U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

4.2 FUEL SYSTEM DESIGN

REVIEW RESPONSIBILITIES

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

Secondary - None

I. AREAS OF REVIEW

The thermal, mechanical, and materials design of the fuel system is evaluated by ~~CPB~~ SRXB[2]. 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. In the case of the control rods, this section covers the reactivity control elements that extend from the coupling interface of the control rod drive mechanism into the core. ~~The Mechanical Engineering Branch reviews the design of control rod drive mechanisms in SRP Section 3.9.4 and the design of reactor internals in SRP Section 3.9.5.~~[3]

The objectives of the fuel system safety review are to provide assurance that (a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (b) fuel system damage is never so severe as to prevent control rod insertion when it is required, (c) the number of fuel rod failures is not underestimated for postulated accidents, and (d) coolability is always maintained. "Not damaged," as used in the above statement, means that fuel rods do not fail, that fuel system dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. This objective implements General Design Criterion 10 ~~(Ref. 1)~~[4], and the design limits that accomplish this are called Specified Acceptable Fuel Design Limits (SAFDLs). "Fuel rod failure" means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached. Fuel rod failures must be accounted for in the dose analysis required by 10 CFR Part 100 ~~(Ref. 2)~~[5] for postulated accidents. "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 General Design Criteria (e.g., GDC 27 and 35). Specific coolability requirements for the loss-of-coolant accident are given in 10 CFR Part 50, §50.46 ~~(Ref. 3)~~[6].

All fuel damage criteria are described in SRP Section 4.2. For those criteria that involve DNBR or CPR limits, specific thermal-hydraulic criteria are given in SRP Section 4.4. The available radioactive fission product inventory in fuel rods (i.e., the gap inventory expressed as a release fraction) is provided to the ~~Accident Evaluation Branch~~ Emergency Preparedness and Radiation Protection Branch (PERB)[7] for use in estimating the radiological consequences of plant releases.

The fuel system review covers the following specific areas.

A. <u>Design Bases</u>

Design bases for the safety analysis address fuel system damage mechanisms and provide limiting values for important parameters such that damage will be limited to acceptable levels. The design bases should reflect the safety review objectives as described above.

B. <u>Description and Design Drawings</u>

The fuel system description and design drawings are reviewed. In general, the description will emphasize product specifications rather than process specifications.

C. <u>Design Evaluation</u>

The performance of the fuel system during normal operation, anticipated operational occurrences, and postulated accidents is reviewed to determine if 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. The review consists of an evaluation of operating experience, direct experimental comparisons, detailed mathematical analyses, and other information.

D. <u>Testing, Inspection, and Surveillance Plans</u>

Testing and inspection of new fuel is performed by the licensee 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. On-line 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 ensure against reactivity loss. The testing, inspection, and surveillance plans along with their reporting provisions are reviewed by ~~CPB~~ SRXB[8] to ensure that the important fuel design considerations have been addressed.

Review Interfaces:[9]

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

1. The SRXB reviews the nuclear design of the fuel assemblies, control systems, and reactor core as part of its review responsibility for SRP Section 4.3.[10]

2. The SRXB reviews the thermal margins, the effects of corrosion products (crud), and the acceptability of hydraulic loads as part of its review responsibility for SRP Section 4.4.[11]

3. The SRXB reviews the design bases for the ECCS, including general design criteria and ECCS acceptance criteria, as part of its review responsibility for SRP Section 6.3.[12]

4. The SRXB reviews the postulated fuel failures resulting from overheating of cladding, overheating of fuel pellets, excessive fuel enthalpy, pellet/cladding interaction and bursting as part of its review responsibilities in Chapter 15.[13]

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

1. The Mechanical Engineering Branch (EMEB) reviews the control rod drive mechanism design in SRP Section 3.9.4 and the reactor internals design in SRP Section 3.9.5.[14]

2. The Emergency Preparedness and Radiation Protection Branch (PERB) reviews the estimates of radiological dose consequences as part of its review responsibilities in Chapter 15.[15]

II. ACCEPTANCE CRITERIA

Acceptability of the fuel system design as described in the applicant's safety analysis report (SAR) is based on regulations, general design criteria, regulatory guides, industry standards, and on independent calculations and staff judgements with respect to fuel system functions and component selections. The requirements relevant to the fuel system design are as follows:[16]

1. 10 CFR Part §50.46 as it relates to the cooling performance analysis of the emergency core cooling system (ECCS) using an acceptable evaluation model, and establishing acceptance criteria for light water nuclear power reactor ECCS.[17]

2. 10 CFR Part 100 as it relates 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.[18]

3. General Design Criterion 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 anticipated operational occurrences.[19]

4. General Design Criterion 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 post accident conditions.[20]

5. General Design Criterion 35 as it relates to providing an emergency core cooling system 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.[21]

Specific criteria necessary to meet the relevant requirements of 10 CFR Part 50, §50.46; General Design Criteria 10, 27, and 35; ~~Appendix K to 10 CFR Part 50;~~ [22]and 10 CFR Part 100 ~~identified in subsection I of this SRP section~~[23] are as follows:

A. 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:

1. Fuel System Damage

This subsection applies to normal operation, and the information to be reviewed should be contained in Section 4.2 of the Safety Analysis Report.

To meet the requirements of General Design Criterion 10 as it relates to Specified Acceptable Fuel Design Limits for normal operation, including anticipated operational occurrences, fuel system damage criteria should be given for all known damage mechanisms.

Fuel system damage includes fuel rod failure, which is discussed below in subsection II.A.2. In addition to precluding fuel rod failure, 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. Where applicable, the fuel damage criteria should consider high burnup effects based on irradiated material properties data.[24] Such damage criteria should address the following to be complete.

(a) 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 ASME Code (Reference~~.~~[25] ~~4~~29) are acceptable. Other proposed limits must be justified.

(b) The cumulative number of strain fatigue cycles on the structural members mentioned in paragraph (a) 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 (Reference~~.~~[26] ~~5~~31). Other proposed limits must be justified.

(c) Fretting wear at contact points on the structural members mentioned in paragraph (a) above should be limited. The allowable fretting wear should be stated in the Safety Analysis Report and the stress and fatigue limits in paragraphs (a) and (b) above should presume the existence of this wear.

(d) Oxidation, hydriding, and the buildup of corrosion products (crud) should be limited. Allowable oxidation, hydriding, and crud levels should be discussed in the Safety Analysis Report and shown to be acceptable. These levels should be presumed to exist in paragraphs (a) and (b) above. The effect of crud on thermal-hydraulic considerations is reviewed as described in SRP Section 4.4.

(e) Dimensional changes such as rod bowing or irradiation growth of fuel rods, control rods, and guide tubes need not be limited to set values (i.e., damage limits), but they must be included in the design analysis to establish operational tolerances.

(f) Fuel and burnable poison rod internal gas pressures should remain below the nominal system pressure during normal operation unless otherwise justified.

(g) 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.

(h) Control rod reactivity must be maintained. This may require the control rods to remain watertight if water-soluble or leachable materials (e.g., B_4C) are used.

