U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
PROPOSED REVISION
TO
STANDARD REVIEW PLAN PSRP-4.2, REVISION 2, DRAFT 1

SECTION 4.2 FUEL SYSTEM DESIGN

REVIEW RESPONSIBILITIES

Primary - Core Performance Branch (CPB)
Secondary - None

I. AREAS OF REVIEW

The thermal, mechanical, and materials design of the fuel system is evaluated by CPB. 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; guide tubes and other non-fueled tubes; 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.

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. "Fuel rod failure" means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached. Coolability, in general, means that the fuel assembly retains its rodbundle geometry with adequate coolant channels to permit removal of residual heat after accidents analyzed in Chapter 15.

This proposed revision of the Standard Review Plan and the support value/impact statement have not received a complete staff review and approval and do not represent an official NRC staff position. Public comments are being solicited on both the revision and the value/impact statement (including any implementation schedules) prior to a review by the Regulatory Requirements Review Committee and their recommendation as to whether this revision should be approved. Comments should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch. All comments received by ________________ will be considered by the Regulatory Requirements Review Committee. A summary of the meeting of the Committee at which this revision is considered, the Committee recommendations and all of the associated documents and comments considered by the Committee will be made publicly available prior to a decision by the Director, Office of Nuclear Reactor Regulation, on whether to implement this revision.

Fuel failure criteria and coolability criteria that involve thermal-hydraulic considerations are provided by the Core Performance Branch to the Analysis Branch for implementation in SRP Section 4.4. The Analysis Branch provides hydraulic loads under SRP Section 4.4 to the Core Performance Branch for evaluation (in SRP Section 4.2) of fuel assembly mechanical response under normal and accident conditions. The available radioactive fission product inventory in fuel rods (i.e., the gap inventory expressed as a release fraction) is provided to the Accident Analysis Branch for use in estimating the radiological consequences of plant releases.

The fuel system review covers the following specific areas.

A. Design Bases

The principles and related assumptions of the fuel system design should be reviewed. These bases may be expressed as explicit numbers or as general criteria. The bases will include traditional fuel design limits, industry codes and standards, and limits related to the safety analysis (i.e., related to fuel damage, rod failure, or coolability requirements). Once such limits are approved in the safety evaluation report, they become the specified acceptable fuel design limits referred to in General Design Criterion 10 (Ref. 1). The design bases should reflect the safety review objectives as described above.

B. Description and Design Drawings

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

C. Design Evaluation

The performance of the fuel system during normal operation, anticipitated 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. Testing, Inspection, and Surveillance Plans

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 to ensure that the important fuel design considerations have been addressed.

II. ACCEPTANCE CRITERIA

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

Fuel system da ge 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. Such damage criteria should include 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 (Ref. 2) 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 (Ref. 3). 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 by the Analysis Branch 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 provided by the Analysis Branch 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. Fuel Rod Failure

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 known failure mechanisms during normal operation and anticipated operational occurrences. Fuel rod failures can be caused by overheating, pellet/cladding interaction (PCI), hydriding, cladding collapse, bursting, mechanical fracturing, and fretting. A fuel failure criterion should be given for each known failure mechanism. Such criteria should address the following to be complete.

(a) Overheating: No useful mechanistic criteria exist at present for fuel rod failure due to overheating. However, to show that overheating will be avoided, it will be sufficient to show that (1) cladding temperatures do not greatly exceed the coolant temperature and (2) fuel melting does not occur.

Adequate cooling is assumed to exist when the thermal margin criterion to limit the departure from nucleate boiling (DNB) or boiling transition condition in the core is satisfied. The review of this criterion is detailed in SRP Section 4.4.

For a severe reactivity initiated accident (RIA), Regulatory Guide 1.77 (Ref. 4) relies on a DNB criterion for determining failures in PWRs, whereas a radial average energy density of 170 cal/g is accepted for BWRs under zero and low power conditions. Other limits may be more accurate for an RIA, but continued approval of these limits may be given until generic studies yield improvements.

Although a DNB 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. Although there is at present little experience with other approaches, positions recommending different criteria should address cladding temperature, pressure, time duration, oxidation, and embrittlement.

