U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

4.2 FUEL SYSTEM DESIGN

REVIEW RESPONSIBILITIES

Primary - The organization responsible for the review of transient and accident analyses

Secondary - None

I. AREAS OF REVIEW

The organization responsible for the review of transient and accident analyses evaluates the thermal, mechanical, and materials design of the fuel system. The fuel system consists of arrays (assemblies or bundles) of fuel rods, including fuel pellets, insulator pellets, springs, tubular cladding, end closures, hydrogen getters, and fill gas; burnable poison rods including components similar to those in fuel rods; spacer grids and springs; end plates; channel boxes; and reactivity control rods. This section discusses the reactivity control elements of the control rods that extend from the coupling interface of the control rod drive mechanism into the core.

The fuel system safety review provides assurance that (1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs), (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained. General Design Criterion (GDC) 10, within Appendix A to 10 CFR Part 50, also addresses item 1 above. Specifically, GDC 10 establishes specified acceptable fuel design limits (SAFDLs) that should not be exceeded during any condition of normal operation, including the effects of AOOs. Therefore, the SAFDLs are established to ensure that the fuel is not damaged. Within this context, “not damaged” means that the fuel rods do not fail, fuel system dimensions remain within operational tolerances, and functional capabilities are not reduced below those assumed in the safety analysis. The design limits of GDC 10 (i.e., the SAFDLs) accomplish these objectives. In a “fuel rod failure,” the fuel rod leaks and the first fission product barrier (the cladding) is breached. The dose analysis required by 10 CFR Part 100 for postulated accidents must account for fuel rod failures. “Coolability,” in general, means that the fuel assembly retains its rod-bundle geometry with adequate coolant channels to permit removal of residual heat even after a severe accident. The general requirements to maintain control rod insertability and core coolability appear repeatedly in the GDC found in Appendix A to 10 CFR Part 50 (e.g., GDC 27 and 35). In particular, 10 CFR 50.46 provides the specific coolability requirements for the loss-of-coolant accident (LOCA).

Standard Review Plan (SRP) Section 4.2 describes all fuel damage criteria. SRP Section 4.4 provides specific thermal-hydraulic criteria for instances involving limits to the departure from nucleate boiling ratio (DNBR) and the critical power ratio (CPR). The available radioactive fission product inventory in fuel rods (i.e., the gap inventory expressed as a release fraction) is provided to the U.S. Nuclear Regulatory Commission’s (NRC) organization that is responsible for the review of design basis accident radiological consequence analyses for use in estimating the radiological consequences of plant releases.

The specific areas of review are as follows:

1. Design Bases. Design bases for the safety analysis address fuel system damage mechanisms and provide limiting values for important parameters to prevent damage from exceeding acceptable levels. The design bases should reflect the safety review objectives as described above.

   The reviewer should evaluate established (past) design-basis limits and associated SAFDLs to determine whether they remain applicable to the new fuel design (including the introduction of new materials) given the operating conditions (temperature, burnup, and power). If they do not apply, new limits must be established based on appropriate data.

2. Description and Design Drawings. The reviewer examines the fuel system description and design drawings. In general, the description will emphasize product specifications rather than process specifications.

3. Design Evaluation. The reviewer evaluates the performance of the fuel system during normal operation, AOOs, and postulated accidents to determine whether all design bases are met. The fuel system components, as listed above, are reviewed not only as separate components but also as integral units such as fuel rods and fuel assemblies. New fuel designs, new operating limits (e.g., rod burnup and power), and the introduction of new materials to the fuel system require a review to verify that existing design-basis limits, analytical models, and evaluation methods remain applicable for the specific design for normal operation, AOOs, and postulated accidents. The review also evaluates operating experience, direct experimental comparisons, detailed mathematical analyses (including fuel performance codes), and other information.

4. <u>Testing, Inspection, and Surveillance Plans</u>. The licensee performs testing and inspection of new fuel to ensure that the fuel is fabricated in accordance with the design and that it reaches the plant site and is loaded in the core without damage. Online fuel rod failure monitoring and postirradiation surveillance should be performed to detect anomalies or confirm that the fuel system is performing as expected; surveillance of control rods containing B_4C should be performed to preclude reactivity loss. The organization responsible for reactor systems reviews the testing, inspection, and surveillance plans, along with their reporting provisions, to ensure that the important fuel design considerations have been addressed.

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

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

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

<u>Review Interfaces</u>

SRP Section 4.2 describes all fuel damage criteria. SRP Section 4.3 establishes fuel criteria for axial offset anomaly (AOA). For those criteria that involve DNBR or CPR limits, SRP Section 4.4 provides specific thermal-hydraulic criteria. The available radioactive fission product inventory in fuel rods (i.e., the gap inventory expressed as a release fraction) is provided to those organizations that estimate the radiological consequences of plant releases in accordance with SRP Chapter 15. Fuel stored energy, flow blockage, peak cladding temperature, and equivalent cladding reacted (ECR) limits defined in SRP Section 4.2 are provided to those organizations that review Chapter 15.

Other SRP sections interface with this section as follows:

1. Review of the nuclear design of the fuel assemblies, control systems, and reactor core under SRP Section 4.3.

2. Review of the thermal margins, the effects of corrosion products (crud), and the acceptability of hydraulic loads under SRP Section 4.4.

3. Review of the design bases for the emergency core cooling system (ECCS), including GDC and ECCS acceptance criteria, under SRP Section 6.3.

4. Review of the postulated fuel failures resulting from overheating of cladding, overheating of fuel pellets, excessive fuel enthalpy, pellet/cladding interaction (PCI), and bursting under Chapter 15.

5. Review of the control rod drive mechanism design in SRP Section 3.9.4 and the reactor internals design under SRP Section 3.9.5.

6. Review of the estimates of radiological dose consequences under Chapter 15.

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

II. ACCEPTANCE CRITERIA

Requirements

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

1. 10 CFR 50.46, 10 CFR 50.34, and 10 CFR 50.67, as they relate to the cooling performance analysis of the ECCS using an acceptable evaluation model and establishing acceptance criteria for light-water nuclear power reactor ECCSs.

2. 10 CFR Part 100 and 10 CFR 50.67, as they relate to determining the acceptability of a reactor site based on calculating the exposure to an individual as a result of fission product releases to the environment following a major accident scenario.

3. GDC 10, as it relates to assuring that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs.

4. GDC 27, as it relates to the reactivity control system being designed with appropriate margin and, in conjunction with the ECCS, being capable of controlling reactivity and cooling the core under postaccident conditions.

5. GDC 35, as it relates to providing an ECCS to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

6. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;

7. 10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.

SRP Acceptance Criteria

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

Specific criteria necessary to meet the relevant requirements of 10 CFR 50.46; GDC 10, 27, and 35; Appendix K to 10 CFR Part 50; and 10 CFR Part 100 are as follows:

1. Design Bases

   The fuel system design bases must reflect the four objectives described in Subsection I, Areas of Review. To satisfy these objectives, acceptance criteria are needed for fuel system damage, fuel rod failure, and fuel coolability. These criteria are discussed in the following paragraphs:

   A. Fuel System Damage

      This subsection applies to normal operation, and Section 4.2 of the safety analysis report should contain the information to be reviewed.

      To meet the requirements of GDC 10, as it relates to SAFDLs for normal operation, including AOOs, fuel system damage criteria should be included for all known damage mechanisms.

      Fuel damage criteria should assure that fuel system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis. When applicable, the fuel damage criteria should consider high burnup effects based on irradiated material properties data. Complete damage criteria should address the following:

      i. Stress, strain, or loading limits for spacer grids, guide tubes, thimbles, fuel rods, control rods, channel boxes, and other fuel system structural members should be provided. Stress limits that are obtained by methods similar to those given in Section III of the Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers (ASME) are acceptable. Other proposed limits must be justified.

ii. The cumulative number of strain fatigue cycles on the structural members mentioned in item (i) above should be significantly less than the design fatigue lifetime, which is based on appropriate data and includes a safety factor of 2 on stress amplitude or a safety factor of 20 on the number of cycles. Other proposed limits must be justified.

iii. Fretting wear at contact points on the structural members mentioned in item (i) above should be limited. Fretting wear tests and analyses that demonstrate compliance with this design basis should account for grid spacer spring relaxation. The allowable fretting wear should be stated in the safety analysis report, and the stress and fatigue limits in items (i) and (ii) above should presume the existence of this wear.

iv. Oxidation, hydriding, and the buildup of corrosion products (crud) should be limited, with a limit specified for each fuel system component. These limits should be established based on mechanical testing to demonstrate that each component maintains acceptable strength and ductility. The safety analysis report should discuss allowable oxidation, hydriding, and crud levels and demonstrate their acceptability. These levels should be presumed to exist in items (i) and (ii) above. The effect of crud on thermal-hydraulic considerations and neutronic (AOA) considerations are reviewed as described in SRP Sections 4.3 and 4.4.

v. Dimensional changes, such as rod bowing or irradiation growth of fuel rods, fuel assemblies, control rods, and guide tubes, should be limited to prevent fuel failures or a situation in which the thermal-hydraulic limits established in Section 4.4 are exceeded. Irradiation growth can result in a significant interference fit between the rod upper end cap and the tie plate (in a boiling-water reactor (BWR)) or the upper nozzle (in a pressurized-water reactor (PWR)), resulting in rod bowing.