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

This subsection applies to normal operation, anticipated operational occurrences, and postulated accidents. Paragraphs (a) through (c) address failure mechanisms that are more limiting during normal operation, and the information to be reviewed should be contained in Section 4.2 of the Safety Analysis Report. Paragraphs (d) through (h) address failure mechanisms that are more limiting during anticipated operational occurrences and postulated accidents, and the information to be reviewed will usually be contained in Chapter 15 of the Safety Analysis Report. Paragraph (i) should be addressed in Section 4.2 of the Safety Analysis Report because it is not addressed elsewhere.

To meet the requirements of (a) General Design Criterion 10 as it relates to Specified Acceptable Fuel Design Limits for normal operation, including anticipated operational occurrences, and (b) 10 CFR Part 100 as it relates to fission product releases for postulated accidents, fuel rod failure criteria should be given for all known fuel rod failure mechanisms. Fuel rod failure is defined as the loss of fuel rod hermeticity. Although we recognize that it is not possible to avoid all fuel rod failures and that cleanup systems are installed to handle a small number of leaking rods, it is the objective of the review to assure that fuel does not fail due to specific causes during normal operation and anticipated operational occurrences. 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, pellet/cladding interaction (PCI), hydriding, cladding collapse, bursting, mechanical fracturing, and fretting. Where applicable, the fuel rod failure criteria should consider high burnup effects based on irradiated material properties data.[27] Fuel failure criteria should address the following to be complete.

(a) Internal[28] Hydriding: Hydriding as a cause of failure (i.e., primary hydriding) is prevented by keeping the level of moisture and other hydrogenous impurities within the fuel[29] very low during fabrication. Acceptable moisture levels for Zircaloy-clad uranium oxide fuel should be no greater than 20 µg/g (20 ppm)[30]. Current ASTM specifications (Reference~~.~~[31] ~~7~~30) for $U0_2$ fuel pellets state an equivalent limit of 2 µg/g (2 ppm)[32] 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 mg H_20 per cm^3 of hot void volume within the Zircaloy cladding has been shown (Reference~~.~~[33] ~~8~~32) to be insufficient for primary hydride formation.

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

(c) Fretting: Fretting is a potential cause of fuel failure, but it is a gradual process that would not be effective during the brief duration of an ~~abnormal~~anticipated[34] operational occurrence or a postulated accident. Therefore, the fretting wear requirement in paragraph (c) of subsection II.A.1, Fuel Damage, is sufficient to preclude fuel failures caused by fretting during these[35] transients.

(d) Overheating of Cladding: It has been traditional practice to assume that failures will not occur if the thermal margin criteria (DNBR for PWRs and CPR for BWRs) are satisfied. The review of these criteria is detailed in SRP Section 4.4. For normal operation and anticipated operational occurrences, violation of the thermal margin criteria is not permitted. 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 the avoidance of overheating from a deficient cooling mechanism, it is not a necessary condition (i.e., DNB is not a failure mechanism) and other mechanistic methods may be acceptable. There is at present little experience with other approaches, but new positions recommending different criteria should address cladding temperature, pressure, time duration, oxidation, and embrittlement.

(e) Overheating of Fuel Pellets: It has also been traditional practice to assume 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 anticipated operational occurrences, 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 come into contact with the cladding nor produce local hot spots. The assumption that centerline melting results in fuel failure is conservative.

(f) Excessive Fuel Enthalpy: For a severe reactivity initiated accident (RIA) in a BWR at zero or low power, fuel failure is assumed to occur if the radially averaged fuel rod enthalpy is greater than 711 J/g (170 cal/g)[36] at any axial location. For full-power RIAs in a BWR and all RIAs in a PWR, the thermal margin criteria (DNBR and CPR) are used as fuel failure criteria to meet the guidelines of Regulatory Guide 1.77 ~~(Ref. 6)~~[37] as it relates to fuel rod failure. The 711 J/g (170 cal/g)[38] enthalpy criterion is primarily intended to address cladding overheating effects, but it also indirectly addresses pellet/cladding interactions (PCI). Other criteria may be more appropriate for an RIA, but continued approval of this enthalpy criterion and the thermal margin criteria may be given until generic studies yield improvements.

(g) Pellet/Cladding Interaction: There is no current criterion for fuel failure resulting from PCI, and the design basis can only be stated generally. Two related criteria should be applied, but they are not sufficient to preclude PCI failures. (1) The uniform strain of the cladding should not exceed 1%. In this context, uniform strain (elastic and inelastic) is defined as transient-induced deformation with gage lengths corresponding to cladding dimensions; steady-state creepdown and irradiation growth are excluded. Although observing this strain limit may preclude some PCI failures, it will not preclude the corrosion-assisted failures that occur at low strains, nor will it preclude highly localized overstrain failures. (2) 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. Such a PCI is avoided by avoiding fuel melting. Note that this same criterion was invoked in paragraph (e) to ensure that overheating of the cladding would not occur.

(h) Bursting: To meet the requirements of 10 CFR 50.46 as it relates to ECCS performance evaluation, 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 should be included in the ECCS evaluation model. Regulatory Guide 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 of 10 CFR 50 presents acceptable features of an evaluation model for predicting the degree of swelling and rupture in the Zircaloy cladding. ~~To meet the requirements of Appendix K of 10 CFR Part 50 (Ref. 9) as it relates to the incidence of rupture during a LOCA, a rupture temperature correlation must be used in the LOCA ECCS analysis. Zircaloy cladding will burst (rupture) under certain combinations of temperature, heating rate, and differential pressure.~~[39]Although fuel suppliers may use different rupture-temperature vs differential-pressure curves, an acceptable curve should be similar to the one described in Reference~~.~~[40] ~~10~~22.

(i) 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% 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.

3. <u>Fuel Coolability</u>

This subsection applies to postulated accidents, and most of the information to be reviewed will be contained in Chapter 15 of the Safety Analysis Report. Paragraph (e) addresses the combined effects of two accidents, however, and that information should be contained in Section 4.2 of the Safety Analysis Report. To meet the requirements of General Design Criteria 27 and 35 as they relate to control rod insertability and core coolability for postulated accidents, fuel coolability criteria should be given 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. Control rod insertability criteria are also addressed in this subsection. Such criteria should address the following to be complete:

(a) Cladding Embrittlement: To meet the requirements of 10 CFR Part 50, §50.46, as it relates to cladding embrittlement for a LOCA, acceptance criteria of 1204°C (2200°F)[41] on peak cladding temperature and 17% on maximum cladding oxidation must be met. (Note: If the cladding were predicted to collapse in a given cycle, it would also be predicted to fail and, therefore, should not be irradiated in that cycle; consequently, the lower peak cladding temperature limit of 982°C (1800°F)[42] previously described in Reference ~~11~~14[43] is no longer needed.) Similar temperature and oxidation criteria may be justified for other accidents.