The second criterion used to assure that the cladding does not overheat is that fuel melting will not occur. There would otherwise be concern that molten fuel might contact the cladding and cause local hotspots. This criterion also avoids the axial relocation of molten fuel that could cause local overheating.

(b) Pellet/Cladding Interaction (PCI): 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 core 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 (a) to ensure that overheating of the cladding would not occur.

(c) Hydriding: Hydriding as a cause of failure (i.e., primary hydriding) is prevented by keeping the level of moisture and other hydrogenous impurities very low during fabrication. Acceptable moisture levels for Zircaloy-clad uranium oxide fuel should be no greater than 20 ppm. Current ASTM specifications (Ref. 5) for UO_2 fuel pellets state an equivalent limit of 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 mg H_2O per cm^3 of hot void volume within the Zircaloy cladding has been shown (Ref. 6) to be insufficient for primary hydride formation.

(d) 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.

(e) Bursting: Zircaloy cladding will burst (rupture) under certain combinations of temperature, heating rate, and differential pressure. Although fuel suppliers may use different rupture-temperature vs differential-pressure curves, an acceptable curve should be similar to the one determined by Oak Ridge National Laboratory (Ref. 7). This criterion is included in the ECCS evaluation model required by Appendix K (Ref. 8).

(f) 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.

(g) 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 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 transients.

3. Fuel Coolability

Coolability 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. Coolability criteria should include the following to be complete:

(a) Cladding Embrittlement: Oxygen contamination and hydriding in Zircaloy cladding are the primary causes of cladding embrittlement. For the LOCA, Appendix K addresses these phenomena with a criterion of 2200°F peak cladding temperature and a criterion of 17% maximum cladding oxidation. (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 1800°F previously described in Reference 9 is no longer needed in CP and OL reviews.) Specific temperature and oxidation criteria have not been derived for other accidents, but should they be needed, Appendix K can be used as guidance.

(b) Violent Expulsion of Fuel: In severe reactivity initiated accidents (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 to produce pressure pulses in the primary system.

Observing the 280 cal/g limit specified by Regulatory Guide 1.77 prevents widespread fragmentation and dispersal of the fuel and avoids generating pressure pulses in the primary system during an RIA. This 280 cal/g limit should be used for PWRs and 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) Structural Deformation: Analytical procedures are discussed in Appendix A, "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces."

(e) Fuel Rod Ballooning: For the LOCA analysis, Appendix K requires that flow blockage resulting from cladding ballooning (swelling) be taken into account in the analysis of core flow distribution. Flow blockage models must be based on applicable data (Refs. 7, 11, and 12) in such a way that (1) the temperature and differential pressure at which the cladding will rupture are properly estimated (see paragraph (e) 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 Analysis Branch for incorporation in the comprehensive ECCS model evaluation to show that the 2200°F 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.

B. Description and Design Drawings

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 i.d. 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 dimension
Fissile enrichment
Equivalent hydraulic diameter
Coolant pressure

The following design drawings 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 PSARs and FSARs by reference.

1. <u>Operating Experience</u>

Operating experience with fuel systems of the same or similar design should be described. 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 needed not be reviewed. Design criteria for fretting wear, oxidation, hydriding, and crud buildup might be addressed in this manner.

2. <u>Prototype Testing</u>

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 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 incompatability 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. 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.

(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. 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 (Refs. 13 and 14). Examples of previous fuel performance code reviews are given in References 15 through 18.

(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 19 and in Regulatory Guide 1.126 (Ref. 20). Models for cladding collapse times must also be reviewed, and previous review examples are given in References 21 and 22.

(c) Fuel Rod Bowing: Guidance for the analysis of fuel rod bowing is given in Reference 23. Interim methods that may be used prior to compliance with this guidance are given in Reference 24. 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. The models are empirical and should be compared with relevant data. Examples of such data and a previous review are contained in References 7, 11, 12, and 26.

(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: The rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction should be calculated using the Baker-Just equation (Ref. 27) as required by Appendix K. For non-LOCA applications, other correlations may be used if justified.

