NUREG-75/087

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 4.2 FUEL SYSTEM DESIGN

REVIEW RESPONSIBILITIES

Primary - Core Performance Branch (CPB)

Secondary - Mechanical Engineering Branch (MEB)
Quality Assurance Branch (QAB)

I. AREAS OF REVIEW

The mechanical, thermal, and chemical design of the fuel assembly is evaluated by CPB. The fuel assembly is generally a square array of fuel rods (varying from 36 rods to 264 rods) which are mechanically secured together. The fuel rods are laterally supported by grid subassemblies at intervals along their length to maintain the assembly geometry. Some fuel assemblies allow control rods to be inserted within the square array. Those parts of the control rods which are inserted into the core although not considered as part of the assembly will be evaluated under this section. The fuel assembly is considered to include fuel pellets, burnable poisons, fill gas, getters, cladding, springs, end closures, spacer grids and springs, end fittings, guide thimbles, and channel boxes.

The review considers specific aspects of fuel behavior which affect and limit the safe and reliable operation of the plant. Steady state, anticipated reactor transient, and design basis accident conditions, including loss-of-coolant accidents (LOCA), are evaluated for both initial and reload cores. The specific aspects of interest are listed below:

1. The cladding mechanical property limits are reviewed. Mechanical properties include Young's modulus, Poisson's ratio, design dimensions, and allowable tolerances on wall thickness, diameters, and ovality as well as material strength and ductility properties. Yield and ultimate strength, uniform and total ductility, and creep rupture limits must reflect the effects of temperature and neutron fluence on these properties. Dimensional changes due to temperature, pressure, and neutron effects are reviewed.

2. The design against fatigue failure from either flow-induced vibration or power cycling is reviewed for spacer grids, fuel rods, springs, guide thimbles, and flow channel boxes. The consideration of stress levels, amplitudes of vibration, and life fraction are included. The form of the design criteria used may be curves of strain or stress amplitude versus the number of cycles.

3. The review includes an evaluation of the predicted time to cladding creep collapse into a fuel stack axial gap or the gas plenum. The appropriate design creep rate for predicting radial fuel-clad gap closure is considered separately from that for creep collapse because the safety implications differ.

4. The analytical model for fuel densification is evaluated, including both the extent and the kinetics of densification during operation. The effects of densification, which may cause changes in the stored energy, linear thermal output, axial gap, and thermal impedance are evaluated in the fuel design review (Ref. 2).

5. The fuel system is reviewed for maximum permissible power density to assure the appropriate margin between anticipated duty and the power density at which fuel rod failure would be expected. The permissible power densities should include local peaking as affected by anticipated transients.

6. The total internal pressure in the fuel rod is evaluated to assure the adequacy of the gas plenum design against rod burst. Additionally, the internal pressure calculations are reviewed for the effects of internal pressure on predictions of flow blockage during transients and accidents.

7. The potential for adverse chemical interactions either among the fuel assembly components or between a fuel component and the reactor environment is evaluated. The potential for adverse chemical interactions among the control rod subassembly components must be evaluated.

8. The fuel system design and the control rod subassembly design are evaluated for the physically feasible combinations of chemical, thermal, mechanical, and hydraulic interaction. Evaluation of these interactions includes the effects of normal reactor operation, anticipated transients, and postulated design accidents. Examples of possible interactions are: fuel-cladding mechanical interaction, fuel fission product-cladding attack, stress-accelerated corrosion, fretting corrosion, fuel rod burn-out, crevice corrosion, crud deposition, material wastage due to mass transfer, axial thermal expansion of fuel against collapsed cladding, and thermal and creep-induced dimensional changes.

9. The fuel system design is reviewed to assure that the appropriate physical and thermal properties for the materials used are being employed. These properties include thermal expansion (may be direction dependent), thermal conductivity, thermal diffusivity, specific heat, specific gravity, and temperatures of phase changes.

10. The potential for subassembly flow blockage arising from either external or internal causes is reviewed.

11. The review includes the effects of shock loadings (including LOCA) on both the fuel assembly geometry and fuel rod integrity. The effects of combined shock and seismic loads are analyzed.

12. The completeness of the applicant's design analysis is reviewed to assure that all criteria and the appropriate margins have been considered. The analysis is reviewed to assure that some surveillance of actual performance is included as a verification of the design.