(b) Violent Expulsion of Fuel: In severe reactivity initiated accidents, 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 to produce pressure pulses in the primary system. To meet the guidelines of Regulatory Guide 1.77 as it relates to preventing widespread fragmentation and dispersal of the fuel and avoiding the generation of pressure pulses in the primary system of a PWR, a radially averaged enthalpy limit of 1.17 KJ/g (280 cal/g)[44] should be observed. This 1.17 KJ/g (280 cal/g)[45] limit should also be used for BWRs.

(c) Generalized Cladding Melting: Generalized (i.e., non-local) melting of the cladding could result in the loss of rod-bundle fuel geometry. Criteria for cladding embrittlement in paragraph (a) above are more stringent than melting criteria would be; therefore, additional specific criteria are not used.

(d) Fuel Rod Ballooning: To meet the requirements of 10 CFR 50.46 as it relates to evaluating ECCS performance during accidents, burst strain and flow blockage caused by ballooning (swelling) of the cladding must be accounted for in the analysis of the core flow distribution. Regulatory Guide 1.157 describes models, correlations, data, and methods that are acceptable for meeting the requirements for a realistic calculation of ECCS performance during a LOCA. Alternatively, Appendix K to 10 CFR 50 presents acceptable features of a conservative evaluation model to consider burst strain and flow blockage.~~To meet the requirements of Appendix K of 10 CFR Part 50 as it relates to degree of swelling, burst strain and flow blockage resulting from cladding ballooning (swelling) must be taken into account in the analysis of core flow distribution.~~[46] Burst strain and flow blockage models must be based on applicable data (such as References~~.~~[47] ~~10~~22, ~~13~~27, and ~~12~~33) in such a way that (1) the temperature and differential pressure at which the cladding will rupture are properly estimated (see paragraph (h) of subsection II.A.2), (2) the resultant degree of cladding swelling is not underestimated, and (3) the associated reduction in assembly flow area is not underestimated.

The flow blockage model evaluation is provided to the Reactor Systems Branch for incorporation in the comprehensive ECCS evaluation model to show that the 1204°C (2200°F)[48] cladding temperature and 17% cladding oxidation limits are not exceeded. The reviewer should also determine if fuel rod ballooning should be included in the analysis of other accidents involving system depressurization.

(e) Structural Deformation: Analytical procedures are discussed in Appendix A, "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces."

B. <u>Description and Design Drawings</u>

The reviewer should see that the fuel system description and design drawings are complete enough to provide an accurate representation and to supply 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
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[49]

The following design drawings[50] have also been found 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
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. <u>Design Evaluation</u>

The methods of demonstrating that the design bases are met must be reviewed. Those 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.

1. Operating Experience

Operating experience with fuel systems of the same or similar design should be described, including the maximum burnup experience[51]. When adherence to specific design criteria can be conclusively demonstrated with operating experience, prototype testing and design analyses that were performed prior to gaining that experience need not be reviewed. Design criteria for fretting wear, oxidation, hydriding, and crud buildup might be addressed in this manner.

2. 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 prior to 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, 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 experience associated with such tests should also be reviewed and considered in relation to the specified maximum burnup limit for the new design.[52] The following phenomena that have been tested in this manner in new designs will serve as a guide to the reviewer:

Fuel and burnable poison rod growth
Fuel rod bowing
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 prior to operation of a full core of that design. This 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.D below).

3. <u>Analytical Predictions</u>

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.

(a) Fuel Temperatures (Stored Energy): Fuel temperatures and stored energy during normal operation are needed as input to ECCS performance calculations. The temperature calculations require complex computer codes that model many different phenomena. Regulatory Guide 1.157 describes models, correlations, data, and methods for a realistic calculation of ECCS performance during a LOCA and for estimating the uncertainty in that calculation. Alternatively, an ECCS evaluation model may be developed in conformance with the acceptable features of Appendix K of 10 CFR 50.[53] 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[*]

Because of the strong interaction between these models, overall code behavior must be checked against data (standard problems or benchmarks) and the NRC audit codes (References~~.~~[54] ~~14~~34 and ~~15~~35). Examples of

[*]Although needed in fuel performance codes, this model is reviewed as described in SRP Section 4.4.

previous fuel performance code reviews are given in References ~~16~~36[55] through ~~20~~40[56].

(b) Densification Effects: 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. Densification magnitudes for power spike and LHGR analyses are discussed in Reference ~~21~~16[57] and in Regulatory Guide 1.126 ~~(Ref. 22)~~[58]. To be acceptable, densification models should follow the guidelines of Regulatory Guide 1.126. Models for cladding-collapse times must also be reviewed, and previous review examples are given in References ~~23~~41[59] and ~~24~~42[60].

(c) Fuel Rod Bowing: Guidance for the analysis of fuel rod bowing is given in Reference ~~25~~43[61]. Interim methods that may be used prior to compliance with this guidance are given in Reference ~~26~~44[62]. At this writing, the causes of fuel rod bowing are not well understood and mechanistic analyses of rod bowing are not being approved.

(d) Structural Deformation: Acceptance Criteria are discussed in Appendix A, "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces."

(e) Rupture and Flow Blockage (Ballooning): Zircaloy rupture and flow blockage models are part of the ECCS evaluation model and should be reviewed by ~~CPB~~ SRXB[63]. The models are empirical and should be compared with relevant data. Examples of such data and previous reviews are contained in References ~~10~~22[64], ~~12~~27[65], and ~~13~~33[66].

(f) Fuel Rod Pressure: The thermal performance code for calculating temperatures discussed in paragraph (a) above should be used to calculate fuel rod pressures in conformance with fuel damage criteria of Subsection II.A.1, paragraph (f). The reviewer should ensure that conservatisms that were incorporated for calculating temperatures do not introduce nonconservatisms with regard to fuel rod pressures.

(g) Metal/Water Reaction Rate: To meet the requirements of 10 CFR 50.46 as it relates to the evaluation of 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. Regulatory Guide 1.157 allows the use of a best-estimate model provided its technical basis is demonstrated with appropriate data and analyses. Alternatively, Appendix K of 10 CFR 50.46 specifies that~~To meet the requirements of Appendix K of 10 CFR Part 50 (Ref. 9) as it relates to metal/water reaction rate,~~[67] the rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction should be calculated using the Baker-Just equation (Reference~~.~~[68] ~~27~~45). For non-LOCA applications, other correlations may be used if justified.

(h) Fission Product Inventory: To meet the guidelines of Regulatory Guides 1.3, 1.4, 1.25 and 1.77 ~~(Refs. 6, 28-30)~~[69] as they relate to fission product release, the available radioactive fission product inventory in fuel rods (i.e., the gap inventory) is presently specified by the assumptions in those Regulatory Guides. These assumptions should be used until improved calculational methods are approved by ~~CPB~~SRXB[70] (see Reference~~.~~[71] ~~31~~15). One such method currently approved is presented in ANS 5.4 (Reference 28) and 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 give the inventory of volatile fission products that could be available for release from the fuel rod if the cladding were breached.[72]

D. <u>Testing, Inspection, and Surveillance Plans</u>

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

1. <u>Testing and Inspection of New Fuel</u>

Testing and inspection plans for new fuel should include verification of cladding integrity, fuel system dimensions, fuel enrichment, burnable poison concentration, and absorber composition. Details of the manufacturer's testing and inspection programs should be documented in quality control reports, which 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. Where the overall testing and inspection programs are essentially the same as for previously approved plants, a statement to that effect should be made. In that case, the details of the programs need not be included in the Safety Analysis Report, but an appropriate reference should be cited and a (tabular) summary should be presented.