*Although needed in fuel performance codes, this model is reviewed by the Analysis Branch as described in SRP Section 4.4.

(h) Fission Product Inventory: The available radioactive fission product inventory in fuel rods (i.e., the gap inventory) is presently specified by assumptions in Regulatory Guides (Refs. 4, 28-30). These assumptions should be used until improved calculational methods are approved by CPB (see Ref. 31).

D. 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.

1. Testing and Inspection of New Fuel

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 on-site 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. On-line Fuel System Monitoring

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. Reference 32 evaluates 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 33 are acceptable.

3. Post-irradiation Surveillance

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.

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 reviewers with findings issued as NUREG- or WASH-series reports. At the present time these reports include References 9, 19, 31, 32, 34 and 35, and they should all be appropriately cited in the plant safety evaluation report. Applicable Regulatory Guides (Refs. 4, 20, 28-30) and Branch Technical Positions (there are none at present) 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. 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.

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 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.

"The applicant has provided sufficient evidence that these design objectives will be met based on operating experience, prototype testing, and analytical predictions.

"The applicant has described methods of adequately predicting fuel rod failures during postulated accidents so that radioactivity releases are not underestimated.

"The applicant has also 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 post-irradiation surveillance to detect anomalies or confirm that the fuel has performed as expected.

"On the basis of our review of the fuel system design, we conclude that the applicant has met all the requirements of the applicable regulations, current regulatory positions, and good engineering practice."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 10, "Reactor Design."
2. "Rules for Construction of Nuclear Power Plant Components," ASME Boiler and Pressure Vessel Code, Section III, 1977.
3. W. J. O'Donnel and B. F. Langer, "Fatigue Design Basis for Zircaloy Components," Nucl. Sci. Eng. 20, 1 (1964).

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

5. "Standard Specification for Sintered Uranium Dioxide Pellets," ASTM Standard C776-76, Part 45, 1977.

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

7. R. H. Chapman, "Multirod Burst Test Program Quartarly Progress Report for April - June 1977," Oak Ridge National Laboratory Report ORNL/NUREG/TM-135, December 1977.

8. 10 CFR Part 50, Appendix K, "ECCS Evaluation Models."

9. "Technical Report on Densification of Light Water Reactor Fuels," AEC Regulatory Staff Report WASH-1236, November 14, 1972.

10. (Deleted)

11. F. Erbacher, "Single and Multirod Tests, Transient and Steady State, Internal Conduction Heating," Fifth NRC Water Reactor Safety Research Information Meeting, Gaithersburg, Maryland, November 11, 1977.

12. R. H. Chapman, "Some Preliminary Results of Single Rod and Multirod Tests With Internal Heaters," NRC Zircaloy Cladding Review Group Meeting, Silver Spring, Maryland, January 18, 1978.

13. 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.

14. 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.

15. 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.

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

17. "Supplement 1 to the Technical Report on Densification of General Electric Reactor Fuels," AEC Regulatory Staff Report, December 14, 1973.

18. "Technical Report on Densification of Exxon Nuclear PWR Fuels," AEC Regulatory Staff Report, February 27, 1975.

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

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

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

22. Memorandum from D. F. [illegible]ss, NRC, to R. C. DeYoung, Subject: CEPAN -- Method of Analyzing Creep C[illegible]e of Oval Cladding, dated February 5, 1976.

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

24. 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.

25. (Moved to Appendix A)

26. Letter from D. F. Ross, NRC, to A.E. Scherer, Combustion Engineering, dated March 22, 1978.

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.

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

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

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."

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

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

33. "Safety Evaluation Report Related to Operation of Arkansas Nuclear One, Unit 2," USNRC Report NUREG-0308, Supp. 2 (to be issued).

34. B. L. Siegel, "Evaluation of the Behavior of Waterlogged Fuel Rod Failures in LWRs," USNRC Report NUREG-0303, March 1978.

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

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

PROPOSED ADDITION OF

APPENDIX A

EVALUATION OF FUEL ASSEMBLY STRUCTURAL RESPONSE

TO EXTERNALLY APPLIED FORCES

TO

STANDARD REVIEW PLAN PSRP-4.2, REVISION 2, DRAFT 1

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 during these low probability accidents. This Appendix describes the review that should performed of the fuel assembly structural response to seismic and LOCA loads. Background material for this Appendix is given in Refs. 1-3.

