide analyses of overall reactor stability against power oscillations (other than xenon).

C.I.4.3.2.8 ***Vessel Irradiation***

This section should provide the neutron flux distribution and spectrum in the core, at core boundaries, and at the pressure vessel wall for appropriate times in the reactor life for nil ductility temperature determinations. It should clearly state the assumptions used in the calculations, including power level, use factor, type of fuel cycle, and vessel design life. Applicants should also discuss the computer codes used in the analysis database for fast neutron cross-sections, geometric modeling of the reactor, support barrel, water annulus, and pressure vessel, as well as the calculation uncertainties.

C.I.4.3.3 **<u>Analytical Methods</u>**

This section should describe in detail the analytical methods used in the nuclear design, including those for predicting criticality, reactivity coefficients, and burnup effects. This detailed description should include the computer codes used, including the code name and type, how it is used, its validity (based on critical experiments or confirmed predictions of operating plants), and methods of obtaining nuclear parameters (such as neutron cross-sections). In addition, the detailed descriptions of analytical methods should include estimates of the accuracy of each method.

C.I.4.3.4 **<u>Changes</u>**

This section should list any changes in reactor core design features, calculational methods, data, or information relevant to determining important nuclear design parameters that depart from prior practice of the reactor designs and identify the parameters affected by each change. Details regarding the nature and effects of these changes should be treated in appropriate subsections.

C.I.4.4 *Thermal-Hydraulic Design*

C.I.4.4.1 **<u>Design Bases</u>**

This section should provide the design bases for the thermal-hydraulic design of the reactor. It should include such items as maximum fuel and clad temperatures and cladding-to-fuel gap characteristics as a function of burnup (at rated power, at design overpower, and during transients), critical heat flux ratio (at rated power, at design overpower, and during transients), flow velocities and distribution control, coolant and moderator voids, hydraulic stability, transient limits, fuel cladding integrity criteria, and fuel assembly integrity criteria.

C.I.4.4.2 **<u>Description of Thermal-Hydraulic Design of the Reactor Core</u>**

This section should describe the thermal-hydraulic characteristics of the reactor design and include the information indicated in the following subsections.

C.I.4.4.2.1 ***Summary Comparison***

Applicants should provide a summary comparison of the reactor's thermal-hydraulic design parameters with previously approved reactors of similar design. This should include, for example, primary coolant temperatures, fuel temperatures, maximum and average LHGRs, critical heat flux ratios, critical heat flux correlations used, coolant velocities, surface heat fluxes, power densities, specific powers, surface areas, and flow areas.

C.I.4.4.2.2 ***Critical Heat Flux Ratios***

This section should provide the critical heat flux ratios for the core hot spot at normal full-power and design overpower conditions. It should state the critical heat flux correlation used, analysis techniques, method of use, method of employing peaking factors, and comparison with other correlations.

C.I.4.4.2.3 ***Linear Heat Generation Rate***

This section should provide the core average LHGR as well as the maximum LHGR anywhere in the core. It should also indicate the method of using hot channel factors and power distribution information to determine the maximum LHGR.

C.I.4.4.2.4 ***Void Fraction Distribution***

This section should provide curves showing the predicted radial and axial distributions of steam quality and steam void fraction in the core. It should state the predicted core average void fraction as well as the maximum void fraction anywhere in the core.

C.I.4.4.2.5 ***Core Coolant Flow Distribution***

Applicants should describe and discuss the coolant flow distribution and orificing as well as the basis on which orificing is designed (relative to shifts in power production during core life).

C.I.4.4.2.6 ***Core Pressure Drops and Hydraulic Loads***

Applicants should identify core pressure drops and hydraulic loads during normal and accident conditions that Chapter 15 of the FSAR does not address.

C.I.4.4.2.7 ***Correlations and Physical Data***

This section should discuss the correlations and physical data employed in determining important characteristics such as heat transfer coefficients and pressure drop.

C.I.4.4.2.8 ***Thermal Effects of Operational Transients***

This section should evaluate the capability of the core to withstand thermal effects resulting from anticipated operational transients.

C.I.4.4.2.9 ***Uncertainties in Estimates***

Applicants should discuss the uncertainties associated with estimating the peak or limiting conditions for thermal-hydraulic analysis (e.g., fuel temperature, clad temperature, pressure drops, and orificing effects).

C.I.4.4.2.10 ***Flux Tilt Considerations***

This section should discuss the margin provided in the peaking factor to account for flux tilts to ensure that flux limits are not exceeded during operation. It should describe plans for power reduction in the event of flux tilts and provide criteria for selecting a safe operating power level.