2. <u>On-line Fuel System Monitoring</u>

The applicant's on-line 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. References ~~32~~19[73] and ~~33~~26[74] evaluate several common detection methods and should be utilized 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 described in Reference ~~34~~17[75] are acceptable.

3. <u>Post-irradiation Surveillance</u>

A post-irradiation 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 like 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, or crud deposition. There should also be a commitment in the program 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, there should exist a continuing fuel surveillance effort 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 prototype testing discussed in subsection II.C.2. When prototype testing cannot be performed, a special detailed surveillance program should be planned for the first irradiation of a new design.

Technical Rationale:[76]

The technical rationale for application of the above acceptance criteria to the fuel system design is discussed in the following paragraphs:

1. 10 CFR Part 50, §50.46 requires each PWR and BWR to be provided with an emergency core cooling system (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, §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. Acceptable methods are presented in Regulatory Guide 1.157, or alternatively Appendix K to 10 CFR 50, to evaluate the performance of the ECCS. Regulatory Guide 1.126 provides an acceptable model for predicting the effects of fuel densification in commercial light water reactors. Application of acceptance criteria established in 10 CFR 50, §50.46 significantly reduces the possibility of a violent chemical reaction occurring between the Zircaloy cladding and the coolant, which would result 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 the core.

2. 10 CFR Part 100 requires that exposure to an individual caused by the release of fission products to the environment during a postulated reactor accident be calculated, and that the result be considered when determining the acceptability of a reactor site. Acceptable fission gas release models which are necessary for performing radiological dose

calculations are discussed in this section and ensure that doses are not underestimated. Regulatory Guides 1.3 and 1.4 provide acceptable assumptions that may be used in evaluating the radiological consequences associated with a LOCA for BWRs and PWRs respectively. Regulatory Guide 1.25 provides acceptable assumptions that may be used in evaluating the radiological consequences associated with a fuel handling accident at a Fuel Handling and Storage Facility at reactor sites. And Regulatory Guide 1.77 identifies acceptable analytical methods and assumptions that may be used in evaluating 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 the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. 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 anticipated operational occurrences. Design limits are specified in Section 4.2 to accomplish this objective. Compliance with GDC 10 significantly reduces the likelihood of fuel failures during normal operations or anticipated operational occurrences, thereby minimizing the possible release of fission products.

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 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 under postulated accident conditions. 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 high enough temperature to allow a significant metal-water reaction to occur, thereby minimizing the potential for off-site release.

III. REVIEW PROCEDURES

For construction permit (CP) applications, the review should assure that the design bases set forth in the Preliminary Safety Analysis Report (PSAR) meet the acceptance criteria given in subsection II.A. The CP review should further 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.

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 reviewed 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 is dependent should be referenced so that a completely documented safety evaluation is contained in the plant safety evaluation report. In particular, the NRC safety evaluation reports for all relevant licensing topical reports should be cited. Certain generic reviews have also been performed by ~~CPB~~ SRXB[77] reviewers with findings issued as NUREG- or WASH-series reports. At the present time these reports include References ~~9,~~[78] ~~11~~14, ~~31~~15, ~~21~~16, ~~35~~18, ~~32~~19, and ~~36~~20[79], and they should all be appropriately cited in the plant safety evaluation report. Applicable Regulatory Guides (References~~.~~ ~~6~~7-13~~, 22, 28-30, and 41~~[80])[81] should also be mentioned in the plant safety evaluation reports. Deviation from these guides or positions should be explained. After briefly discussing related previous reviews, the plant safety evaluation should concentrate on areas where the application is not identical to previously reviewed and approved applications and areas related to newly discovered problems.

Analytical predictions discussed in Subsection II.C.3 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 limit, should be reviewed. Fuel burnup 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 need be provided.[82] When the methods are being reviewed, calculations by the staff may be performed to verify the adequacy of the analytical methods. Thereafter, audit calculations will not usually be performed to check 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 the clear need arises to reconfirm the adequacy of the method.

For standard design certification reviews under 10 CFR Part 52, the procedures above should be followed, as modified by the procedures in SRP Section 14.3 (proposed), to verify that the design set forth in the standard safety analysis report, including inspections, tests, analysis, and acceptance criteria (ITAAC), site interface requirements and combined license action items, meet the acceptance criteria given in subsection II. SRP Section 14.3 (proposed) contains procedures for the review of certified design material (CDM) for the standard design, including the site parameters, interface criteria, and ITAAC.[83]

IV. EVALUATION FINDINGS

The reviewer should verify that sufficient information has been provided to satisfy the requirements of this SRP section and that the evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

> The staff concludes that the fuel system of the ________ plant has been designed so that (a) the fuel system will not be damaged as a result of normal operation and anticipated operational occurrences, (b) fuel damage during postulated accidents would not be severe enough to prevent control rod insertion when it is required, and (c) core coolability will always be maintained, even after severe postulated accidents and thereby meets the related requirements of 10 CFR Part 50, §50.46; 10 CFR Part 50, Appendix A, General

Design Criteria 10, 27 and 35;~~ 10 CFR Part 50, Appendix K;~~[84] and 10 CFR Part 100. 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, control rod ejection (PWR) or drop (BWR), and fuel densification have been performed in accordance with (a) the guidelines of Regulatory Guides 1.60, 1.77, and 1.126, or methods that the staff has reviewed and found to be acceptable alternatives to those Regulatory Guides, and (b) the guidelines for "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces" in Appendix A 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 on-line 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. In meeting these requirements, the applicant has (a) used the fission-product release assumptions of Regulatory Guides 1.3 (or 1.4), 1.25, and 1.77 and (b) performed the analysis for fuel rod failures for the rod ejection accident in accordance with the guidelines of Regulatory Guide 1.77 or with methods that the staff has reviewed and found to be an acceptable alternative to Regulatory Guide 1.77.

For design certification reviews, the findings will also summarize, to the extent that the review is not discussed in other safety evaluation report sections, the staff's evaluation of inspections, tests, analyses, and acceptance criteria (ITAAC), including design acceptance criteria (DAC), site interface requirements, and combined license action items that are relevant to this SRP section.[85]

V. IMPLEMENTATION

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

This SRP section will be used by the staff when performing safety evaluations of license applications submitted by applicants pursuant to 10 CFR 50 or 10 CFR 52.[86] Except in those cases in which the applicant roposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications docketed six months or more after the date of issuance of this SRP section.[87]

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

VI. REFERENCES[88]

1~~3~~. 10 CFR Part §50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."