Control blade/rod, channel, and guide tube bow as a result of (1) differential irradiation growth (from fluence gradients), (2) shadow corrosion (hydrogen uptake results in swelling), and (3) stress relaxation, which can impact control blade/rod insertability from interference problems between these components. For BWRs, the effects of shadow corrosion should be considered for new control blade or channel designs, dimensions (e.g., the distance between control blade and channel is important), or materials. The effects of channel bulge should also be considered for interference problems for BWRs. Design changes can alter the pressure drop across the channel wall, thus necessitating an evaluation of such changes. Channel material changes can also impact the differential growth, stress relaxation, and the amount of bulge and therefore must be evaluated. If interference is determined to be possible, tests are needed to demonstrate control blade/rod insertability consistent with assumptions in safety analyses. Additional in-reactor surveillance (e.g., insertion times) may also be necessary for new designs, dimensions, and materials to demonstrate satisfactory performance.

vi. Fuel and burnable poison rod internal gas pressures should remain below the nominal system pressure during normal operation or other limits must be justified based on, but not limited to, the following minimum criteria.

(1) No cladding liftoff during normal operation

(2) No reorientation of the hydrides in the radial direction in the cladding

(3) A description of any additional failures resulting from departure of nucleate boiling (DNB) caused by fuel rod overpressure during transients and postulated accidents (see Subsection II, item 1.B.vii)

vii. Because unseating a fuel bundle may challenge control rod/blade insertion, an evaluation of worst-case hydraulic loads should be performed for normal operation, AOOs, and accidents. These worst-case hydraulic loads for normal operation should not exceed the holddown capability of the fuel assembly (either gravity or holddown springs). Hydraulic loads for this evaluation are reviewed as described in SRP Section 4.4.

viii. Control rod reactivity and insertability must be maintained. This requires that, at a minimum, the following may need to be reviewed:

(1) Changes in control rod configuration

(2) Introduction of new materials

(3) Changes in neutronics and mechanical lifetime

(4) Changes in mechanical design

(5) The ability to exclude water/coolant if water-soluble or leachable materials (e.g., B_4C) are used

Changes in mechanical and neutronics lifetimes need to be calculated using acceptable methods. Safety analyses must specifically account for the reduction in neutron-absorbing capabilities with time in-reactor.

B. <u>Fuel Rod Failure</u>

This subsection applies to normal operation, AOOs, and postulated accidents. Items 1.B.i through 1.B.iii below address failure mechanisms that are more limiting during normal operation; Section 4.2 of the safety analysis report should contain the information to be reviewed. Items 1.B.iv through 1.B.viii below address failure mechanisms that are more limiting during AOOs and postulated accidents; Chapter 15 of the safety analysis report usually contains the information to be reviewed.

To meet the requirements of (1) GDC 10 as it relates to SAFDLs for normal operation, including AOOs and (2) 10 CFR Part 100 as it relates to fission product releases for postulated accidents, fuel rod failure criteria should be provided for all known fuel rod failure mechanisms. Fuel rod failure is defined as the loss of fuel rod hermeticity. Although the staff recognizes that it is impossible to avoid all fuel rod failures and that cleanup systems are installed to handle a small number of leaking rods, the review must ensure that fuel does not fail as a result of specific causes during normal operation and AOOs. Fuel rod failures are permitted during postulated accidents, but they must be accounted for in the dose analysis.

Fuel rod failures can be caused by overheating, PCI, hydriding, cladding collapse, bursting, mechanical fracturing, and fretting. When applicable, the fuel rod failure criteria should consider high burnup effects based on irradiated material properties data.

Complete fuel failure criteria should address the following:

i. Hydriding. Both internal and external sources of hydriding can cause a zirconium alloy component to fail. To prevent failure from internal hydriding (i.e., primary hydriding), the level of moisture and other hydrogenous impurities within the fuel is kept very low during fabrication. Acceptable moisture levels for Zircaloy-clad uranium oxide fuel should be no greater than 20 micrograms per gram (μg/g) (20 parts per million (ppm)). Current specifications of the American Society for Testing and Materials (ASTM),1989 edition, Standard C776-89, Part 45, for uranium oxide fuel pellets state an equivalent limit of 2 μg/g (2 ppm) of hydrogen from all sources. For other materials clad in Zircaloy tubing, an equivalent quantity of moisture or hydrogen can be tolerated. A moisture level of 2 milligrams of water per cubic centimeter of hot void volume within the Zircaloy cladding has been shown to be insufficient for primary hydride formation. External hydriding is caused by waterside corrosion in which the water reaction with the zirconium alloy results in zirconium hydrides as well as zirconium dioxide.

ii. Cladding Collapse. If axial gaps in the fuel pellet column result from densification, the cladding has the potential to collapse into a gap (i.e., flattening). Because of the large local strains that accompany this process, collapsed (flattened) cladding is assumed to fail.

iii. Overheating of Cladding. Traditional practice assumes that failures will not occur if the thermal margin criteria (DNBR for PWRs and CPR for BWRs) are satisfied. SRP Section 4.4 details the review of these criteria. Violation of the thermal margin criteria is not permitted for normal operation and AOOs. For postulated accidents, the total number of fuel rods that exceed the criteria has been assumed to fail for radiological dose calculation purposes. Although a thermal margin criterion is sufficient to demonstrate that overheating from a deficient cooling mechanism can be avoided, it is not a necessary condition (i.e., DNB is not a failure mechanism) and other mechanistic methods may be acceptable. At present, there is little experience with other approaches, but new positions recommending different criteria should address cladding temperature, pressure, time duration, oxidation, and embrittlement.

iv. Overheating of Fuel Pellets. Traditional practice has also assumed that failure will occur if centerline melting takes place. This analysis should be performed for the maximum linear heat generation rate anywhere in the core, including all hot spots and hot channel factors, and should account for the effects of burnup and composition on the melting point. For normal operation and AOOs, centerline melting is not permitted. For postulated accidents, the total number of rods that experience centerline melting should be assumed to fail for radiological dose calculation purposes. The centerline melting criterion was established to assure that axial or radial relocation of molten fuel would neither allow molten fuel to contact the cladding nor produce local hot spots. The assumption that centerline melting results in fuel failure is conservative.

v. Excessive Fuel Enthalpy. The sudden increase in fuel enthalpy from a reactivity initiated accident (RIA) below fuel melting can result in fuel failure due to pellet/cladding mechanical interaction (PCMI) (see Subsection II, item 1.B.vii). Exceeding the DNBR for a PWR or the CPR for a BWR may result in cladding failure during an RIA. See Appendix B for criteria.

vi. Pellet/Cladding Interaction. No criterion currently exists for fuel failure resulting from PCI or PCMI. The difference between PCI and PCMI is subtle, and it is sometimes difficult to differentiate the two types of failures from visual observation of the failure. PCI is generally caused by stress-corrosion cracking due to fission product (iodine) embrittlement of the cladding, while PCMI is primarily a stress-driven failure. The design basis for PCI and PCMI can only be generally stated.

Two related criteria should be applied, but they are not sufficient to preclude PCI or PCMI failures. The first criterion limits uniform strain of the cladding to no more than 1 percent. In this context, uniform strain (elastic and inelastic) is defined as transient-induced deformation with gauge lengths corresponding to cladding dimensions; steady-state creepdown and irradiation growth are excluded. Mechanical testing must demonstrate that the irradiated cladding ductility at maximum waterside corrosion (hydride embrittlement) is well within the 1-percent strain criterion. Although observing this strain limit may preclude some PCI and PCMI failures, it will neither preclude the corrosion-assisted failures that occur at low strains nor the highly localized overstrain failures introduced by pellet chips on the outer fuel diameter. The second criterion states that fuel melting should be avoided. The large volume increase associated with melting may cause a pellet with a molten center to exert a stress on the cladding. Avoiding fuel melting can preclude such a PCI. Note that item 1.B.iv above invoked this same criterion to ensure that overheating of the cladding would not occur.