13. The applicant's proposed technical specifications related to areas covered in this plan are reviewed for operating license (OL) cases.

The primary review responsibility rests with the Core Performance Branch. Other branches provide assistance as requested by CPB. The QAB provides consultation on matters concerning the representative nature of test results and the characterization of the component materials. The MEB provides consultation both on the interaction of the fuel assembly with adjacent core components and on the applied mechanics used in design. In addition, the Advanced Program Development Branch of the Office of Inspection and Enforcement may be consulted by CPB on fuel vendor practices and reactor performance of specific design features.

II. ACCEPTANCE CRITERIA

The general purpose of the review is to establish that all safety-related aspects of the fuel system design have been adequately considered and that the proposed fuel design limits have appropriate margin and are acceptable, as required by General Design Criterion 10 (Ref. 1). The specific criteria for the fuel system design are listed below:

1. The fuel cladding mechanical properties used in the design should be consistent with generally accepted values and characteristics of the material.

2. The general membrane stress limits for the cladding must be reasonably less than the corresponding material strengths for the design service temperatures and neutron fluences. The procedure for calculation of the maximum cladding strain fatigue should be one approved by the staff.

3. The cumulative number of strain fatigue cycles should be significantly less than the design fatigue life of the particular material. For example, design allowances may be based on appropriate data which has been modified by a factor of 2 on stress amplitude or of 20 on the number of cycles (Ref. 3).

4. The predicted time to cladding creep collapse should be compatible with the allowable peak cladding temperature (PCT) for LOCA analysis. When no zircalloy cladding collapse is expected, the calculated PCT should be less the 2200°F. For reactor service beyond the predicted time to collapse, the calculated PCT should be less than 1800°F. The analytical model used for the prediction should be one approved by the staff. Staff approval of a model will be based in part on a comparsion with results from a staff creep collapse code, e.g., BUCKLE (Ref. 4), or COVE (Ref. 5).

5. The analytical thermal performance model for the fuel should be one approved by the staff and should include the effects of fuel densification, fission gas release, and

burnable poisons. The size and probability of fuel column axial gaps should be predicted by an approved method. To be approved, the analytical model should be capable of predicting appropriate test data and be corroborated by a staff thermal performance code, e.g., GAPCON (Ref. 6). The results of calculations with the model should show compliance with design limits such as fuel temperatures and maximum stored energy.

6. The maximum power density in the fuel should be less than the value at which fuel rod failure is predicted. A margin should be included that allows for calculational uncertainty, experimental error, and operational transients.

7. The calculated differential pressure across the fuel rods cladding during normal in-reactor service should be less than the pressure at which cladding failure would be expected.

8. The calculations for waterlogged rods during anticipated transients should include the hydrostatic pressure contribution of the contained water. Two elements to be considered in the analysis are: (a) the amount of water available inside the cladding and (b) the rate of change of temperature during a transient. The amount of water in a waterlogged rod may be determined either by inspection and test data or by a bounding calculation which determines the amount of water to equalize the system and internal pressures. The appropriate rate of change of temperature for rupture may be determined by test data, e.g., from the SPERT tests (Ref. 7).

   The flow blockage associated with rupture from internal pressure should be consistent with appropriate test data.

9. The potential adverse chemical interactions should be considered on the basis of satisfactory operating experience of similar designs and other appropriate date.

10. Fuel system thermal-mechanical interactions may be evaluated by fuel behavior codes such as LIFE-II (Ref. 8), CYGRO (Ref. 9), or FRAP (Ref. 10). The results of analyses of fuel-clad mechanical interaction should compare favorably with correlations of data relating fuel performance and power density conditions that would occur during normal and transient operations. The design provisions for prevention of excessive fretting should be shown to be adequate by data from design verification or proof tests.

11. The mechanical aspects of flow blockage should be determined by examination of appropriate data. Analysis of the thermal aspects of flow blockage should be done by methods approved by the staff as a part of the reactor thermal-hydraulic design review, or previously approved in case or generic reviews.

12. Design methods for predicting creep deformation and plasticity should be verified against appropriate test data approved by the staff. Values of creep deformation to be used for design purposes generally require some margin from predicted values, to account for the scatter inherent in creep data. The magnitude and direction of the margin depends upon both the extent of scatter of the data and the design application.