2~~2~~. 10 CFR Part 100, "Reactor Site Criteria."

3~~1~~. 10 CFR Part 50, Appendix A, ~~"General Design Criteria for Nuclear Power Plants."~~General Design Criterion 10, "Reactor Design."[89]

4. 10 CFR Part 50, Appendix A, General Design Criterion 27, "Combined Reactivity Control Systems Capability."

5. 10 CFR Part 50, Appendix A, General Design Criterion 35, "Emergency Core Cooling."

6~~9~~. 10 CFR Part 50, Appendix K, "ECCS Evaluation Models."

7~~28~~. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors."

8~~29~~. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of Loss-of-Coolant Accident for Pressurized Water Reactors."

9~~30~~. 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."

10~~41~~. Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants."

11~~6~~. Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors."

12~~22~~. Regulatory Guide 1.126, "An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification."

13. Regulatory Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance."[90]

14~~11~~. WASH-1236 "Technical Report on Densification of Light Water Reactor Fuels," AEC Regulatory Staff Report~~ WASH-1236~~[91], November 14, 1972.

15~~31~~. NUREG-75/077, "The Role of Fission Gas Release in Reactor Licensing," ~~USNRC Report NUREG-75/077, ~~[92]November 1975.

16~~21~~. NUREG-0085~~R. O. Meyer~~, "The Analysis of Fuel Densification," ~~USNRC Report NUREG-0085, ~~July 1976.

17~~34~~. NUREG-0308 Supp. 2, "Safety Evaluation Report Related to Operation of Arkansas Nuclear One, Unit 2," ~~USNRC Report NUREG-0308, Supp. 2, ~~September 1978.

18~~35~~. NUREG-0303~~B. L. Siegel~~[93], "Evaluation of the Behavior of Waterlogged Fuel Rod Failures in LWRs," ~~USNRC Report NUREG-0308,~~ March 1978.

19~~32~~. NUREG-0401~~B. L. Siegel and H. H. Hagen~~, "Fuel Failure Detection in Operating Reactors," ~~USNRC Report NUREG-0401,~~ March 1978.

20~~36~~. NUREG-0418~~R. O. Meyer, C. E. Beyer and J. C. Voglewede~~, "Fission Gas Release from Fuel at High Burnup," ~~USNRC Report NUREG-0418,~~ March 1978.

21~~40~~. NUREG-0609~~S. B. Hosford, et al.~~, "Asymmetric Blowdown Loads on PWR Primary Systems," ~~USNRC Report NUREG-0609,~~ January 1981.

22~~10~~. NUREG-0630~~D. A. Powers and R. O. Meyer~~, "Cladding Swelling and Rupture Models for LOCA Analysis," ~~USNRC Report NUREG-0630,~~ April 1980.

23~~37~~. NUREG/CR-1018~~R. L. Grubb~~, "Review of LWR Fuel System Mechanical Response with Recommendations for Component Acceptance Criteria," ~~Idaho National Engineering Laboratory, NUREG/CR-1018,~~ September 1979.

24~~38~~. NUREG/CR-1019~~R. L. Grubb~~, "Pressurized Water Reactor Lateral Core-Response Routine, FAMREC (Fuel Assembly Mechanical Response Code)," ~~Idaho National Engineering Laboratory, NUREG/CR-1019,~~ September 1979.

25~~39~~. NUREG/CR-1020~~R. L. Grubb~~, "Technical Evaluation of PWR Fuel Spacer Grid Response Load Sensitivity Studies," ~~Idaho National Engineering Laboratory, NUREG/CR-1020,~~ September 1979.

26~~33~~. NUREG/CR-1380~~W. J. Bailey, et al.~~, "Assessment of Current Onsite Inspection Techniques for LWR Fuel Systems," ~~Battelle Pacific Northwest Laboratory Report NUREG/CR-1380,~~ Vol. 1, July 1980, Vol. 2, January 1981.

27~~13~~. NUREG/CR-1883~~R. H. Chapman~~, "Multirod Burst Test Program Progress Report for January-June 1980," ~~Oak Ridge National Laboratory Report NUREG/CR-1883,~~ March 1981.

28. American National Standard Institute, ANSI/ANS 5.4, "Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel," American Nuclear Society, November, 10, 1982.[94]

29~~4~~. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, ~~1977 edition,~~ [95]Section III, "Rules for Construction of Nuclear Power Plant Components," ~~ASME Boiler and Pressure Vessel Code, Section III, 1977~~New York.

30~~7~~. American Society for Testing and Materials, 1989~~77~~ edition, Standard C776-89~~76~~[96], Part 45, "Standard Specification for Sintered Uranium Dioxide Pellets," Philadelphia.

31~~5~~. W. J. O'Donnel and B. F. Langer, "Fatigue Design Basis for Zircaloy Components," Nucl. Sci. Eng. 20, 1 (1964).

32~~8~~. K. Joon, "Primary Hydride Failure of Zircaloy-Clad Fuel Rods," Trans. Am. Nucl. Soc. 15, 186 (1972).

33~~12~~. F. Erbacher, H. J. Neitzel, H. Rosinger, H. Schmidt, and K. Wiehr, "Burst Criterion of Zircaloy Fuel Claddings in a LOCA," ASTM Fifth International Conference on Zirconium in the Nuclear Industry, August 4-7, 1980, Boston, Massachusetts.

34~~14~~. M. E. Cunningham and C. E. Beyer, "GT2R2: An Updated Version of GAPCON-THERMAL-2," Pacific Northwest Laboratory Report PNL-5178, September 1984.~~C. E. Beyer, C. R. Hann, D. D. Lanning, F. E. Panisko and L. J. Parchen, "User's Guide for GAPCON-THERMAL-2: A Computer Program for Calculating the Thermal Behavior of an Oxide Fuel Rod," Battelle Pacific Northwest Laboratory Report BNWL-1897, November 1975.~~[97]

35~~15~~. C. E. Beyer, C. R. Hann, D. D. Lanning, F. E. Panisko and L. J. Parchen, "GAPCON-THERMAL-2: A Computer Program for Calculating the Thermal Behavior of an Oxide Fuel Rod," Battelle Pacific Northwest Laboratory Report BNWL-1898, November 1975.

36~~16~~. R. H. Stoudt, D. T. Buchanan, B. J. Buescher, L. L. Losh, H. W. Wilson and P. J. Henningson, "TACO - Fuel Pin Performance Analysis, Revision 1," Babcock & Wilcox Report BAW-10087A, Rev. 1, August 1977.[98]

37~~17~~. "Fuel Evaluation Model," Combustion Engineering Report CENPD-139-A, July 1974 (Approved version transmitted to NRC April 25, 1975).[99]

38~~18~~. "Supplement 1 to the Technical Report on Densification of General Electric Reactor Fuels," AEC Regulatory Staff Report, December 14, 1973.[100]

39~~19~~. "Technical Report on Densification of Exxon Nuclear PWR Fuels," AEC Regulatory Staff Report, February 27, 1975.[101]

40~~20~~. Letter from J. F. Stolz, NRC, to T. M. Anderson, Westinghouse, Subject: Safety Evaluation of WCAP-8720, dated February 9, 1979.[102]

41~~23~~. Memorandum from V. Stello, NRC, to R. C. DeYoung, Subject: Evaluation of Westinghouse Report, WCAP-8377, Revised Clad Flattening Model, dated January 14, 1975.[103]

42~~24~~. Memorandum from D. F. Ross, NRC, to R. C. DeYoung, Subject: CEPAN -- Method of Analyzing Creep Collapse of Oval Cladding, dated February 5, 1976.[104]

43~~25~~. Memorandum from D. F. Ross, NRC, to D. B. Vassallo, Subject: Request for Revised Rod Bowing Topical Reports, dated May 30, 1978.