Fuel vendors have introduced fuel design limits on power maneuvering and rate of power ascension to prevent PCI or PCMI. These design limits have primarily been based on power ramp data from test reactors for a specific fuel design. Recently, however, fuel vendors have been relying more on their predictions of cladding strain and less on their power ramp data to verify that PCMI will not occur. Convincing evidence exists that gaseous swelling and fuel thermal expansion is responsible for cladding strains at high burnup levels and perhaps at even moderate burnups. Therefore, PCI or PCMI analyses of cladding strain for AOO transients and accidents should apply approved fuel thermal expansion and gaseous fuel swelling models, as well as irradiated cladding properties.

vii. Bursting. To meet the requirements of 10 CFR 50.46, as it relates to ECCS performance evaluation, the ECCS evaluation model should include a calculation of the swelling and rupture of the cladding resulting from the temperature distribution in the cladding and from pressure differences between the inside and outside of the cladding. Regulatory Guide (RG) 1.157 provides guidelines for performing a realistic (i.e., best estimate) model to calculate the degree of cladding swelling and rupture. Alternatively, Appendix K to 10 CFR Part 50 presents the acceptable features of an evaluation model for predicting the degree of swelling and rupture in the Zircaloy cladding. Although fuel suppliers may use different rupture-temperature vs. differential-pressure curves, an acceptable curve should be similar to the one described in NUREG-0630 based on similar data for a specific material. Cladding burst from non-LOCA accidents also needs to be evaluated and addressed in terms of impact on cladding temperatures and radiological consequences.

viii. Mechanical Fracturing. A mechanical fracture refers to a defect in a fuel rod caused by an externally applied force such as a hydraulic load or a load derived from core-plate motion. Cladding integrity may be assumed if the applied stress is less than 90 percent of the irradiated yield stress at the appropriate temperature. Other proposed limits must be justified. Results from the seismic and LOCA analysis (see Appendix A to this SRP section) may show that failures by this mechanism will not occur for less severe events.

C. Fuel Coolability

This subsection applies to postulated accidents, and Chapter 15 of the safety analysis report will contain most of the information to be reviewed. Item 1.C.v below addresses the combined effects of two accidents, and Section 4.2 of the safety analysis report should include that information. To meet the requirements of GDC 27 and 35 as they relate to control rod insertability and core coolability for postulated accidents, fuel coolability criteria should be provided for all severe damage mechanisms. Coolability, or coolable geometry, has traditionally implied that the fuel assembly retains its rod-bundle geometry with adequate coolant channels to permit removal of residual heat. Reduction of coolability can result from cladding embrittlement, violent expulsion of fuel, generalized cladding melting, gross structural deformation, and extreme coplanar fuel rod ballooning. This subsection also addresses control rod insertability criteria. Complete criteria should address the following:

i. Cladding Embrittlement. The ECCS performance analysis must satisfy the fuel design criteria specified within 10 CFR 50.46(b). These criteria ensure a coolable core geometry by preserving adequate postquench ductility in the fuel rod cladding. The current criteria require that (1) the peak cladding temperature remains below 2200 °F and (2) the peak cladding oxidation remains below 17 percent ECR. These criteria were originally developed on the basis of unirradiated Zircaloy test specimens. Zirconium alloy composition, manufacturing process, and in-reactor corrosion alter the postquench characteristics of the fuel cladding material. Rulemaking pursuant to 10 CFR 50.46 is planned to implement a performance-based test program that will dictate postquench performance requirements and provide an acceptable means to establish specific limits for new cladding materials. Future cladding alloys must comply with the postquench performance requirements specified by the new rule and provide the empirical database to support any limits assigned to the new alloy.

ii. Violent Expulsion of Fuel. In severe RIAs, such as rod ejection in a PWR or rod drop in a BWR, the large and rapid deposition of energy in the fuel can result in melting, fragmentation, and dispersal of fuel. The mechanical action associated with fuel dispersal can be sufficient to destroy the cladding and the rod-bundle geometry of the fuel and produce pressure pulses in the primary system. (See Appendix B for criteria.)

iii. Generalized Cladding Melting. Generalized (i.e., nonlocal) melting of the cladding could result in the loss of rod-bundle fuel geometry. Criteria for cladding embrittlement in item 1.C.i above are more stringent than melting criteria. Therefore, additional specific criteria are not used. However, this may not always be the case for newer alloys or reactor types.

iv. Fuel Rod Ballooning. To meet the requirements of 10 CFR 50.46 as it relates to ECCS performance during accidents, the analysis of the core flow distribution must account for burst strain and flow blockage caused by ballooning (swelling) of the cladding. RG 1.157 describes acceptable models, correlations, data, and methods that can be used to meet the requirements for a realistic calculation of ECCS performance during a LOCA. Alternatively, Appendix K to 10 CFR Part 50 outlines the acceptable features of a conservative evaluation model to consider burst strain and flow blockage. Burst strain and flow blockage models must be based on applicable data to (1) properly estimate the temperature and differential pressure at which the cladding will rupture (see item 1.B.vii above), (2) avoid underestimating the resultant degree of cladding swelling, and (3) avoid underestimating the associated reduction in assembly flow area.

The flow blockage model evaluation is provided to the organization responsible for the review of transient and accident analyses for incorporation in the comprehensive ECCS evaluation model to demonstrate that the criteria in 10 CFR 50.46(b) are not exceeded. The reviewer also determines whether the analysis of AOOs and other accidents should include fuel rod ballooning. The possibility of ballooning during an AOO transient or accident increases as the fuel rod pressure exceeds the system pressure. Those non-LOCA accidents that result in clad ballooning should examine the possibility of DNB propagation resulting from ballooning. The impact of ballooning on non-LOCA accidents should not be underestimated. A limit on ballooning (circumferential strain) may be required to prevent DNB propagation for these accidents.

v. Structural Deformation. Appendix A discusses the applicable analytical procedures.

2. Description and Design Drawings

The reviewer determines that the fuel system description and design drawings provide an accurate representation and supply the information needed in audit evaluations. Completeness is a matter of judgment, but the following fuel system information and associated tolerances are necessary for an acceptable fuel system description:

- 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
- Pellet grain size and open porosity
- 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
- Control blade/rod descriptions, dimensions, and lifetime limits
- Fit of control blade/rod interference with surrounding structure (e.g., channel box or guide tube)

The following design drawings and dimensions are also necessary for an acceptable fuel system description:

- Fuel assembly cross section
- Fuel assembly outline
- Fuel rod schematic
- Spacer grid cross section
- Guide tube and nozzle joint
- Guide tube with respect to control rod dimensions
- Control blade/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

3. Design Evaluation

The reviewer will evaluate the methods for demonstrating that the design bases are met. Methods include operating experience, prototype testing, and analytical predictions. Many of these methods will be presented generically in topical reports and will be incorporated in the safety analysis report by reference.

A. Operating Experience

Operating experience with fuel systems of the same or similar design should be described, including the maximum burnup experience. When adherence to specific design criteria can be conclusively demonstrated with operating experience, prototype testing and design analyses that were performed before gaining that experience need not be reviewed. Design criteria for fretting wear, oxidation, hydriding, and crud buildup might be addressed in this manner.

B. Prototype Testing

When conclusive operating experience is not available, as with the introduction of a design change, prototype testing should be reviewed. Out-of-reactor tests should be performed, when practical, to determine the characteristics of the new design. No definitive requirements have been developed regarding those design features that must be tested before irradiation, but the following out-of-reactor tests have been performed for this purpose and will serve as a guide to the reviewer:

- Spacer grid structural tests
- Control rod structural and performance tests
- Fuel assembly structural tests (lateral, axial and torsional stiffness, frequency, and damping)
- Fuel assembly hydraulic flow tests (lift forces, control rod wear, vibration, fuel rod fretting (should account for spacer spring relaxation), and assembly wear and life)

In-reactor testing of design features and lead-assembly irradiation of whole assemblies of a new design should be reviewed. The maximum burnup or fluence experience associated with such tests should also be reviewed and considered in relation to the specified maximum burnup or fluence limit for the new design. The following phenomena have been tested in this manner in new designs and will serve as a guide to the reviewer:

- Fuel and burnable poison rod growth
- Fuel rod bowing
- Fuel rod, spacer grid, and channel box oxidation and hydride levels
- Fuel rod fretting
- Fuel assembly growth
- Fuel assembly bowing
- Channel box wear and distortion
- Fuel rod ridging (PCI)

- Crud formation
- Fuel rod integrity
- Holddown spring relaxation
- Spacer grid spring relaxation
- Guide tube wear characteristics

In some cases, in-reactor testing of a new fuel assembly design or a new design feature cannot be accomplished before operation of the design's full core. The inability to perform in-reactor testing may result from an incompatibility of the new design with the previous design. In such cases, special attention should be given to the surveillance plans (see Subsection II.4 below).