When a creep prediction is used in gap conductance calculations, it is acceptable if it underpredicts a significant fraction of the appropriate test data. When a creep prediction is used for cladding collapse calculations, it is acceptable if it overpredicts measured deformations.

13. Calculations of the effects of shock loadings on the fuel, including those from LOCA, should be based upon established methods and codes. The methods may be either time history, shock spectrum, or statistical, and should include any environmental degradation effects in the material.

14. The completeness of the design analyses should be demonstrated by a listing of all design criteria, the corresponding design values, and the "best engineering estimates" for normal operating values. The criteria, design values, and "best estimates" may be in the form of stresses, strains, times, or cycles. Results from or plans for a surveillance program should provide a reasonable means of verifying the actual fuel performance.

15. The design must assure that reactivity control materials remain below their melting point and that it provides a means of accommodating or venting gaseous fission products.

III. REVIEW PROCEDURES

The reviewer should assure that the intent of each of the acceptance criteria of Section II has been complied with fully. The assurance is provided by a systematic evaluation of the design against each criterion above. The various aspects of the design may be considered adequate based upon corroborating computer code calculations, confirmatory hand calculations, generally accepted engineering conventions and industry standards, comparisons with appropriate data, or results of operating experience. A list of commonly used codes, standards, and specifications is given in Table 4.2-1, for information only. The reviewer should assure that:

1. The data base used for the fuel system design is applicable to the particular design.

2. The design parameters pertinent to safety have been appropriately considered in relation to each particular design aspect.

3. The expected variance in parameter values has been accommodated.

Most of the detailed safety review of fuel systems designs is accomplished on a continuing generic basis outside the docketed applications. Thus, there are no unique review procedures for the evaluation of a fuel system design. The CPB deals directly with fuel vendors and evaluates the engineering methods employed in each aspect of a fuel system design (Refs. 11-17). Consequently, much of the review procedure in evaluating a specific plant is directed to assuring that the design methods used have been approved by the staff and are being correctly applied.

The full scope of a fuel system design safety evaluation at the operating license (OL) stage is covered in this plan. The safety evaluation at the construction permit (CP) stage need not be specific in all aspects of the fuel system design. Those aspects that may change between the CP and OL stages need not be addressed, e.g., degree of fuel densification, cladding mechanical properties, level of prepressurization, and as-manufactured dimensions. The primary interests at the CP stage are:

1. The completeness of the design analysis should be assured, such that all the design criteria have been or will be addressed.

2. The engineering methods being employed in the design should be either already approved by the staff or review of the methods should have progressed sufficiently that approval may be reasonably anticipated prior to the OL stage.

3. The proposed fuel system design should be consistent with that of previously approved plants of the same type.

Review of the technical specifications related to the fuel system is carried out as part of the review for operating license applications. Appropriate technical specifications for limiting power density values are developed in the review. Various aspects of fuel systems components that must be considered in the design and evaluated by the reviewer in the OL review are listed below.

1. For the fuel cladding, the design must consider dimensions, composition, thermal-mechanical processing, and the optimum strength and ductility capabilities of the tubing for the expected duty. The cladding design should be such as to accommodate the fission gas evolved in operation, so that the fuel can reach design burnup without exceeding the cladding structural design criteria. These design aspects require adequate plenum volume and cladding thickness, including allowances for surface defects and manufacturing tolerances. Cladding design requires calculations of mechanical limits, e.g., by computer codes such as BUCKLE (Ref. 4) and COVE (Ref. 5) and of the effects of operation on cladding geometry, e.g., by computer codes such as FRAP (Ref. 10) and CYGRO (Ref. 9). Computer models which have been indexed to appropriate test data may be used by the reviewer, e.g., the computer codes GAPCON (Ref. 6) and LIFE-II (Ref. 8).

2. For the fuel assembly, the design must consider rigidity during shock loading, hydraulic loading, and transportation loadings both before and after reactor service. The potential dimensional changes of components resulting from thermal, chemical, mechanical, and irradiation-induced degradation; loads applied by the core restraint system; and loads applied during grappling (in fuel handling), including those from misaligned handling tools, are considered in the fuel assembly performance review. The assembly end fittings must properly mate with the assembly positioning system and preclude misorientation or mislocation of the assembly.