B. ANALYSIS OF LOADS

1. Input

Input for the fuel assembly structural analysis comes from results of the primary coolant system 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 reactor primary coolant system 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 (see Regulatory Guide 1.60).

This proposed revision of the Standard Review Plan and the support value/impact statement have not received a complete staff review and approval and do not represent an official NRC staff position. Public comments are being solicited on both the revision and the value/impact statement (including any implementation schedules) prior to a review by the Regulatory Requirements Review Committee and their recommendation as to whether this revision should be approved. Comments should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch. All comments received by ____________________ will be considered by the Regulatory Requirements Review Committee. A summary of the meeting of the Committee at which this revision is considered, the Committee recommendations and all of the associated documents and comments considered by the Committee will be made publicly available prior to a decision by the Director, Office of Nuclear Reactor Regulation, on whether to implement this revision.

2. Methods

Analytical methods used in performing structural response analyses must 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) must also be reviewed. Experimental verification of the analytical representation of the fuel assembly components should be provided when pratical.

A sample problem of a simplified nature must be worked by the applicant and compared by the reviewer with either hand calculations or results generated by the reviewer with an independent code (2). 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 $\pm$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 $\pm$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.

4. Audit

Independent audit calculations for a typical full-sized core must be performed by the reviewer to verify that the overall structural representation is adequate. An independent audit code (2) should be used for this audit during the generic review of the analytical methods.

5. Combination of Loads

General Design Criterion 2 requires an appropriate combination of loads from natural phenomena and accident conditions. 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. DETERMINATION OF STRENGTH

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 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. ACCEPTANCE CRITERIA

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 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 for the LOCAs that require insertion to assure subcriticality. Loads from the most severe 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 rod insertability: (a) combined loads on the channel box must remain below the allowable value defined above for components other than grids because a small amount of channel deformation could interfere with control blade insertion, and (b) vertical liftoff forces must not unseat the lower tieplate from the fuel support piece because the resulting loss of lateral fuel bundle positioning could also 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.

E. REFERENCES

1. 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.

2. 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.

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

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Value-Impact Statement on
PROPOSED ADDITION OF
APPENDIX A
TO
STANDARD REVIEW PLAN PSRP 4.2, REVISION 2, DRAFT 1

I. PROPOSED ACTION

This appendix provides (a) guidance for the analysis of fuel assembly loads, (b) guidance for the determination of strength of fuel assembly components, and (c) acceptance criteria for the fuel assembly structural response to externally applied forces such as arise in loss-of-coolant accidents and earthquakes.

II. BACKGROUND

No systematic guidance or acceptance criteria exist for this analysis. The North Anna and Diablo Canyon methods are presently being used as precedents, but there are many problems with this procedure. For example, (a) those methods accommodate the Westinghouse analysis and fuel design, but they are difficult to apply to other vendors, (b) those methods focus undue attention on spacer grids, (c) non-standard analyses, which are inconvenient to require, were done for North Anna and Diablo Canyon, and (d) some of the conservatisms accepted for those plants are unwarranted. This guidance plays an important role in resolving two Unresolved Safety Issues (A-2 and B-6).

III. VALUE ASSESSMENT

This appendix will (a) provide fixed criteria and suitable methods for all designs and vendors, (b) allow the reduction of unnecessary margin in areas that are now preventing (and would increasingly prevent) OL approval of the fuel for certain plants, and (c) result in a reduction in time spent reviewing relatively unimportant analyses. The appendix is a clear statement of review requirements and will result in an overall improvement in the quality of this review.

IV. IMPACT ASSESSMENT

A. NRC

There will be an additional expense for the evaluation of the grid strength test rigs. This will involve consulting services from a testing laboratory, and the cost should not exceed $10,000. Any additional review effort created by the more comprehensive nature of this guidance should be offset by savings in areas now thought to be less important.