44~~26~~. 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.[105]

45~~27~~. 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.

APPENDIX A
EVALUATION OF FUEL ASSEMBLY STRUCTURAL RESPONSE
TO EXTERNALLY APPLIED FORCES
TO
STANDARD REVIEW PLAN SECTION 4.2

A. 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. Background material for this Appendix is given in References ~~37-40~~21 and 23-25[106].

B. ANALYSIS OF LOADS

1. Input

Input for the fuel assembly structural analysis comes from results of the primary coolant system and reactor internals structural analysis, which is reviewed by the Mechanical Engineering Branch. Input for the fuel assembly response to a LOCA should include (a) 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 (b) transient pressure differences that apply loads directly to the fuel assembly. If the earthquake loads are large enough to produce a non-linear fuel assembly response, input for the seismic analysis should use structure motions corresponding to the reactor primary coolant system analysis for the SSE; if a linear response is produced, a spectral analysis may be used in accordance with the guidelines of Regulatory Guide 1.60 ~~(Ref. 41)~~[107].

2. Methods

Analytical methods used in performing structural response analyses should be reviewed. Justification should be supplied to show that the numerical solution techniques are appropriate.

Linear and non-linear 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.

A sample problem of a simplified nature should be worked by the applicant and compared by the reviewer with either hand calculations or results generated by the reviewer with an independent code (Reference~~.~~[108] ~~38~~24). 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), simplifying assumptions may be made (e.g., one might use a 3-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 probably conservative and input parameters are also conservative. However, to ensure that the fuel assembly analysis does not introduce any non-conservatisms, two precautions should be taken: (a) If it is not explicitly evaluated, impact loads from the PWR LOCA analysis should be increased (by about 30%) to account for a pressure pulse, which is associated with steam flashing that affects only the PWR fuel assembly analysis. (b) 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 +10% variations in input amplitude and frequency; 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%. For example, if +10% variations in input magnitude or frequency produce a maximum resultant increase of 35%, 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

Independent audit calculations for a typical full-sized core should be performed by the reviewer to verify that the overall structural representation is adequate. An independent audit code (Reference~~.~~[109] ~~38~~24) should be used for this audit during the generic review of the analytical methods.

5. Combination of Loads

To meet the requirements of General Design Criterion 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 above in Paragraph 1, and the resulting loads should be added by the square-root-of-sum-of-squares (SRSS) method. These combined loads should be compared with the component strengths described in Section C according to the acceptance criteria in Section D.

C. <u>DETERMINATION OF STRENGTH</u>

1. Grids

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

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% 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%) of the distribution of component strengths. Therefore, ASME Boiler and Pressure Vessel Code values and procedures may be used where appropriate for determining yield and ultimate strengths. Specification of allowable values may follow the ASME Code requirements and should include consideration of buckling and fatigue effects.

D. <u>ACCEPTANCE CRITERIA</u>

1. Loss-of-Coolant Accident

Two principal criteria apply for the LOCA: (a) fuel rod fragmentation must not occur as a direct result of the blowdown loads, and (b) the 10 CFR Part 50, §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 control rod insertion would not be interfered with by lateral displacement of the guide tubes. If combined loads on the grids exceed P(crit), then additional analysis is needed to show that deformation is not severe enough to prevent control rod insertion.

For a BWR, several conditions must be met to demonstrate control blade insertability: (a) 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 deformation is not severe enough to prevent control blade insertion, and (b) 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 for the SSE: (a) fuel rod fragmentation must not occur as a result of the seismic loads, and (b) control rod insertability must be assured. The first criterion is satisfied by the criteria in Paragraph 1. The second criterion must be satisfied for SSE loads alone if no analysis for combined loads is required by Paragraph 1.

SRP Draft Section 4.2

Attachment A - Proposed Changes in Order of Occurrence

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

Item	Source	Description
1.	SRP-UDP Format Item, Update PRB names.	Changed PRB name to reflect latest responsibility assignments for SRP section 4.2.
2.	SRP-UDP Format Item, Update PRB names.	Changed PRB name to reflect latest responsibility assignments for SRP section 4.2.
3.	Editorial	This information was relocated to the Review Interface section.
4.	SRP-UDP Format Item, update reference citations.	Revised reference citation to be consistent with SRP-UDP required format that excludes parenthetical notation for CFR and GDC citations.
5.	SRP-UDP Format Item, update reference citations.	Revised reference citation to be consistent with SRP-UDP required format that excludes parenthetical notation for CFR and GDC citations.
6.	SRP-UDP Format Item, update reference citations.	Revised reference citation to be consistent with SRP-UDP required format that excludes parenthetical notation for CFR and GDC citations.
7.	SRP-UDP Format Item, Update PRB names.	Changed PRB name to reflect the current name of the branch which reviews the radiological consequence estimates associated with the accident analyses.
8.	SRP-UDP Format Item, Update PRB names.	Changed PRB name to reflect latest responsibility assignments for SRP section 4.2.
9.	SRP-UDP format item, Reformat Areas of Review	Added Review Interface heading to Areas of Review. Review interfaces did not exist in SRP 4.2, so appropriate interfaces were developed from statements in the acceptance criteria and review procedures.
10.	SRP-UDP format item, Reformat Areas of Review	A review interface with SRP Section 4.3 was adapted from existing Acceptance Criteria II.C.3(b).
11.	SRP-UDP format item, Reformat Areas of Review	A review interface with SRP Section 4.4 was adapted from existing Acceptance Criteria II.A.1(d), II.A.1(g), and II.A.2(d).
12.	SRP-UDP format item, Reformat Areas of Review	A review interface with SRP Section 6.3 was adapted from existing Acceptance Criteria II.1 and II.A.2.(h) regarding ECCS performance.