3. The spacer grid design must consider spring loads, dimensional tolerances, materials, and joining methods. The design must consider axial and radial growth due to temperatures, burn-up, and neutron irradiation. The spacer must prevent radial oscillations, allow adequate cooling by maintaining the specified pitch to diameter ratio, and remain chemically compatible with the fuel cladding material.

4. For oxide fuel, the design must consider the size, shape, density, and composition of the pellets. The size considerations include the effects of temperature, density, and thermal performance margins. The shape is dependent upon design exposures, anticipated methods of operation, and cladding characteristics. Design densities are affected by thermal performance, fuel rod lengths, manufacturing variations, and pellet shape. The pellet composition requires consideration of nuclear and thermal performance requirements and compatibility with other fuel rod components during the anticipated service, including the effects of burnable neutron poisons. The complexity and interaction of all these components necessitates the use of sophisticated analytical computer models.

5. For springs, the design must consider the dimensions required for the requisite positioning of components. Spring dimensional considerations include allowances for thermal and irradiation-enhanced stress relaxation. Of particular importance is the cumulative effect of a series of springs.

IV. EVALUATION FINDINGS

The general scope of the review of the fuel system design is the same at the CP stage as at the OL stage. However, the review for an OL is more detailed and specific than for a CP. At the OL stage, the generically approved design methods and codes, the detailed materials properties, and the appropriate reactor environmental conditions for the as-fabricated fuel system are utilized to establish that the particular plant has met the design criteria. The staff's safety evaluation report (SER) should reflect this difference. The following typical evaluation findings are given for OL and CP reviews:

1. Operating License

"The fuel system for the ___________ plant includes the fuel assembly, which is composed of ___________ fueled rods, ___________ nonfueled tubes, ___________ spacer grids, and end-fittings. The fuel rod includes fuel pellets, plenum springs, cladding, end closures, and thermal and chemical buffers. The basic mechanical function of the fuel system is to provide a controlled core geometry during normal operations, anticipated transients, and accidents. The review has considered the specific aspects of fuel system behavior which affect and can limit the safe, reliable operation of the plant.

"The evaluation of the fuel system mechanical design was based upon mechanical tests, in-reactor operating experience, and engineering analyses. Additionally, the in-reactor performance of the fuel system design will be subject to the continuing surveillance programs of the fuel system vendor, the staff, and the applicant. These programs provide confirmatory performance information. In reviewing the engineering analyses, the applicability of the design criteria and the rigor of the applied methods were evaluated to confirm compliance with design objectives.

"Part of the basis for acceptance in the staff review has resulted from a systematic evaluation of the fuel system design with regard to the design criteria. The engineering analyses are considered adequate based upon corroborating computer code calculations using staff-approved methods, confirmatory hand calculations, generally accepted engineering conventions and standards, and comparisons with appropriate test data.

"Further bases for acceptance are favorable results of out-of-reactor mechanical tests and in-reactor performance of directly comparable fuel systems. The staff has reviewed the tests and found the quality of the reactor simulation adequate for that aspect of the fuel system being examined. Fuel systems of similar design have been successfully irradiated for up to _______ years and have had peak exposures of _______ MWD/MT.

"The staff concludes that, based upon operating experience with similar fuel systems, results of out-of-reactor tests, technical specification requirements to monitor and limit off-gas and effluent activity, and the continuance of a fuel rod surveillance program including destructive and non-destructive post-irradiation examinations, the integrity of the fuel system will be maintained during both normal operations and incidents of moderate frequency, and the proposed fuel design limits are adequate and acceptable, as required by General Design Criterion 10. Further, we conclude that accidents or earthquake-induced loads will not result in an inability to cool the fuel _______ or significant interference with control rod insertion."

2. <u>Construction Permit</u>
(Some paraphrasing of the first paragraph for an OL).

"The analytical models employed by the applicants have been shown to be acceptable by comparison with measurements on fuel rods which have been subjected to reactor operating conditions. These models, described in topical reports, are based on data for fuel similar to that proposed for use in ___________. These analytical models, which have been reviewed in detail by the staff, provide acceptable assessments of the anticipated fuel rod behavior.

"On the basis of our review of the proposed analytical models and the confirmatory results from tests on irradiated fuel rods, we have concluded that, (1) the fuel rod mechanical design will provide acceptable engineering safety margins for normal operation, and (2) the effects of densification will be acceptably accounted for in the fuel design."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 10, "Reactor Design."