C. Analytical Predictions

Some design bases and related parameters can only be evaluated with calculational procedures. The analytical methods that are used to make performance predictions must be reviewed. Many such reviews have been performed establishing numerous examples for the reviewer. The following paragraphs discuss the more established review patterns and provide many related references.

i. Fuel Temperatures (Stored Energy). Fuel temperatures and stored energy during normal operation serve as input to ECCS performance calculations. Temperature calculations require complex computer codes that model many different phenomena. RG 1.157 describes models, correlations, data, and methods to realistically calculate ECCS performance during a LOCA and to estimate the uncertainty in that calculation. Alternatively, an ECCS evaluation model may be developed in conformance with the acceptable features of Appendix K to 10 CFR Part 50. Phenomenological models that should be reviewed include the following:

  - 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
- Cladding hydriding

Because of the strong interaction between these models, overall code behavior should be checked against data (standard problems or benchmarks) and the NRC audit codes. NUREG/CR-6534 (PNNL-11513) Vol. 2, December 1997, FRAPCON-3 NUREG/CR-6534 (PNNL-11513) Vol. 4, May 2005, Babcox & Wilcox Report BAW-10087A, Rev. 1, August 1977, CENPD-139-A, July 1974, Supplement 1 to Technical Report on General Electric Reactor Fuels, December 14, 1973, Technical Report on Exxon Nuclear PWR Fuels, February 27, 1975, and the Letter on Westinghouse Safety Evaluation of WCAP-8720, February 9, 1979, provide examples of previous fuel performance code reviews.

ii. <u>Densification Effects</u>. In addition to its effect on fuel temperatures (discussed above), densification affects (1) core power distributions (power spiking - see SRP Section 4.3), (2) the fuel linear heat generation rate (LHGR) - see SRP Section 4.4, and (3) the potential for cladding collapse. NUREG-0085 and RG 1.126 discuss densification magnitudes for power spike and LHGR analyses. To be acceptable, densification models should follow the guidelines of RG 1.126. Models for cladding-collapse times should also be reviewed. The memorandums on Evaluation of Westinghouse Report, WCAP-8377, January 14, 1975 and on CEPAN-Method of Analyzing Creep Collapse of Oval Cladding, February 5, 1976, provide previous review examples.

iii. Fuel Rod Bowing. The memorandum on Request for Revised Rod Bowing Topical Reports, May 30, 1978, includes guidance for the analysis of fuel rod bowing. The memorandum on Revised Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing in Thermal Margin Calculations for Light Water Reactors, February 16, 1977, presents interim methods that may be used. At this writing, the causes of fuel rod bowing are not well understood and mechanistic analyses of rod bowing have not been approved.

iv. Cladding Collapse. Approved analytical models/methods are used to demonstrate that cladding collapse is not possible within the fuel lifetime. A change in cladding or fuel material (additives or significant changes in fabrication) and/or a reduction in the as-fabricated fuel cladding gap can impact the approved analytical model/methods used for this analysis. A change in fuel material can impact fuel densification and a change in cladding material can impact cladding creep, both of which can impact cladding collapse. If any of these parameters change, they must be evaluated in terms of their impact on the approved analytical models and methods for evaluating cladding collapse.

v. Structural Deformation. Appendix A discusses the acceptance criteria.

vi. Rupture and Flow Blockage (Ballooning). The ECCS evaluation model includes Zircaloy rupture and flow blockage models, which should be reviewed by the organization responsible for reactor systems. The models are empirical and should be compared with relevant data. NUREG-0630, NUREG/CR-1883, and the publication on Burst Criterion of Zircaloy Fuel Cladding in a LOCA, August 4-7, 1980, provide examples of such data and previous reviews. These models should account for the phase transformation in the cladding at high temperatures.

vii. Fuel Rod Pressure. The thermal performance code for calculating temperatures discussed in item 3.C.i above should be used to calculate fuel rod pressures in conformance with the fuel damage criteria of item 1.A.vi in Subsection II. This calculation should account for uncertainties in the estimated rod powers, code models, and fuel rod fabrication. The reviewer should ensure that conservatisms that were incorporated for calculating temperatures do not introduce nonconservatisms with regard to fuel rod pressures.

viii. Metal/Water Reaction Rate. To meet the requirements of 10 CFR 50.46(b) as it relates to the performance of the ECCS during accidents, the rate of energy release, hydrogen generation, and cladding oxidation resulting from the reaction of the Zircaloy cladding with steam should be calculated. Currently this can be calculated in two ways. RG 1.157 allows the use of a best-estimate model, provided its technical basis is demonstrated with appropriate data and analyses. Alternatively, Appendix K to 10 CFR Part 50 specifies that the rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction should be calculated using the Baker-Just equation (Argonne National Laboratory Report ANL-6548, May 1962). For non-LOCA applications, other correlations may be used if justified. These reaction rate models were originally developed based upon unirradiated Zircaloy test specimens. Zirconium alloy composition, manufacturing process, and in-reactor corrosion alter the reaction rate characteristics of the fuel cladding material. Rulemaking pursuant to 10 CFR 50.46 is planned to implement a performance-based test program to provide an acceptable means for establishing specific reaction rate models for new cladding materials. Future cladding alloys must comply with the new rules and the need to provide an empirical database to support applicable reaction rate models.

ix. Fission Product Inventory. The assumptions in RG 1.3, RG 1.4, RG 1.5, RG 1.25, RG 1.77, RG 1.195, and RG 1.196, as they relate to fission product release for existing reactors (i.e., DC applications before January 10, 1997), currently specify the available radioactive fission product inventory in fuel rods (i.e., the gap inventory). RG 1.195 and RG 1.196 can be used in place of RG 1.3, RG 1.4, RG 1.5, RG 1.25, and RG 1.77. RG 1.183 and the requirements of 10 CFR 50.34 apply to fission product release for new reactors. An alternate source term (AST), specified in 10 CFR 50.67, can be applied to existing reactors as an alternative to 10 CFR Part 100 as defined in these documents. American Nuclear Society (ANS) 5.4 presents an approved method for release during non-LOCAs and situations that do not involve accidents in which the fuel temperature exceeds the temperature experienced during normal operation and AOOs. ANS 5.4 also provides an acceptable analytical model for calculating the release of volatile fission products from oxide fuel pellets during normal steady-state conditions. When used with nuclide yields, this model will define the inventory of volatile fission products that could be available for release from the fuel rod if the cladding were breached, sometimes referred to as gap inventory. Recent experimental data from RIA tests in Nuclear Safety Research Reactor (NSRR) and Cabri (Publication on NSRR/RIA Experiments with High Burnup PWR Duels, March 2-6, 1997, Publication on High-Burnup BWR Fuel Behavior Under Simulated Reactivity-Initiated Accident Conditions, Nuclear Technology Vol. 38, June 2002, and Publication on The Role of Grain Boundary Fission Gases in High Burn-Up Fuel Under Reactivity Initiated Accident Conditions, September 2000) suggest that the gap inventory for a BWR rod drop accident specified in RG 1.183 and for a PWR control rod ejection accident may need modification. The NRC has plans to issue new guidelines for gap inventory (fission product release) from these accidents.

4. Testing, Inspection, and Surveillance Plans

Plans must be reviewed for each plant for testing and inspection of new fuel and for monitoring and surveillance of irradiated fuel.

A. Testing and Inspection of New Fuel

Testing and inspection plans for new fuel should verify cladding integrity, fuel system dimensions, fuel enrichment, burnable poison concentration, and absorber composition. Quality control reports should document the details of the manufacturer's testing and inspection programs and should be referenced and summarized in the safety analysis report. The program for onsite inspection of new fuel and control assemblies after they have been delivered to the plant should also be described. When the overall testing and inspection programs are essentially the same as those for previously approved plants, a statement to that effect should be made. In that case, the safety analysis report need not include program details, but an appropriate reference should be cited and a summary (tabular) should be presented.

B. Online Fuel System Monitoring

The applicant's online fuel rod failure detection methods should be reviewed. Both the sensitivity of the instruments and the applicant's commitment to use the instruments should be evaluated. NUREG-0401 and NUREG/CR-1380 evaluate several common detection methods and should be used in this review.