C.I.4.4.3 <u>Description of the Thermal and Hydraulic Design of the Reactor Coolant System</u>

This section should describe the thermal-hydraulic design of the reactor coolant system. The description should include the information indicated in the following subsections. If Chapter 5 of the FSAR provides the applicable information, this section may incorporate it by reference.

C.I.4.4.3.1 *Plant Configuration Data*

This section should provide the following 10 pieces of information on plant configuration and operation:

(1) description of the reactor coolant system, including isometric drawings that show the configuration and approximate dimensions of the reactor coolant system piping

(2) listing of all valves and pipe fittings (e.g., elbows, tees) in the reactor coolant system

(3) total coolant flow through each flowpath (e.g., total loop flow, core flow, bypass flow)

(4) total volume of each plant component, including ECCS components, with sufficient detail to define each part (e.g., downcomer, lower plenum, upper head) of the reactor vessel and steam generator (for pressurized-water reactors (PWRs))

(5) length of the flowpath through each volume

(6) height and liquid level of each volume

(7) elevation of the bottom of each volume with respect to some reference elevation (preferably the centerline of the outlet piping)

(8) lengths and sizes of all safety injection lines

(9) minimum flow areas of each component

(10) steady-state pressure and temperature distribution throughout the system

C.I.4.4.3.2 *Operating Restrictions on Pumps*

This section should state the operating restrictions that will be imposed on the coolant pumps to meet net positive suction head requirements.

C.I.4.4.3.3 *Power-Flow Operating Map (BWR)*

For BWRs, this section should provide a power-flow operating map, indicating the limits of reactor coolant system operation. This map should indicate the permissible operating range, as bounded by minimum flow, design flow, maximum pump speed, and natural circulation.

C.I.4.4.3.4 *Temperature-Power Operating Map (PWR)*

For PWRs, this section should provide a temperature-power operating map. This map should indicate the effects of reduced core flow attributable to inoperative pumps, including system capability during natural circulation conditions.

C.I.4.4.3.5 *Load-Following Characteristics*

Applicants should describe the load-following characteristics of the reactor coolant system as well as the techniques employed to provide this capability.

C.I.4.4.3.6 *Thermal and Hydraulic Characteristics Summary Table*

Applicants should provide a table summarizing the thermal-hydraulic characteristics of the reactor coolant system.

C.I.4.4.4 <u>Evaluation</u>

This section should provide an evaluation of the thermal-hydraulic design of the reactor and the reactor coolant system. This evaluation should include the information indicated in the following subsections.

C.I.4.4.4.1 *Critical Heat Flux*

This section should identify the critical heat flux, departure from nucleate boiling, or critical power ratio correlation used in the core thermal-hydraulic analysis. It should describe the experimental basis for the correlation (preferably by reference to documents available to the U.S. Nuclear Regulatory Commission (NRC)) and discuss the applicability of the correlation to the proposed design. The description should place particular emphasis on the effect of the grid spacer design, the calculational technique used to determine coolant mixing, and the effect of axial power distribution.

C.I.4.4.4.2 *Core Hydraulics*

The core hydraulics evaluation should include (1) a discussion of the results of flow model tests (with respect to pressure drop for the various flowpaths through the reactor and flow distributions at the core inlet), (2) the empirical correlation selected for use in analyses for both single-phase and two-phase flow conditions and applicability over the range of anticipated reactor conditions, and (3) the effect of partial or total isolation of a loop.

C.I.4.4.4.3 *Influence of Power Distribution*

This section should discuss the influence of axial and radial power distributions on the thermal- and hydraulic design. It should include an analysis to determine which fuel rods control the thermal limits of the reactor.

C.I.4.4.4.4 *Core Thermal Response*

Applicants should evaluate the thermal response of the core at rated power, at design overpower, and during expected transient conditions.

C.I.4.4.4.5 *Analytical Methods*

This section should describe the analytical methods and data used to determine the reactor coolant system flow rate. This should include classical fluid mechanics relationships and empirical correlations and should address both single-phase and two-phase fluid flow, as applicable. In addition, this description should provide estimates of the uncertainties in the calculations as well as the resultant uncertainty in reactor coolant system flow rate.

This section should provide a comprehensive discussion of the analytical techniques used in evaluating the core thermal-hydraulics, including estimates of uncertainties. This discussion should include such items as hydraulic instability, application of hot spot factors and hot channel factors, subchannel hydraulic analysis, effects of crud (in the core and reactor coolant system), and operation with one or more loops isolated. Applicants may describe computer codes by referencing documents available to the NRC.