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

3. W. J. O'Donnel and B. F. Langer, Nuclear Science and Engineering, Vol 20, 1-12 (1964).

4. P. J. Pankaskie, "BUCKLE, An Analytical Computer Code for Calculating Creep Buckling of an Initially Oval Tube," BNWL-B-1974, Battelle Northwest Laboratories, May 1974.

5. C. Mohr, "COVE, A Finite Difference Creep Collapse Code for Oval Fuel Pin Cladding Material," BNWL-1896, Battelle Northwest Laboratories, March 1975.

6. C. R. Hann, C. E. Beyer, and L. J. Prachen, "GAPCON Thermal-1: A Computer Program for Calculating the Gap Conductance in Oxide Fuel Pins," BNWL-1778, Battelle Northwest Laboratories, September 1973.

7. L. A. Stephan, "The Effects of Cladding Material and Heat Treatment on the Response of Waterlogged UO_2 Fuel Rods to Power Bursts," IN-ITR-III, Idaho Nuclear Corporation, January 1970.

8. V. Z. Jankus and R. W. Weeks, "LIFE-II - A Computer Analysis of Fast Reactor Fuel Element Behavior As a Function of Reactor Operating History," Nuclear Engineering and Design, Vol. 18, No. 1, January 1972.

9. E. Duncombe, C. M. Friedrich, and J. K. Fischer, "CRYGRO 3 - A Computer Program to Determine Temperature, Stress, and Deformations in Oxide Fuel Rods," WAPD-TM-961, Bettis Atomic Power Laboratory, March 1970.

10. J. A. Dearien and L. J. Siefken, "FRAP-T: A Computer Program for Calculating the Transient Response of Oxide Fuel Rods," I-243-3-51.1, Aerojet Nuclear Company, (to be published).

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

12. Safety Evaluation by the Directorate of Licensing, U.S. Atomic Energy Commission, in the matter of Baltimore Gas and Electric Company's Calvert Cliffs Nuclear Power Plant Units 1 & 2, Dockets 50-317/318, Supplements 1 and 2, October 1973.

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

14. Regulatory Staff, "Technical Report on Densification of General Electric Reactor Fuels," U.S. Atomic Energy Commission, August 23, 1973, and Supplement 1, December 14, 1973.

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

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

17. Regulatory Staff, "Technical Report on Densification of Combustion Engineering Reactor Fuels," U.S. Atomic Energy Commission, August 19, 1974.

TABLE 4.2-1

REFERENCE CODES, STANDARDS, AND SPECIFICATIONS

CODE, STANDARD, OR SPECIFICATION		TITLE
ASME		Boiler and Pressure Vessel Code, Section III Nuclear Power Plant Components
ASTM	E-8	Tension Testing of Metallic Materials
ASTM	E-21	Short Time Elevated Temperature Tension Testing of Materials.
ASTM	E-112	Estimating Average Grain Size of Metals
ASTM	G-2	Aqueous Corrosion Testing of Samples of Zirconium and Zirconium Alloys
ASTM	E-29	Indicating Which Place of Figures are to be Considered Significant in Specified Limiting Values
MIL-STD-105D		Sampling Procedures and Tables for Inspection by Attributes
ASTM	A-370	Mechanical Testing of Steel Products
ASTM	A-393	Recommended Practice for Conducting Acidified Copper Sulfate Test for Intergranular Attack in Austenitic Stainless Steels
ASTM	A-262	Recommended Practice for Detecting Susceptibility to Intergranular Attack in Stainless Steel
ASTM	E-94	Recommended Practice for Radiographic Testing
RDT	M3-28T	Austenitic Stainless Steel Tubing for LMFBR Core Components
RDT	M1-16T	Zirconium and Zirconium Alloy Bare Welding Rods
RDT	M2-9T	Zirconium and Zirconium Alloy Forgings and Extrusions
RDT	M5-6T	Zirconium and Zirconium Alloy Plate, Sheet, and Strip
RDT	M7-9T	Zirconium and Zirconium Alloy Bars, Rod, and Wire
RDT	M-10-1T	Zirconium and Zirconium Alloy Ingots
Bureau of Mines		Helium Grade A Specification

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