Surveillance is also needed to assure that B_4C control rods are not losing reactivity. Boron compounds are susceptible to leaching in the event of a cladding defect. Periodic reactivity worth tests such as those described in NUREG-0308 are acceptable.

C. Postirradiation Surveillance

A postirradiation fuel surveillance program should be described for each plant to detect anomalies or confirm expected fuel performance. The extent of an acceptable program will depend on the history of the fuel design being considered (i.e., whether the proposed fuel design is the same as current operating fuel or incorporates new design features).

For a fuel design similar to that in other operating plants, a minimum acceptable program should include a qualitative visual examination of some discharged fuel assemblies from each refueling. Such a program should be sufficient to identify gross problems of structural integrity, fuel rod failure, rod bowing, dimension changes, or crud deposition. The program should also commit to perform additional surveillance if unusual behavior is noticed in the visual examination or if plant instrumentation indicates gross fuel failures. The surveillance program should address the disposition of failed fuel.

In addition to the plant-specific surveillance program, a continuing fuel surveillance effort should exist for a given type, make, or class of fuel that can be suitably referenced by all plants using similar fuel. In the absence of such a generic program, the reviewer should expect more detail in the plant-specific program.

For a fuel design that introduces new features, a more detailed surveillance program commensurate with the nature of the changes should be described. This program should include appropriate qualitative and quantitative inspections to be carried out at interim and end-of-life refueling outages. This surveillance program should be coordinated with the prototype testing discussed in Subsection II.3.B. When prototype testing cannot be performed, a special detailed surveillance program should be planned for the first irradiation of a new design.

Technical Rationale

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

1. 10 CFR 50.46 requires each PWR and BWR to be provided with an ECCS that must be designed so that its calculated cooling performance following a postulated LOCA conforms to acceptance criteria set forth in the regulation. Acceptance criteria in 10 CFR 50.46 establish both fuel system design limits and core cooling requirements. SRP Section 4.2 reviews the performance of the fuel system during postulated LOCAs related to flow blockage and the methods used to establish the initial fuel conditions before the LOCA. RG 1.157 or Appendix K to 10 CFR Part 50 presents acceptable methods to evaluate the performance of the ECCS. RG 1.126 provides an acceptable model for predicting the effects of fuel densification in commercial LWRs. Application of acceptance criteria established in 10 CFR 50.46 significantly reduces the possibility of a violent chemical reaction between the Zircaloy cladding and the coolant, which would result, if it were to occur, in the production of explosive hydrogen gas following an accident. It also ensures that damage to the fuel system in the event of an accident is never so severe as to prevent cooling of the core.

2. 10 CFR Part 100 requires the calculation of the exposure to an individual caused by the release of fission products to the environment during a postulated reactor accident and consideration of the result when determining the acceptability of a reactor site. 10 CFR Part 100 and RG 1.195 and RG 1.196 apply to reactors with DC applications before January 10, 1997, unless the reactor has adopted the AST, as defined in 10 CFR 50.67 and RG 1.183. RG 1.195 and RG 1.196 can be used in place of RG 1.3, RG 1.4, RG 1.5, RG 1.25, and RG 1.77. RG 1.183 and the requirements of 10 CFR 50.34 apply to new reactors with DC applications after January 10, 1997; the source terms for both new reactors and the AST are based on total effective dose equivalent rather than whole body dose as used in 10 CFR Part 100 and RG 1.195 and RG 1.196. This section discusses acceptable fission gas release models to perform radiological dose calculations; these models ensure that doses are not underestimated. RG 1.3, RG 1.4, RG 1.183, and RG 1.195 provide acceptable assumptions that may be used to evaluate the radiological consequences associated with a LOCA for BWRs and PWRs. RG 1.25, RG 1.183, and RG 1.196 provide acceptable assumptions that may be used to evaluate the radiological consequences associated with a fuel-handling accident at a fuel handling and storage facility at reactor sites. RG 1.77, RG 1.183, and RG 1.195 identify acceptable analytical methods and assumptions that may be used to evaluate the consequences of a rod ejection accident in PWRs. Evaluation of the radiological dose consequences associated with a postulated reactor accident, as prescribed in 10 CFR Part 100, provides assurance that nuclear reactors can be operated safely under worst-case conditions.

3. GDC 10 requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs. One objective of the fuel system safety review cited in this section is to ensure that the fuel system is not damaged during normal operations or AOOs. SRP Section 4.2 specifies design limits to accomplish this objective, while this section reviews alternative design limits proposed by vendors. Compliance with GDC 10 significantly reduces the likelihood of fuel failures during normal operations or AOOs, thereby minimizing the possible release of fission products. In addition, preventing fuel damage during normal operation and AOOs may also reduce the severity of fuel damage during an accident. For example, an increase in the severity of fuel damage for normal operation may result in an increase in source term consequences, along with a decrease in core coolability and/or control rod insertability for postulated accidents.

4. GDC 27 requires that the reactivity control system be designed with margin to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes. The review of Section 4.2 ensures that fuel system damage is never so severe as to prevent control rod insertion when it is required. Maintaining the ability to insert control rods during postulated accidents minimizes the extent of fuel damage, thus reducing the amount of fission products released to the primary coolant system in the event an accident occurs.

5. GDC 35 requires that a system be provided to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) cladding metal-water reaction is limited to negligible amounts. This section reviews fuel system performance analysis methods under postulated accident conditions to ensure compliance with GDC 35. Application of GDC 35 to the design of the fuel system ensures that fuel rod damage will not interfere with effective emergency core cooling and that cladding temperatures will not reach a temperature high enough to allow a significant metal-water reaction to occur, thereby minimizing the potential for offsite release.

III. REVIEW PROCEDURES

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

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

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

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

2. For review of both DC and COL applications, SRP Section 14.3 should be followed for the review of ITAAC. The review of ITAAC cannot be completed until after the completion of this section.

3. For construction permit (CP) applications, the review should ensure that the design bases set forth in the preliminary safety analysis report (PSAR) meet the acceptance criteria given in Subsection II.A. In addition, the CP review should determine, from a study of the preliminary fuel system design, that there is reasonable assurance that the final fuel system design will meet the design bases. This judgment may be based on experience with similar designs.

4. For operating license (OL) applications, the review should confirm that the design bases set forth in the final safety analysis report (FSAR) meet the acceptance criteria given in Subsection II.A and that the final fuel system design meets the design bases.

Much of the fuel system review is generic and is not repeated for each similar plant. That is, the reviewer will have evaluated the fuel design or certain aspects of the fuel design in previous PSARs, FSARs, and licensing topical reports. All previous reviews on which the current review depends should be referenced so that the plant safety evaluation report comprises a completely documented safety evaluation. In particular, the NRC safety evaluation reports for all relevant licensing topical reports should be cited. Staff in the organization responsible for reactor systems has also performed certain generic reviews, the findings of which have been issued as NUREG or WASH series reports. At the present time, these reports include WASH-1236, NUREG-75/077, NUREG-0085, NUREG-0303, NUREG-0401, and NUREG-0418, and they should all be appropriately cited in the plant safety evaluation reports. These reports should also cite the applicable RGs (RG 1.3, RG 1.4, RG 1.25, RG 1.60, RG 1.77, RG 1.126, RG 1.157, and RG 1.183). Deviation from these guides or positions should be explained. After briefly discussing related reviews, the plant safety evaluation should concentrate on those areas in which the application is not identical to one previously reviewed and approved and on areas related to newly discovered problems.

Analytical predictions discussed in Subsection II.3.C will be reviewed in PSARs, FSARs, or licensing topical reports. The validity of analytical models used to predict the performance of the fuel system design, and their applicability up to the design's specified burnup and power limit, should be reviewed. Fuel burnup and power limits should be specified for each fuel type used in the reactor and justified based on irradiated material properties data and prototypic test results. An exception may be made for prototype test assemblies, in which case only an estimate of the maximum burnup and power needs to be provided. When the methods are being reviewed, the staff may perform calculations to verify the adequacy of the analytical methods. Thereafter, audit calculations will not typically be performed to verify the results of an approved method that has been submitted in a safety analysis report. Calculations, benchmarking exercises, and additional reviews of generic methods may be undertaken, however, at any time a clear need arises to reconfirm the adequacy of the method.