Item	Source	Description
13.	SRP-UDP format item, Reformat Areas of Review	A review interface with SRP Chapter 15 was adapted from existing Acceptance Criteria II.A.2.
14.	SRP-UDP format item, Editorial	Added a review interface with the EMEB. This interface was relocated from the first paragraph of the existing Areas of Review for SRP Section 4.2.
15.	SRP-UDP format item, Editorial	Added a review interface with the PERB. This interface was adopted from the fourth paragraph of the existing Areas of Review for SRP Section 4.2.
16.	Editorial	Added typical lead-in paragraph for Acceptance Criteria subsections to make SRP Section 4.2 consistent with other SRP sections.
17.	Editorial	Added acceptance criteria relating to requirements established by 10 CFR 50.46. The acceptance criteria was adapted from the existing Areas of Review discussion.
18.	Editorial	Added acceptance criteria relating to requirements established by 10 CFR 100. The acceptance criteria was adapted from the existing Areas of Review discussion.
19.	Editorial	Added acceptance criteria relating to requirements established by GDC 10. The acceptance criteria was adapted from the existing Areas of Review discussion.
20.	Editorial	Added acceptance criteria relating to requirements established by General Design Criterion 27. The acceptance criteria was adapted from the existing Areas of Review discussion.
21.	Editorial	Added acceptance criteria relating to requirements established by General Design Criterion 35. The acceptance criteria was adapted from the existing Areas of Review discussion.
22.	**Integrated Impact 556**	Deleted "Appendix K to 10 CFR Part 50" from the list of acceptance criteria since it is now only one of two options allowed by 10 CFR 50.46.
23.	Editorial.	The text was modified to refer to "relevant" requirements and the phrase "identified in subsection I of this SRP section" was deleted as unnecessary.

Item	Source	Description
24.	**Integrated Impact 558**	Added a third sentence to explicitly state that high burnup effects should be considered when addressing fuel system damage criteria.
25.	SRP-UDP Format Item, update reference citations.	Revised reference citation to be consistent with SRP-UDP required format that specifies spelling out the word Reference. Changed reference number to agree with changes in the Reference subsection numbering.
26.	SRP-UDP Format Item, update reference citations.	Revised reference citation to be consistent with SRP-UDP required format that specifies spelling out the word Reference. Changed reference number to agree with changes in the Reference subsection numbering.
27.	**Integrated Impact 558**	Added a sentence to explicitly state that high burnup effects should be considered when addressing fuel rod failure criteria.
28.	Editorial	This discussion regarding hydriding pertains only to internal hydriding and therefore should be labeled as such (as opposed to external hydriding). External hydriding would be a separate failure mechanism which is currently not addressed in this SRP Section.
29.	Editorial.	Added the words "within the fuel" for clarity.
30.	SRP-UDP format item - NRC Metrication policy implementation.	Converted 20 ppm to 20 µg/g and placed 20 ppm in parentheses.
31.	SRP-UDP Format Item, update reference citations.	Revised reference citation to be consistent with SRP-UDP required format that specifies spelling out the word Reference. Changed reference number to agree with changes in the Reference subsection numbering.
32.	SRP-UDP format item - NRC Metrication policy implementation.	Converted 2 ppm to 2 µg/g and placed 2 ppm in parentheses.
33.	SRP-UDP Format Item, update reference citations.	Revised reference citation to be consistent with SRP-UDP required format that specifies spelling out the word Reference. Changed reference number to agree with changes in the Reference subsection numbering.
34.	GSI B-3 Resolution	Corrected "abnormal operational occurrence" to "anticipated operational occurrence."

Item	Source	Description
35.	Editorial	Added the word "these" for clarification since transients refers to both AOOs and accidents mentioned in the preceding sentence.
36.	SRP-UDP format item - NRC Metrication policy implementation.	Converted 170 cal/g to 711 J/g and placed 170 cal/g in parentheses.
37.	SRP-UDP Format Item, update reference citations.	Revised reference citation to be consistent with SRP-UDP required format that excludes parenthetical notation for Regulatory Guide citations.
38.	SRP-UDP format item - NRC Metrication policy implementation.	Converted 170 cal/g to 711 J/g and placed 170 cal/g in parentheses.
39.	**Integrated Impact 556**	Revised paragraph A.2.(h) to consider either; 1) realistic calculations using guidance in RG 1.157, or 2) conservative assumptions in Appendix K to 10 CFR 50.
40.	SRP-UDP Format Item, update reference citations.	Revised reference citation to be consistent with SRP-UDP required format that specifies spelling out the word Reference. Changed reference number to agree with changes in the Reference subsection numbering.
41.	SRP-UDP format item - NRC Metrication policy implementation.	Converted 2200°F to 1204°C and placed 2200°F in parentheses.
42.	SRP-UDP format item - NRC Metrication policy implementation.	Converted 1800°F to 982°C and placed 1800°F in parentheses.
43.	Editorial	Revised the reference number to reflect the re-numbering of the list of references.
44.	SRP-UDP format item - NRC Metrication policy implementation.	Converted 280 cal/g to 1.17 KJ/g and placed 280 cal/g in parentheses.
45.	SRP-UDP format item - NRC Metrication policy implementation.	Converted 280 cal/g to 1.17 KJ/g and placed 280 cal/g in parentheses.
46.	**Integrated Impact 556**	Replaced the first sentence of A.3.(d) to consider either: 1) realistic calculations using guidance in RG 1.157, or 2) conservative assumptions in Appendix K to 10 CFR 50.

Item	Source	Description
47.	SRP-UDP Format Item, update reference citations.	Revised reference citation to be consistent with SRP-UDP required format that specifies spelling out the word Reference. Changed reference numbers to agree with changes in the Reference subsection numbering.
48.	SRP-UDP format item - NRC Metrication policy implementation.	Converted 2200°F to 1204°C and placed 2200°F in parentheses.
49.	**Integrated Impact 558.**	Added the words "Design Specific Burnup Limit" to the list of fuel system information to be reviewed.
50.	Editorial	Added "s" to the word drawing to make it plural.
51.	**Integrated Impact 558**	Added the words, "including the maximum burnup experience" to the end of the sentence to explicitly require that burnup experience be given.
52.	**Integrated Impact 558**	Added a sentence to consider the burnup experience associated with prototype test assemblies.
53.	**Integrated Impact 556**	Added a third and fourth sentence to paragraph C.3.(a) to reference guidance presented in RG 1.157. Regulatory Guide 1.157 provides additional references to applicable stored energy models.
54.	SRP-UDP Format Item, update reference citations.	Revised reference citation to be consistent with SRP-UDP required format that specifies spelling out the word Reference. Changed reference numbers to agree with changes in the Reference subsection numbering.
55.	Editorial	Changed reference number to agree with changes in the Reference subsection numbering.
56.	Editorial	Changed reference number to agree with changes in the Reference subsection numbering.
57.	Editorial	Changed reference number to agree with changes in the Reference subsection numbering.
58.	SRP-UDP Format Item, update reference citations.	Revised reference citation to be consistent with SRP-UDP required format that excludes parenthetical notation for Regulatory Guide citations.
59.	Editorial	Changed reference number to agree with changes in the Reference subsection numbering.