B. Industry

There will be a small number of additional code runs required during the generic review of vendor methods and there will be a small increase in required laboratory testing of spacer grids. The potential exists that some vendors may have to reconstruct their test rigs if our evaluation shows them to be inadequate.

From preliminary information it appears that all current fuel designs will meet the new criteria. Therefore, the impact will be essentially confined to the first-time generic demonstration of compliance, and little or no plant-specific impact is expected.

V. DECISION ON THE TECHNICAL APPROACH

Many of the considerations in this appendix were carried over from past review policies and do not need further discussion. Several new considerations were questioned and warrant highlighting.

A differing opinion was expressed by an NRC reviewer during the development of procedures in this appendix (see Attachment A). That reviewer recommended that a best-estimate analysis of loads be compared with a 95x95 lower tolerance limit (LTL) on measured grid strength and that the comparison show a safety factor (conservative margin) of at least 1.35. He offered comments on several other positions in the proposed appendix. We have chosen a different approach for the following reasons.

1. Best-estimate codes are usually not available for this analysis. The fuel assembly and reactor coolant system structural codes that have been submitted to NRC for review are designed to be inherently conservative. The input values that we approve are usually also conservative.

2. In the precedent-setting North Anna case, where we attempted to use best-estimate analytical methods, the conservative methods, which had been reviewed, were modified in a non-rigorous way to eliminate some of the conservatisms. We believe it is better to keep inherent conservatisms that have been reviewed than to remove them and substitute an arbitrary safety factor. For North Anna, 1.35 times the best-estimate value is only 10% larger than the conservative value without a safety factor.

3. There is no assumption in our work that calculational methods are perfect. To the contrary, we recognize the difficulty in modeling non-linear phenomena and find it more reliable to allow simplifying assumptions that are patently conservative rather than striving for a best-estimate prediction with error bounds. The latter is probably not achievable with the present state of the art.

4. The proposed safety factors for steam flashing and pronounced sensitivity are conserative margins that the minority position would eliminate. These are recognized sources of potential error that we believe should be accounted for.

5. We retain the use of 95x95 LTL values (or similar ASME code values) for all components except spacer grids because (a) moderately deformed grids (even severely deformed grids) appear incapable of producing significant consequences and (b) rigid spacer grids might damage fuel rods, control rods and guide tubes. Because they are not acting as supporting structural members, using a mean yield strength value does not mean that 50% of the grids will "fail;" it means that 50% of the grids may be bent and 50% will not. Our work over the past few years has shown spacer grids to be susceptable to deformation, but it has not shown any serious consequences of spacer grid deformation. We believe that excessive attention to grid behavior has diverted attention from the more important components.

6. A benchmark test with standard grids is desirable because grid impact testing is not a standard procedure and large variations may exist from one test rig to another. One vendor found an apparent increase in grid strength on the order of 20% when he changed the hinge arrangement on his impact pendulum. Some vendors swing pendulums while others drop weights or test statically. Our efforts to provide uniform conservatism in the analytical methods would be seriously undercut without some means of knowing that all vendors were measuring approximately the same strength property.

VI. IMPLEMENTATION

The methods and criteria in this appendix update procedures that have been included in plant safety analyses for several years. Implementation of the LOCA portion of these new procedures for operating reactors and recent operating license actions will occur as part of the resolution of Unresolved Safety Issue A-2. For operating reactors, licensees were notified in January 1978 by a letter from Victor Stello to supply analyses conforming to forthcoming guidance. A similar letter went to license applicants from Roger Boyd in November 1978. The forthcoming guidance, which includes the LOCA portions of this appendix in their entirety, is contained in the generic report on the resolution of Task A-2 ("Asymmetric Blowdown Loads on PWR Primary Systems," NUREG-0609). Completed analyses for both of these categories of plants are expected in January 1980. For all other new plants, these new procedures will be implemented routinely in the review process.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAY 2 9 1979

MEMORANDUM FOR: R. O. Meyer, Section Leader, Reactor Fuels Section, CPB, DSS

FROM: S. B. Kim, Reactor Fuels Section, CPB, DSS

SUBJECT: COMMENTS ON THE PROPOSED APPENDIX TO THE STANDARD REVIEW PLAN "FUEL ASSEMBLY DESIGN ACCEPTANCE CRITERIA FOR DYNAMIC LOADS"

Enclosed is my comments on the subject acceptance criteria.