IV. EVALUATION FINDINGS

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

> The staff concludes that the fuel system of the ________ plant has been designed so that (1) the fuel system will not be damaged as a result of normal operation and anticipated operational occurrences, (2) fuel damage during postulated accidents will not be severe enough to prevent control rod insertion when it is required, and (3) core coolability will always be maintained, even after severe postulated accidents, thereby meeting the related requirements of 10 CFR 50.46; GDC 10, 27, and 35 in Appendix A to 10 CFR Part 50; and 10 CFR Part 100 (for existing reactors) or 10 CFR 50.34 (for new reactors) or 10 CFR 50.67 (as an alternative to 10 CFR Part 100 for existing reactors). This conclusion is based on the following:
>
> 1. The applicant has provided sufficient evidence that these design objectives will be met based on operating experience, prototype testing, and analytical predictions. Those analytical predictions dealing with structural response and fuel densification have been performed in accordance with (1) the guidelines of RG 1.60, RG 1.77, and RG 1.126, or methods that the staff has reviewed and found to be acceptable alternatives to those RGs and (2) the guidelines in Appendix A to SRP Section 4.2. Those analytical predictions dealing with control rod ejection (PWR) or drop (BWR) have been performed in accordance with the interim criteria for RIAs in Appendix B to SRP Section 4.2.
>
> 2. The applicant has provided for testing and inspection of new fuel to ensure that it is within design tolerances at the time of core loading. The applicant has made a commitment to perform online fuel failure monitoring and postirradiation surveillance to detect anomalies or confirm that the fuel has performed as expected.
>
> The staff concludes that the applicant has described methods of adequately predicting fuel rod failures during postulated accidents so that radioactivity releases are not underestimated and thereby meets the related requirements of 10 CFR Part 100 or 10 CFR 50.67 or 10 CFR 50.34 (for new reactors). In meeting these requirements, the applicant has (1) used the fission-product release assumptions of RG 1.3 (or RG 1.4), RG 1.25, RG 1.77, and RG 1.183 and (2) performed the analysis for fuel rod failures for the rod ejection accident in accordance with the guidelines of Appendix B to Section 4.2 or with methods that the staff has reviewed and found to be an acceptable alternative to Appendix B.

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

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

V. IMPLEMENTATION

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

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

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced RGs and NUREGs.

VI. REFERENCES

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

2. 10 CFR 50.34, "Contents of Applications; Technical Information."

3. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."

4. 10 CFR 50.67, "Accident Source Term."

5. 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."

6. 10 CFR 52.47, "Contents of Applications."

7. 10 CFR 52.97, "Issuance of Combined Licenses."

8. 10 CFR Part 100, "Reactor Site Criteria."

9. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Basis for Protection Against Natural Phenomena."

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

11. 10 CFR Part 50, Appendix A, General Design Criterion 27, "Combined Reactivity Control Systems Capability."

12. 10 CFR Part 50, Appendix A, General Design Criterion 35, "Emergency Core Cooling."

13. 10 CFR Part 50, Appendix K, "ECCS Evaluation Models."

14. Regulatory Guide 1.3, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors.”

15. Regulatory Guide 1.4, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors.”

16. Regulatory Guide 1.5, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors.”

17. Regulatory Guide 1.25, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors.”

18. Regulatory Guide 1.60, “Design Response Spectra for Seismic Design of Nuclear Power Plants.”

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

20. Regulatory Guide 1.126, “An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification.”

21. Regulatory Guide 1.157, “Best-Estimate Calculations of Emergency Core Cooling System Performance.”

22. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.”

23. Regulatory Guide 1.195, “Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors.”

24. Regulatory Guide 1.196, “Control Room Habitability at Light-Water Nuclear Power Reactors.”

25. WASH-1236, “Technical Report on Densification of Light Water Reactor Fuels,” Atomic Energy Commission Regulatory Staff Report, November 14, 1972.

26. NUREG-75/077, “The Role of Fission Gas Release in Reactor Licensing,” November 1975.

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

28. NUREG-0303, “Evaluation of the Behavior of Waterlogged Fuel Rod Failures in LWRs,” March 1978.

29. NUREG-0308, Supp. 2, “Safety Evaluation Report Related to Operation of Arkansas Nuclear One, Unit 2,” September 1978.

30. NUREG-0401, “Fuel Failure Detection in Operating Reactors,” March 1978.

31. NUREG-0418, “Fission Gas Release from Fuel at High Burnup,” March 1978.

32. NUREG-0609, “Asymmetric Blowdown Loads on PWR Primary Systems,” January 1981.

33. NUREG-0630, “Cladding Swelling and Rupture Models for LOCA Analysis,” April 1980.

34. NUREG/CR-1018, “Review of LWR Fuel System Mechanical Response with Recommendations for Component Acceptance Criteria,” September 1979.

35. NUREG/CR-1019, “Pressurized Water Reactor Lateral Core-Response Routine, FAMREC (Fuel Assembly Mechanical Response Code),” September 1979.

36. NUREG/CR-1020, “Technical Evaluation of PWR Fuel Spacer Grid Response Load Sensitivity Studies,” September 1979.

37. NUREG/CR-1380, “Assessment of Current Onsite Inspection Techniques for LWR Fuel Systems,” Vol. 1, July 1980; Vol. 2, January 1981.

38. NUREG/CR-1883, “Multirod Burst Test Program Progress Report for January–June 1980,” March 1981.

39. American National Standards Institute, ANSI/ANS 5.4, “Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel,” American Nuclear Society, November 10, 1982.

40. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, “Rules for Construction of Nuclear Power Plant Components,” New York.

41. American Society for Testing and Materials, 1989 edition, Standard C776-89, Part 45, “Standard Specification for Sintered Uranium Dioxide Pellets,” Philadelphia, Pennsylvania.

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45. G.A. Berna, C.E. Beyer, K.L. Davis, and D.D. Lanning, “FRAPCON-3: A Computer Code for the Calculation of Steady-State, Thermal-Mechanical Behavior of Oxide Fuel Rods of High Burnup,” NUREG/CR-6534 (PNNL-11513) Vol. 2, Pacific Northwest National Laboratory, December 1997.

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48. "Fuel Evaluation Model," Combustion Engineering Report CENPD-139-A, July 1974 (Approved version transmitted to the NRC on April 25, 1975).

49. "Supplement 1 to the Technical Report on Densification of General Electric Reactor Fuels," Atomic Energy Commission Regulatory Staff Report, December 14, 1973.

50. "Technical Report on Densification of Exxon Nuclear PWR Fuels," Atomic Energy Commission Regulatory Staff Report, February 27, 1975.

51. Letter from J.F. Stolz, NRC, to T.M. Anderson, Westinghouse, Subject: Safety Evaluation of WCAP-8720, dated February 9, 1979.

52. Memorandum from V. Stello, NRC, to R.C. DeYoung, Subject: Evaluation of Westinghouse Report, WCAP-8377, Revised Clad Flattening Model, dated January 1975.

53. Memorandum from D.F. Ross, NRC, to R.C. DeYoung, Subject: CEPAN - Method of Analyzing Creep Collapse of Oval Cladding, dated February 5, 1976.

54. Memorandum from D.F. Ross, NRC, to D.B. Vassallo, Subject: Request for Revised Rod Bowing Topical Reports, dated May 30, 1978.

55. Memorandum from D.F. Ross and D.G. Eisenhut, NRC, to D.B. Vassallo and K.R. Goller, Subject: Revised Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing in Thermal Margin Calculations for Light Water Reactors, dated February 16, 1977.

56. L. Baker and L.C. Just, "Studies of Metal-Water Reactions at High Temperatures, III. Experimental and Theoretical Studies of the Zirconium-Water Reaction," Argonne National Laboratory Report ANL-6548, May 1962.

57. T. Fuketa, H. Sasajima, Y. Tsuchiuchi, Y. Mori, T. Nakamura, and K. Ishijima, "NSRR/RIA Experiments with High Burnup PWR Duels, Proceedings of 1997 International Topical Meetings on Light Water Fuel Performance, Portland, Oregon, March 2-6, 1997.

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PAPERWORK REDUCTION ACT STATEMENT

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

PUBLIC PROTECTION NOTIFICATION

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

APPENDIX A

EVALUATION OF FUEL ASSEMBLY STRUCTURAL RESPONSE TO EXTERNALLY APPLIED FORCES

I. BACKGROUND

Earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assembly. SRP Section 4.2 states that fuel system coolability should be maintained and that damage should not be so severe as to prevent control rod insertion when required during these low probability accidents. This appendix describes the review that should be performed of the fuel assembly structural response to seismic and LOCA loads. NUREG-0609, NUREG/CR-1018, NUREG/CR-1019, and NUREG/CR-1020 provide background material for this appendix.