C.I.4.4.5 Testing and Verification

This section should discuss the testing and verification techniques used to ensure that the planned thermal-hydraulic design characteristics of the core and reactor coolant system have been provided and will remain within required limits throughout the core lifetime. This discussion should address the applicable portions of Regulatory Guide 1.68, “Initial Test Programs for Water-Cooled Nuclear Power Plants.” References to the appropriate portions of Chapter 14 of the FSAR are acceptable.

C.I.4.4.6 Instrumentation Requirements

This section should discuss the functional requirements for instrumentation to be employed in monitoring and measuring those thermal-hydraulic parameters that are important to safety. For example, this discussion should include the requirements for in-core instrumentation to confirm predicted power density distribution and moderator temperature distributions. Chapter 7 of the FSAR should provide details of the instrumentation design and logic.

In addition, applicants should describe the vibration and loose-parts monitoring equipment to be provided in the plant and discuss the procedures for detecting excessive vibration and the occurrence of loose parts.

C.I.4.5 *Reactor Materials*

C.I.4.5.1 Control Rod Drive System Structural Materials

For the purpose of this section, the CRDS includes the CRDM and extends to the coupling interface with the reactivity control (poison) elements in the reactor vessel. It does not include the electrical and hydraulic systems necessary to actuate the CRDMs. This section should provide the information described in the following subsections.

C.I.4.5.1.1 *Materials Specifications*

This section should provide a list of the materials, including weld materials, and their specifications for each CRDM component. Applicants should furnish information regarding the mechanical properties of any material not included in either Appendix I to Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (hereafter referred to as the ASME Code) or Regulatory Guide 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III, Division 1,” and justify the use of such materials.

Applicants should state whether the CRDM design uses any materials that have a yield strength greater than 90,000 pounds per square inch, such as cold-worked austenitic stainless steels, precipitation hardenable stainless steels, or hardenable martensitic stainless steels. If such materials are used, applicants should identify their usage and provide evidence that stress-corrosion cracking will not occur during service life in components fabricated from the materials.

C.I.4.5.1.2 ***Austenitic Stainless Steel Components***

This section should describe the processes, inspections, and tests used to ensure that austenitic stainless steel components are free from increased susceptibility to intergranular stress-corrosion cracking caused by sensitization. If special processing or fabrication methods subject the materials to temperatures between 800–1500 °F (427–816 °C) or involve slow cooling from temperatures over 1500 °F (816 °C), applicants should describe the processing or fabrication methods and provide justification to show that such treatment will not cause susceptibility to intergranular stress-corrosion cracking. Applicants should indicate the degree of conformance to the recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," as well as Regulatory Position C.5 of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," as it relates to controls for abrasive steel surfaces. Applicants should justify any deviations from these recommendations.

C.I.4.5.1.3 ***Other Materials***

This section should describe the tempering and aging temperatures for martensitic precipitation-hardening stainless steels to prevent their deterioration by stress corrosion during plant operation. It should also describe the processing and treatment of other special purpose materials, such as cobalt-base alloys (Stellites), nickel-based alloys (Inconel), titanium, colmonoys, and graphitars. Identify all metallic and non-metallic materials used in the CRDM that are not included in Section III, Appendix I, Division 1 of the ASME Code, Section II, "Materials," Parts A, B, C, and D; and Section III, "Rules for Construction of Nuclear Plant Components," Division 1, including Appendix I.

C.I.4.5.1.4 ***Cleaning and Cleanliness Control***

This section should provide details regarding the steps that will be taken to protect austenitic stainless steel materials and parts of these systems during fabrication, shipping, and onsite storage to ensure that all cleaning solutions, processing compounds, degreasing agents, and detrimental contaminants are completely removed and all parts are dried and properly protected following any flushing treatment with water. It should indicate the degree of conformance to the recommendations of Regulatory Guide 1.37 and justify any deviations from these recommendations.

C.I.4.5.2 **<u>Reactor Internals and Core Support Materials</u>**

This section should discuss the materials used for reactor internals and core support materials and include the information described in the following subsections.

C.I.4.5.2.1 ***Materials Specifications***

This section should list the materials, including weld materials, and their specifications for components of the reactor internals and core support structures. It should include materials treated to enhance corrosion resistance, strength, and hardness. Applicants should furnish information regarding the mechanical properties of any material not included in Appendix I to Section III of the ASME Code and justify the use of such materials.

C.I.4.5.2.2 ***Controls on Welding***

This section should indicate the methods and controls that will be used when welding reactor internals components and core support structures and provide assurance that such welds will meet the acceptance criteria of Article NG-5000 in Section III of the ASME Code.

