NUREG-75/087

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 3.9.5 REACTOR PRESSURE VESSEL INTERNALS

REVIEW RESPONSIBILITIES

Primary - Mechanical Engineering Branch (MEB)

Secondary - Core Performance Branch (CPB)
Materials Engineering Branch (MTEB)

I. AREAS OF REVIEW

For the purpose of this standard review plan, the term "reactor internals" refers to all structural and mechanical elements inside the reactor pressure vessel with the exception of the following:

Reactor core (fuel), including the reactivity control elements out to the coupling interfaces with the drive units, as well as the drive elements inside the guide tubes (guide tubes are considered to be a part of reactor internals) and inside the control rod drive mechanism assemblies (drive elements are covered in Standard Review Plan 3.9.4).

In-core instrumentation (in-core instrumentation support structures are considered part of the reactor internals).

The staff review includes the following specific areas:

1. The physical or design arrangements of all reactor internals structures, components, assemblies, and systems should be presented, including the manner of positioning and securing such items within the reactor pressure vessel, the manner of providing for axial and lateral retention and support of the internals assemblies and components, and the manner of accommodating dimensional changes due to thermal and other effects.

2. The design loading conditions that provide the basis for the design of the reactor internals to sustain normal operation, anticipated operational occurrences, postulated accidents, and seismic events should be specified. All combinations of design loadings should be listed (e.g., operating pressure differences and thermal effects, seismic loads, and transient pressure loads associated with postulated loss-of-coolant accidents) that are accounted for in design of the core support structure.

USNRC STANDARD REVIEW PLAN

Standard review plans are prepared for the guidance of the Office of Nuclear Reactor Regulation staff responsible for the review of applications to construct and operate nuclear power plants. These documents are made available to the public as part of the Commission's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Standard review plans are not substitutes for regulatory guides or the Commission's regulations and compliance with them is not required. The standard review plan sections are keyed to Revision 2 of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. Not all sections of the Standard Format have a corresponding review plan.

Published standard review plans will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience.

Comments and suggestions for improvement will be considered and should be sent to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

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3. Each combination cf design loadings should be categorized with respect to the "normal," "upset," "emergency," or "faulted" condition (defined in the ASME Code, Reference 5) and the associated design stress intensity or deformation limits should be stipulated. Design loadings should include safe shutdown earthquake (SSE) and operating basis earthquake (OBE) loads if applicable.

4. The design bases for the mechanical design of the reactor vessel internals should be presented including limits such as maximum allowable stresses; deflection, cycling, and fatigue limits; and core mechanical and thermal restraints (positioning and holddown). Details of dynamic analyses, input forcing functions, and response loadings are discussed in Standard Review Plan (SRP) 3.9.2.

II. ACCEPTANCE CRITERIA

A discussion of loading combinations applicable to reactor internals is presented in SRP 3.9.3 (Ref. 7).

The design and construction of the core support structures should conform to the requirements of Subsection NG, "Core Support Structures," of the ASME Code (Ref. 5).

The design criteria, loading conditions, and analyses that provide the basis for the design of reactor internals other than the core support structures should be consistent with the same requirements as listed above for core support structures.

Deformation limits for reactor internals should be established by the applicant and presented in his safety analysis report. The basis for these limits should be included. The stresses associated with these displacements should not exceed the specified design limits. The requirements for dynamic analysis of these components are discussed in SRP 3.9.2.

III. REVIEW PROCEDURES

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

The configuration and general arrangement of all mechanical and structural internal elements covered by this plan are reviewed and compared to those of previously licensed similar plants. Any significant changes in design are noted and the applicant is asked to verify that these changes do not affect the flow-induced vibration test results required by SRP 3.9.2.

With respect to the design and analysis of these components, a statement by the applicant that they are designed in accordance with Subsection NG, "Core Support Structures," of Reference 5 is acceptable. In lieu of such a commitment, the reviewer must determine that the design and analysis of these components are consistent with the requirements discussed in II, above. This is accomplished by requiring that the applicant describe the design procedures and criteria used in the design of these components. This includes a list of the design limits used for all of the applicable loading conditions.

The deformation limits specified for these components are reviewed to verify that the applicant has stated that these deflections will not interfere with the functioning of related components, e.g., control rods and standby cooling systems, and that the stresses associated with these displacements are less than the design limits for the core support structures.

At the operating license stage, the calculated stresses and deformations are reviewed to determine that they do not exceed the specified design limits.

Any deviations that have not been adequately justified are identified and findings to that effect are transmitted to the applicant with a request for conformance with the requirements discussed in II above or additional technical justification.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided in accordance with this review plan, and that his evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

"The design procedures and criteria that the applicant has used for the reactor internals are in conformance with established technical procedures, positions, standards, and criteria which are acceptable to the staff.

"The specified design transients, design loadings, and combination of loadings as applied to the design of the reactor internals structures and components provide reasonable assurance that in the event of an earthquake or of a system upset or faulted condition transient during normal plant operation, the resulting deflections and associated stresses imposed on these structures and components would not exceed allowable stresses and deformation limits for the materials of construction. Limiting the stresses and deformations under such loading combinations provides an acceptable basis for the design of these structures and components to withstand the most adverse loading events which have been postulated to occur during service lifetime without loss of structural integrity or impairment of function. In addition, the design procedures and criteria used by the applicant in the design of the reactor internals constitutes an acceptable basis for satisfying the applicable requirements of General Design Criteria 1, 2, 4, and 10."

V. REFERENCES

1. 10 CFR Part 50, Appendix A, Criterion 1, "Quality Standards and Records."

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

3. 10 CFR Part 50, Appendix A, Criterion 4, "Environmental and Missile Design Bases."

4. 10 CFR Part 50, Appendix A, Criterion 10, "Reactor Design."

5. ASME Boiler and Pressure Vessel Code, Section III, Division 1, "Nuclear Power Plant Components," American Society of Mechanical Engineers.

6. Standard Review Plan 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment."

7. Standard Review Plan 3.9.3, "Pressure Retaining Components and Component Supports."

SRP 3.9.6