II. ANALYSIS OF LOADS

1. Input

Input for the fuel assembly structural analysis comes from the results of the primary coolant system and reactor internals structural analysis, which is reviewed by the organization responsible for the review of mechanical engineering issues. Input for the fuel assembly response to a LOCA should include (1) motions of the core plate, core shroud, fuel alignment plate, or other relevant structures (these motions should correspond to the break that produced the peak fuel assembly loadings in the primary coolant system and reactor internals analysis) and (2) transient pressure differences that apply loads directly to the fuel assembly. If the earthquake loads are large enough to produce a nonlinear fuel assembly response, input for the seismic analysis should use structure motions corresponding to the reactor primary coolant system analysis for the safe-shutdown earthquake (SSE). If a linear response is produced, a spectral analysis may be used in accordance with the guidelines of RG 1.60.

2. Methods

Analytical methods used in performing structural response analyses should be reviewed. The appropriateness of numerical solution techniques should be justified.

Linear and nonlinear structural representations (i.e., the modeling) should also be reviewed. Experimental verification of the analytical representation of the fuel assembly components should be provided when practical.

The applicant should work a sample problem of a simplified nature, which the reviewer will compare with either hand calculations or results generated with an independent code (NUREG/CR-1019). Although the sample problem should use a structural representation that is as close as possible to the design in question (and, therefore, would vary from one vendor to another), the applicant may make simplifying assumptions (e.g., one might use a three-assembly core region with continuous sinusoidal input).

The sample problem should be designed to exercise various features of the code and reveal their behavior. The sample problem comparison is not, however, designed to show that one code is more conservative than another, but rather to alert the reviewer to major discrepancies so that an explanation can be sought.

3. Uncertainty Allowances

The fuel assembly structural models and analytical methods are likely to be conservative; input parameters are also conservative. However, to ensure that the fuel assembly analysis does not introduce any nonconservatisms, two precautions should be taken—(1) if it is not explicitly evaluated, impact loads from the PWR LOCA analysis should be increased by about 30 percent to account for a pressure pulse, which is associated with steam flashing that affects only the PWR fuel assembly analysis and (2) conservative margin should be added if any part of the analysis (PWR or BWR) exhibits pronounced sensitivity to input variations.

Variations in resultant loads should be determined for positive variations in input amplitude and frequency of 10 percent; variations in amplitude and frequency should be made separately, not simultaneously. A factor should be developed for resultant load magnitude variations of more than 15 percent. For example, if +10-percent variations in input magnitude or frequency produce a maximum resultant increase of 35 percent, the sensitivity factor would be 1.2. Since resonances and pronounced sensitivities may be plant dependent, the sensitivity analysis should be performed on a plant-by-plant basis until the reviewer is confident that further sensitivity analyses are unnecessary or it is otherwise demonstrated that the analyses performed are bounding.

4. Audit

The reviewer should perform independent audit calculations for a typical full-sized core to verify that the overall structural representation is adequate. An independent audit code (NUREG/CR-1019) should be used for this audit during the generic review of the analytical methods.

5. Combination of Loads

To meet the requirements of GDC 2, as it relates to combining loads, an appropriate combination of loads from natural phenomena and accident conditions must be made. Loads on fuel assembly components should be calculated for each input (i.e., seismic and LOCA) as described in Subsection II.1 of this appendix, and the resulting loads should be added by the square-root-of-sum-of-squares method. These combined loads should be compared with the component strengths described in Subsection III according to the acceptance criteria in Subsection IV.

III. DETERMINATION OF STRENGTH

1. Grids

All modes of loading (e.g., in-grid and through-grid loadings) should be considered, and the vendor's laboratory grid strength tests should represent the most damaging mode. Test procedures and results should be reviewed to assure that the appropriate failure mode is being predicted. The review should also confirm that (1) the testing impact velocities correspond to expected fuel assembly velocities and (2) the crushing load P(crit) has been suitably selected from the load-versus-deflection curves. Because of the potential for different test rigs to introduce measurement variations, the review of the test procedure will evaluate the grid strength test equipment.

The consequences of grid deformation are small. Gross deformation of grids in many PWR assemblies would be needed to interfere with control rod insertion during an SSE (i.e., buckling of a few isolated grids could not displace guide tubes significantly from their proper location), and grid deformation (without channel deflection) would not affect control blade insertion in a BWR. In a LOCA, gross deformation of the hot channel in either a PWR or a BWR would result in only small increases in peak cladding temperature. Therefore, average values are appropriate, and the allowable crushing load P(crit) should be the 95-percent confidence level on the true mean as taken from the distribution of measurements on unirradiated production grids at (or corrected to) operating temperature. While P(crit) will increase with irradiation, ductility will be reduced. The extra margin in P(crit) for irradiated grids is thus assumed to offset the unknown deformation behavior of irradiated grids beyond P(crit).

2. Components Other Than Grids

Strengths of fuel assembly components other than spacer grids may be deduced from fundamental material properties or experimentation. Supporting evidence for strength values should be supplied. Since structural failure of these components (e.g., fracturing of guide tubes or fragmentation of fuel rods) could be more serious than grid deformation, allowable values should bound a large percentage (about 95 percent) of the distribution of component strengths. Therefore, ASME Code values and procedures may be used when appropriate for determining yield and ultimate strengths. Specification of allowable values may follow the ASME Code requirements and should consider buckling and fatigue effects.

IV. ACCEPTANCE CRITERIA

1. Loss-of-Coolant Accident

Two principal criteria apply for the LOCA—(1) fuel rod fragmentation must not occur as a direct result of the blowdown loads and (2) the 10 CFR 50.46 temperature and oxidation limits must not be exceeded. The first criterion is satisfied if the combined loads on the fuel rods and components other than grids remain below the allowable values defined above. The second criterion is satisfied by an ECCS analysis. If combined loads on the grids remain below P(crit), as defined above, then no significant distortion of the fuel assembly would occur and the usual ECCS analysis is sufficient. If combined grid loads exceed P(crit), then grid deformation must be assumed and the ECCS analysis must include the effects of distorted fuel assemblies. An assumption of maximum credible deformation (i.e., fully collapsed grids) may be made unless other assumptions are justified.

Control rod insertability is a third criterion that must be satisfied. Loads from the worst-case LOCA that requires control rod insertion must be combined with the SSE loads, and control rod insertability must be demonstrated for that combined load. For a PWR, if combined loads on the grids remain below P(crit), as defined above, then significant deformation of the fuel assembly would not occur and lateral displacement of the guide tubes would not interfere with control rod insertion. If combined loads on the grids exceed P(crit), then additional analysis is needed to show that the deformation is not severe enough to prevent control rod insertion.

For a BWR, several conditions must be met to demonstrate control blade insertability—(1) combined loads on the channel box must remain below the allowable value defined above for components other than grids (otherwise, additional analysis is needed to show that the deformation is not severe enough to prevent control blade insertion) and (2) vertical liftoff forces must not unseat the lower tieplate from the fuel support piece such that the resulting loss of lateral fuel bundle positioning could interfere with control blade insertion.

2. Safe-Shutdown Earthquake

Two criteria apply to the SSE—(1) fuel rod fragmentation must not occur as a result of the seismic loads and (2) control rod insertability must be assured. The first criterion is satisfied by the criteria in Subsection IV.1 of this appendix. The second criterion must be satisfied for SSE loads alone if Subsection IV.1 does not require an analysis for combined loads.

APPENDIX B

INTERIM ACCEPTANCE CRITERIA AND GUIDANCE FOR THE REACTIVITY INITIATED ACCIDENTS

A. BACKGROUND

This appendix provides the interim acceptance criteria and guidance for the reactivity-initiated accident (RIA). RIAs consist of postulated accidents which involve a sudden and rapid insertion of positive reactivity. These accident scenarios include a control rod ejection (CRE) for pressurized water reactors (PWRs) and a control rod drop accident (CRDA) for boiling water reactors (BWRs). The uncontrolled movement of a single control rod out of the core results in a positive reactivity insertion which promptly increases local core power. Fuel temperatures rapidly increase, prompting fuel pellet thermal expansion. The reactivity excursion is initially mitigated by Doppler feedback and delayed neutron effects followed by reactor trip. Standard Review Plan (SRP) Section 15.4.8 and 15.4.9 provide further detail on the CRE and CRDA respectively. The technical and regulatory basis of this interim criteria is documented in a memorandum dated January 19, 2007 (ADAMS ML070220400).

B. FUEL CLADDING FAILURE CRITERIA

The total number of fuel rods that must be considered in the radiological assessment is equal to the sum of all of the fuel rods failing each of the criteria below. Applicants do not need to double count fuel rods that are predicted to fail more than one of the criteria.