Sang B. Kim
Reactor Fuels Section
Core Performance Branch
Division of Systems Safety

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ENCLOSURE

In my opinion, it is too early to eliminate existing safety factor in the current practice and propose a new acceptance criteria in SRP Appendix. We only reviewed the Westinghouse method so far and we do not know the details of other vendor methods (CE, Exxon Nuclear and B&W). We may be in a position to write a formal acceptance criteria only after we complete review of all the vendors and, hopefully, some full scale test become available. We need this learning curve because the method contains highly complex non-linear analysis which is new to us.

I recommend to continue to use an interim safety factor of 1.35 (instead of 1.75) on grid strength as an engineering safeguard until we reach a point of better understanding.

Following are the reasons for the safety factor. I also made several technical comments on the proposed new criteria as a backup data as to why such criteria is premature and technically unsound.

A. Comments on the Proposed Acceptance Criteria

1) EG&G's proposal assumes that one can devise a calculational method for a grid impact with no error bound (perfect model). To state that non-linear analysis predict best-estimate value without benefit of experimental verification seems to be an unsupportable statement and may turn out to be on an undefendable position. (At one time, EG&G recommended a full scale experiment including reactor and internals because grid impact may be difficult to evaluate by calculation alone.

2) Two of the major items proposed by the EG&G have technical deficiencies.

   a) Steam flashing: safety factor 1.3

      This is a new load proviously not considered by the vendors. Assigning a safety factor at this time is premature since the Analysis Branch with the primary responsibility in the area has not commented on it. This new loading is a typical example of a moving target.

   b) Sensitivity: safety factor

      EG&G proposes that "plant specific sensitivity should be performed and additional safety factor should be imposed on the grid force if the sensitivity calculation shows that such safety is needed." Reason for the above proposal was based on the EG&G sensitivity calculation where acceleration of core plate was used as a variable parameter rather than displacement function. However, when a displacement time history was used, as done by all the vendors, no such sensitivity was apparent. Therefore, we need a better justification before requiring vendors any plant-specific sensitivity calculations.

3) Mean value vs. 95x95 LTL

   It was noted in the new proposal that the grid strength may be determined by selecting a mean value of the test data rather than some upper bound. We may be breaking a new ground in that allowable stress of safety component is selected by a mean value (half of the component is allowed to fail?). I believe that not even a secondary system component allowable was determined by a mean value. It was argued that the mean value is allowed because consequence of failure is small. However this argument was used several times explicitly or implicitly in developing the new criteria. If so, we might as well declare that the spacer grid is not a safety component, and drop from the SRP item.

4) It is stated in section 3.a that we provide standard grids to all the vendors in an audit for the grid strength determination. This approach is premature and may create a problem. We do not know, at present, what is right procedure to determine a dynamic strength capability. Therefore, when we have a scatter in grid strength from different vendors, as we suspect, there is no way of deciding which value is correct. It is best not to do it now and accommodate such uncertainty by a safety factor. We will have better ideas on grid strength determination once we review all the vendor methods.

B. Recommendation

I recommend to use safety factor 1.35 as an interim measure for the spacer grid. The numerical value of the safety factor was obtained from the following:

1) Component (spacer grid) safety factor: 1.[illegible]

ASME (as well as any other engineering practice such as ASCE) recommends minium of 1.1 in safety factor for a service level D (faulted). With a possible exception of CE, all the vendors follows ASME recommendation. It is simply a bad engineering practice not to assign any safety factor for the components especially for a nuclear plant application.

2) Allowance for uncertainty in analysis: 0.25

The magnitude of the uncertainty factor in grid load calculation is somewhat arbitrary. However it provides the following advantages;

i) encourage vendors to provide rigorous verification program.
ii) keep review simple (no plant sensitivity analysis nor detailed mechanistic review)
iii) flexible to adapt new situation such as accommodating moving target.

I also recommend to retain 95x95 lower tolerance limit for the reason stated previously.