Item	Source	Description
60.	Editorial	Changed reference number to agree with changes in the Reference subsection numbering.
61.	Editorial	Changed reference number to agree with changes in the Reference subsection numbering.
62.	Editorial	Changed reference number to agree with changes in the Reference subsection numbering.
63.	SRP-UDP Format Item, Update PRB names.	Changed PRB name to reflect latest responsibility assignments for SRP section 4.2.
64.	Editorial	Changed reference number to agree with changes in the Reference subsection numbering.
65.	Editorial	Changed the reference number 13 to agree with changes in the Reference subsection numbering and moved the reference to allow the references to be cited in numerical order.
66.	Editorial	Changed the reference number 12 to agree with changes in the Reference subsection numbering and moved the reference to allow the references to be cited in numerical order.
67.	**Integrated Impact 556**	Revise paragraph C.3.(g) to consider either: 1) realistic calculations using guidance in RG 1.157, or 2) conservative assumptions in Appendix K to 10 CFR 50.
68.	SRP-UDP Format Item, update reference citations.	Revised reference citation to be consistent with SRP-UDP required format that specifies spelling out the word Reference. Changed reference number to agree with changes in the Reference subsection numbering.
69.	SRP-UDP Format Item, update reference citations.	Revised reference citation to be consistent with SRP-UDP required format that excludes parenthetical notation for Regulatory Guide citations.
70.	SRP-UDP Format Item, Update PRB names.	Changed PRB name to reflect latest responsibility assignments for SRP section 4.2.
71.	SRP-UDP Format Item, update reference citations.	Revised reference citation to be consistent with SRP-UDP required format that specifies spelling out the word Reference. Changed reference number to agree with changes in the Reference subsection numbering.

Item	Source	Description
72.	**Integrated Impact 1322**	Added sentences to discuss the acceptable use of volatile fission product release models presented in ANS 5.4.
73.	Editorial	Changed reference number to agree with changes in the Reference subsection numbering.
74.	Editorial	Changed reference number to agree with changes in the Reference subsection numbering.
75.	Editorial	Changed reference number to agree with changes in the Reference subsection numbering.
76.	SRP-UDP format item, Develop Technical Rationale.	Added Technical Rationale heading to Acceptance Criteria subsection and developed Technical Rationale for 10 CFR 50.46, 10 CFR 50 Appendix K, 10 CFR 100 and GDCs 10, 27, and 35.
77.	SRP-UDP Format Item, Update PRB names.	Changed PRB name to reflect latest responsibility assignments for SRP section 4.2.
78.	Editorial	Removed reference 9 (new reference 5), "10 CFR Part 50, Appendix K," because it is not a generic review or NUREG/WASH report and therefore appears to be inappropriately cited here.
79.	Editorial	Changed reference numbers to agree with changes in the Reference subsection numbering.
80.	SRP-UDP Format Item, update reference citations.	Revised reference citation to be consistent with SRP-UDP required format that specifies spelling out the word Reference and changed reference numbers to agree with changes to the Reference subsection numbering.
81.	**Integrated Impact 556**	Added Reference 11 (RG 1.157) to the list of RGs that are to be mentioned in the plant safety evaluation report.
82.	**Integrated Impact 558**	Added text to review procedures to specify that the review consider burnup limits and their applicability to the analytical models used to predict the performance of the fuel system.
83.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard paragraph to address application of Review Procedures in design certification reviews.

Item	Source	Description
84.	**Integrated Impact 556**	Removed "10 CFR Part 50, Appendix K" from the evaluation findings, since it is now one of two options allowed by 10 CFR 50.46.
85.	SRP-UDP format item, 10 CFR 52 Applicability	Added statement to Evaluation Findings addressing the findings associated with design certification reviews.
86.	SRP-UDP format item, Applicability	Added statement to Implementation subsection to address 10 CFR requirements and section applicability.
87.	SRP-UDP format item, Applicability	Added statement to Implementation subsection to address 10 CFR requirements and section applicability.
88.	SRP-UDP format item.	Reorganized the entire list of references to conform to a format consistent with the Supplemental Guidance.
89.	Editorial.	Added a specific reference to GDC 10 to be consistent with the other sections. Also, added specific citations for GDC 27 and 35.
90.	**Integrated Impact 556**	Added Regulatory Guide 1.157 to the list of References since the guide has been added to this SRP section.
91.	Editorial	Moved WASH-1236 to the beginning of the reference to be consistent with the other references.
92.	Editorial	Changed the citation format of the NUREGs to be consistent with the way NUREGs are cited in other SRP Sections.
93.	Reference Verification, update SRP reference citations.	Changed NUREG-0308 to read NUREG-0303 to correct the typographical error.
94.	**Integrated Impact 1322**	Added ANS 5.4 to the list of References.
95.	SRP-UDP item, update SRP reference citations.	Removed reference to the 1977 edition since the applicable edition of ASME Boiler and Pressure Vessel Code is specified in 10 CFR 50.55a, "Codes and Standards." Also, formatted the citation to conform to requirements in NUREG-0650, Rev. 1.
96.	**Integrated Impact 557**	Updated reference 27 to reflect the lasted edition to ASTM Standard C776.

Item	Source	Description
97.	Reference Verification, update SRP reference citations.	Removed reference to BNWL-1897 report and replaced it with the latest revision to the GT2R2 code/users guide. Some of the analytical models used in GT2R2 were revised, and as a result, PNL-5178 was issued in 1984 to document the most recent revisions to the code.
98.	Unverified Reference.	The reference is a vendor report and could not be verified.
99.	Unverified Reference.	The reference is a vendor report and could not be verified.
100.	Unverified Reference.	The reference is a AEC Staff report and could not be verified.
101.	Unverified Reference.	The reference is a AEC Staff report and could not be verified.
102.	Unverified Reference.	The reference is a vendor report and could not be verified.
103.	Unverified Reference.	The reference is a NRC memorandum and could not be verified.
104.	Unverified Reference.	The reference is a NRC memorandum and could not be verified.
105.	Unverified Reference.	The reference is a NRC memorandum and could not be verified.
106.	Editorial	Changed reference numbers to agree with changes in the Reference subsection numbering.
107.	SRP-UDP Format Item, update reference citations.	Revised reference citation to be consistent with SRP-UDP required format that excludes parenthetical notation for Regulatory Guide citations.
108.	SRP-UDP Format Item, update reference citations.	Revised reference citation to be consistent with SRP-UDP required format that specifies spelling out the word Reference. Changed reference number to agree with changes in the Reference subsection numbering.
109.	SRP-UDP Format Item, update reference citations.	Revised reference citation to be consistent with SRP-UDP required format that specifies spelling out the word Reference. Changed reference number to agree with changes in the Reference subsection numbering.

[This Page Intentionally Left Blank]

SRP Draft Section 4.2
Attachment B - Cross Reference of Integrated Impacts

Integrated Impact No.	Issue	SRP Subsections Affected
556	Revise Acceptance Criteria and Review Procedures to incorporate revisions to 10 CFR 50.46 that allows the use of best-estimate calculational techniques for evaluating the performance of the ECCS.	II, III, IV, and VI
557	Update ASTM C776-76 to the current version.	VI
558	Modify Acceptance Criteria and Review Procedures to address fuel burnup specifications and justification.	II and III
1322	Modify acceptance criteria to include the use of analytical models presented in ANSI/ANS 5.4.	II and VI