1. The high cladding temperature failure criteria for zero power conditions is a peak radial average fuel enthalpy greater than 170 cal/g for fuel rods with an internal rod pressure at or below system pressure and 150 cal/g for fuel rods with an internal rod pressure exceeding system pressure. For intermediate (greater than 5% rated thermal power) and full power conditions, fuel cladding failure is presumed if local heat flux exceeds thermal design limits (e.g. DNBR and CPR).

2. The PCMI failure criteria is a change in radial average fuel enthalpy greater than the corrosion-dependent limit depicted in Figure B-1 (PWR) and Figure B-2 (BWR).

Fuel cladding failure may occur almost instantaneously during the prompt fuel enthalpy rise (due to PCMI) or may occur as total fuel enthalpy (prompt + delayed), heat flux, and cladding temperature increase. For the purpose of calculating fuel enthalpy for assessing PCMI failures, the prompt fuel enthalpy rise is defined as the radial average fuel enthalpy rise at the time corresponding to one pulse width after the peak of the prompt pulse. For assessing high cladding temperature failures, the total radial average fuel enthalpy (prompt + delayed) should be used.

C. CORE COOLABILITY CRITERIA

Fuel rod thermal-mechanical calculations, employed to demonstrate compliance with criteria #1 and #2 below, must be based upon design-specific information accounting for manufacturing tolerances and modeling uncertainties using NRC approved methods including burnup-enhanced effects on pellet power distribution, fuel thermal conductivity, and fuel melting temperature.

1. Peak radial average fuel enthalpy must remain below 230 cal/g.

2. Peak fuel temperature must remain below incipient fuel melting conditions.

3. Mechanical energy generated as a result of (1) non-molten fuel-to-coolant interaction and (2) fuel rod burst must be addressed with respect to reactor pressure boundary, reactor internals, and fuel assembly structural integrity.

4. No loss of coolable geometry due to (1) fuel pellet and cladding fragmentation and dispersal and (2) fuel rod ballooning.

D. FISSION PRODUCT INVENTORY

The total fission-product gap fraction available for release following any RIA would include the steady-state gap inventory (present prior to the event) plus any fission gas released during the event. The steady-state gap inventory would be consistent with the Non-LOCA gap fractions cited in RG 1.183 (Table 3) and RG 1.195 (Table 2) and would be dependent on operating power history. Whereas fission gas release (into the rod plenum) during normal operation is governed by diffusion, pellet fracturing and grain boundary separation are the primary mechanisms for fission gas release during the transient.

Based upon measured fission gas release from several RIA test programs, the staff developed the following correlation between gas release and maximum fuel enthalpy increase:

Transient FGR = $[(0.2286 * \Delta H) - 7.1419]$
Where:
FGR = Fission gas release, % (must be ≥ 0)
ΔH = Increase in fuel enthalpy, Δcal/g

The transient release from each axial node which experiences the power pulse may be calculated separately and combined to yield the total transient FGR for a particular fuel rod. The combined steady-state gap inventory and transient FGR from every fuel rod predicted to experience cladding failure (all failure mechanisms) should be used in the dose assessment. Additional guidance is available within RG 1.183 and 1.195.

FIGURE B-1: PWR PCMI Fuel Cladding Failure Criteria

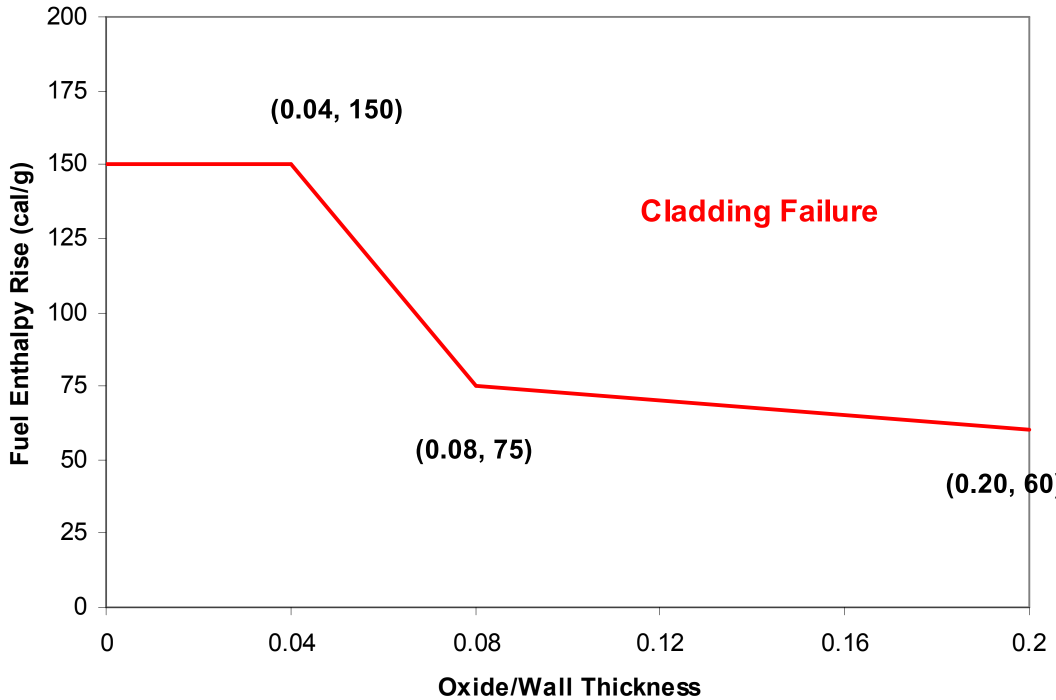

FIGURE B-2: BWR PCMI Fuel Cladding Failure Criteria

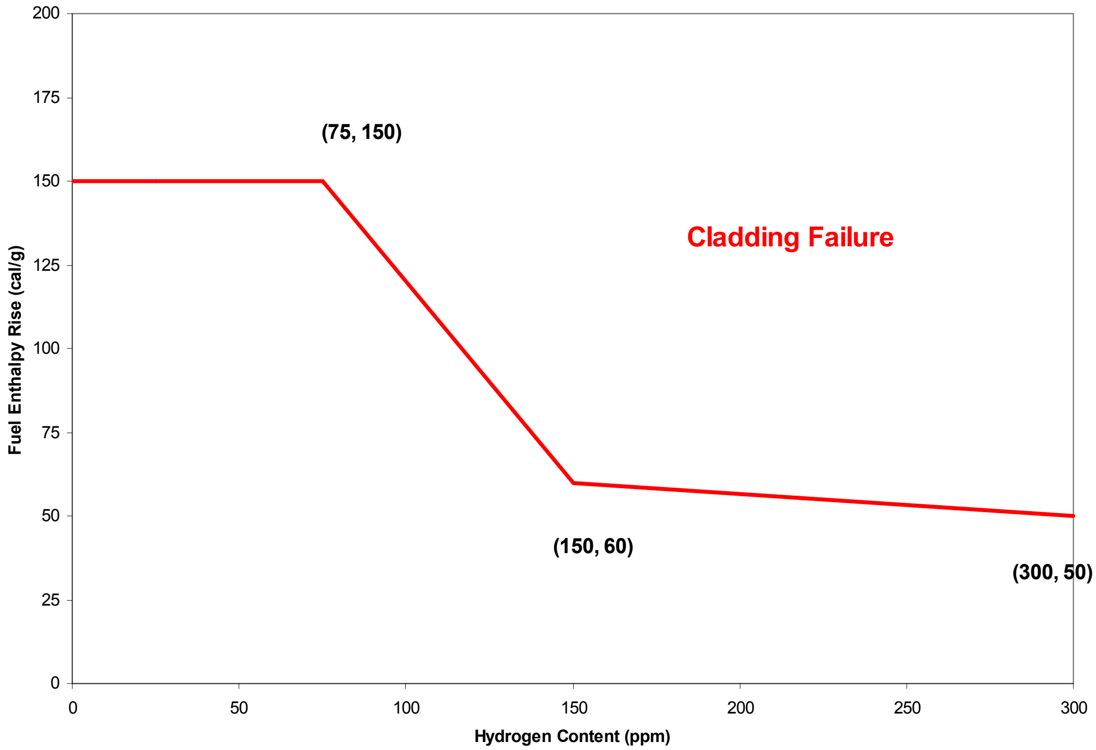

Figure B-2 Note:

The empirical database supporting the BWR fuel cladding failure criteria consists of NSRR tests conducted from initial test temperatures ranging from 20 to 85 °C. Due to temperature and hydrogen solubility effects, application of the BWR cladding failure criteria to higher operating temperatures is conservative. If properly justified, an applicant may adjust the failure criteria to account for hydrogen solubility at initial temperatures above 85 °C.