C.I.4.5.2.3 ***Nondestructive Examination***

This sections should indicate that the nondestructive examination procedures used to examine tubular products conform to the requirements of the ASME Code. Applicants should justify any deviations from these requirements.

C.I.4.5.2.4 ***Fabrication and Processing of Austenitic Stainless Steel Components***

This section should indicate the degree of conformance to the recommendations of Regulatory Guides 1.44 and 1.37. If alternative measures are used, applicants should show that they will provide the same assurance of component integrity as would be achieved by following the recommendations of the listed regulatory guides. Applicants should indicate the maximum yield strength of all cold-worked stainless steels used in the reactor internals.

C.I.4.5.2.5 ***Other Materials***

Applicants should submit information on the mechanical properties, corrosion resistance, and fabrication of any materials other than austenitic stainless steels. In particular, applicants should discuss the tempering temperature of hardenable martensitic stainless steels and the aging temperature and aging time of precipitation-hardening stainless steels. This section should also discuss the processing and treatment of other special purpose materials, such as cobalt-base alloys (Stellites), nickel-based alloys (Inconel), titanium, and colmonoys.

C.I.4.6 *Functional Design of Reactivity Control Systems*

This section should provide information to establish that the CRDS, which includes the essential ancillary equipment and hydraulic systems, is designed to provide the required functional performance and is properly isolated from other equipment. It should also provide information to establish the bases for assessing the combined functional performance of all the reactivity control systems to mitigate the consequences of anticipated transients and postulated accidents.

In addition to the CRDS and ECCS, these reactivity control systems include the chemical and volume control system (CVCS) and the emergency boration system (EBS) for PWRs and include the standby liquid control system (SLCS) and the recirculation flow control system (RFCS) for BWRs.

C.I.4.6.1 Information for CRDS

Information submitted should include drawings of the CRDM, layout drawings of the CRDS, process flow diagrams, piping and instrumentation diagrams, component descriptions and characteristics, and a description of the functions of all related ancillary equipment and hydraulic systems. This should also include the control rod drive cooling system for plants that have this system. Applicants may provide this information in conjunction with the information requested for Section 3.9.4 of the FSAR.

C.I.4.6.2 Evaluations of the CRDS

Applicants should provide failure mode and effects analyses of the CRDS in tabular form, with supporting discussion to delineate the logic employed. The failure analysis should demonstrate that the CRDS, which for purposes of these evaluations includes all essential ancillary equipment and hydraulic systems, can perform the intended safety functions with the loss of any single active component.

These evaluations and assessments should establish that all essential elements of the CRDS are identified and provisions are made for isolation from nonessential CRDS elements. In addition, this discussion should establish that all essential equipment is amply protected from common-mode failures (such as failure of moderate- and high-energy lines).

C.I.4.6.3 Testing and Verification of the CRDS

This section should describe the functional testing program. This should include rod insertion and withdrawal tests, thermal and fluid dynamic tests simulating postulated operating and accident conditions, and test verification of the CRDS with imposed single failures, as appropriate.

Applicants should provide preoperational and initial startup test programs. Program descriptions should include the test objectives, methods, and acceptance criteria. If Chapter 14 of the FSAR provides the applicable information, applicants may incorporate it in this section by reference.

C.I.4.6.4 Information for Combined Performance of Reactivity Systems

Other sections of the FSAR (e.g., Sections 9.3.4 and 9.3.5) provide piping and instrumentation diagrams, layout drawings, process diagrams, failure analyses, descriptive material, and performance evaluations related to specific evaluations of the CVCS, SLCS, and RFCS. This section should include sufficient plan and elevation layout drawings to provide bases for establishing that the reactivity control systems (CRDS, ECCS, CVCS, SLCS, RFCS, and EBS) are not vulnerable to common-mode failures when used in single or multiple redundant modes.

Chapter 15 of the FSAR provides evaluations pertaining to the plant's response to postulated process disturbances and equipment malfunctions or failures. This section should list all postulated accidents evaluated in Chapter 15 that take credit for two or more reactivity control systems to prevent or mitigate each accident. In addition, this section should tabulate the related reactivity systems.

C.I.4.6.5 Evaluations of Combined Performance

This section should evaluate the combined functional performance for accidents where two or more reactivity systems are used. The neutronic, fluid dynamic, instrumentation, controls, time sequencing, and other process-parameter-related features primarily appear in Chapters 4, 7, and 15 of the FSAR. This section should include failure analyses to demonstrate that the reactivity control systems are not susceptible to common-mode failures when used redundantly. These failure analyses should consider failures originating within each reactivity control system as well as those originating from plant equipment other than reactivity systems and should be provided in tabular form with supporting discussion and logic.