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POWER SYSTEMS

May 21, 1980
LD-80-025

Mr. Lester S. Rubenstein, Assistant Director
Reactor Systems
Division of Systems Integration
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Comments on Standard Review Plan SRP 4.2, Revision 2, Draft 1

Reference: (1) NRC letter K. Kniel to A. E. Scherer, dated February 29, 1980 Proposed Revision to Standard Review Plan, Section 4.2, "Fuel System Design."

(2) C-E letter, LD-79-058, A. E. Scherer to H. R. Denton, dated September 17, 1979, Comments on Standard Review Plan SRP 4.2

Dear Mr. Rubenstein:

Combustion Engineering has reviewed the subject revision to Standard Review Plan (SRP), Section 4.2 which was provided for comment by Reference (1). The comments developed by our review are attached.

We noted that the addition of Appendix A constituted the major difference between SRP 4.2, Revision 2 and the earlier version upon which we commented via Reference 2. Accordingly, our review emphasized the new material in Appendix A. For your convenience, however, the attached comments include our earlier comments in Reference 2 in addition to our new comments on Appendix A.

We appreciate the opportunity to comment on proposed revisions to the SRP. If you have any questions regarding our comments, please contact either me or Mr. G. D. Hess of my staff at (203)688-1911, Ext. 4579.

Very truly yours,

COMBUSTION ENGINEERING, INC.

A. E. Scherer
Director
Nuclear Licensing

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Attachment

cc: Mr. K. Kniel

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COMBUSTION ENGINEERING'S COMMENTS
ON STANDARD REVIEW PLAN 4.2

Combustion Engineering has the following concerns on Standard Review Plan 4.2, Draft 1 of Revision 2:

1. Comments on Design Bases; Section I.A.:

Section I.A. of SRP 4.2 suggests that all design bases become Specified Acceptable Fuel Design Limits (SAFDLs). The overall objective of the design bases is to avoid thermally or hydraulically induced fuel damage during normal steady state operation and during anticipated operational occurrences. These design bases have been established to provide automatic reactor trip or other corrective action in order to prevent the Specified Acceptable Fuel Design Limits in the Plant Final Safety Analysis Report (FSAR) from being reached. To include these design bases as SAFDLs would unnecessarily elevate them to a status they do not represent.

Combustion Engineering suggests the following changes:

The third sentence of paragraph I.A (which states "Once such limits are approved...") should be deleted.

The second and third sentences of paragraph II.A should be replaced with the following:

> The design bases should provide reasonable assurance that the four design objectives of Subsection I are met and that fuel rod failure criteria are not violated during anticipated operational occurrences. These design bases are listed in the following:

2. Comments on Acceptance Criteria; Section II:

2.1. Fuel System Damage; Section II.A.1.(b):

In Section II.A.1.(b), it seems that the cause of conservatism is adequately served by using a fatigue design curve which is reduced from data by a factor of 2 on stress or 20 on number of cycles so that requiring the predicted fatigue usage to be "significantly less than" a limit based on such a curve seems unnecessarily severe. Additionally this section makes reference to an article dealing with zircaloy components. Whereas not all structural members are zircaloy the applicability of this section to all structural members is unclear.

We recommend that reference 3 be deleted and that Section II.A.1.(b) be changed to read as follows:

(b) The cumulative number of strain fatigue cycles on the structural members mentioned in paragraph (a) above should not exceed the design fatigue lifetime, which is based on appropriate data and includes a safety factor of 2 on stress amplitude or a safety factor of 20 on the number of cycles. Other proposed limits must be justified.

2.2. Fuel System Damage; Section II.A.1.(c):

Considerable development and prototype testing of the fuel design is performed to demonstrate fretting wear is negligible. Thus any fretting wear which occurs would be very isolated and of an unexpected nature. To designate a specific allowance to provide for the rare and highly localized wear would produce an unnecessarily severe complication in fuel design.

We recommend that Section II.A.1.(c) be reworded as follows:

> Fretting wear at contact points on the structural members mentioned in paragraph (a) is not normally expected due to the considerable development and prototype testing of the fuel design prior to introduction into the reactor. Any fretting wear discovered should be analyzed with respect t the stress and fatigue limits in paragraphs (a) and (b) above and continued safe operation must be justified for those components exhibiting the fretting wear.

2.3 Fuel System Damage; Section II.A.1.(f):

Section II.A.1.(f) suggests that, "Fuel and burnable poison rod internal gas pressures should remain below the nominal system pressure during normal operation unless otherwise justified."

Combustion Engineering recommends that rather than suggesting fuel or poison rod damage phenomena having a threshold which is crossed right at the point where internal pressure begins to exceed the nominal coolant pressure Section II.A.1.(f) be reworded as follows:

> Fuel rod internal pressure increases with increasing burnup and toward end-of-life the total internal pressure, due to combined effects of the initial fill gas (if any) and the released fission gas can approach values comparable to the external coolant pressure. The maximum predicted fuel rod internal pressure should be consistent with the following criteria.
>
> 1. The primary stress in the cladding resulting from differential pressure will not exceed the stress limits specified in the FSAR.
>
> 2. The internal pressure will not cause the clad to creep outward from the fuel pellet surface while operating at the design peak linear heat rate for normal operation. Where the occurrence of internal rod pressures exceed normal system pressure safe operation may be justified by satisfying the appropriate criteria for cladding stress, strain and strain rate.

2.4. Fuel Rod Failure; Section II.A.2:

Section II.A.2 requires a fuel failure criterion to be given for each known failure mechanism. Failure criteria do not exist for all potential failure mechanisms and is so indicated in the SRP. Therefore, each known failure mechanism should be addressed rather than requiring a failure criterion to be established.

We recommend that the last two sentences of Section II.A.2 be replaced with the following:

> Fuel failure criteria selected by the licensee should provide reasonable assurance that fuel rods do not fail (during normal operation or anticipated operational occurrences) due to the following phenomena:

2.5. Fuel Rod Failure; Section II.A.2.(d):

Section II.A.2 lists phenomena which should be considered when evaluating fuel rod failure criteria. With respect to cladding collapse the following is stated in Section II.A.2.(d):

> "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".

The above statement should be revised to read:

> If axial gaps in the fuel 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 unless clad integrity in the collapsed condition is adequately proven.

2.5 Fuel Coolability; Section II.A.3:

Section II.A.3 discusses coolability and maintenance of a coolable geometry. As mentioned in the comments on Section I.A above the design criteria as described in the design bases of the FSARs have been established to avoid thermally or hydraulically induced fuel damage during normal steady state operation and during anticipated operational occurrences.

We recommend that the last sentence of Section II.A.3 be replaced with the following:

> The fuel design bases should provide a reasonable assurance that a coolable geometry is maintained as a result of the following phenomena:

2.6. Fuel Coolability; Section II.A.3.(b):

Section II.A.3 lists phenomena which should be considered when evaluating fuel rod coolability. With respect to violent expulsion of fuel the following is stated in Section II.A.3.(b).

> "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. The 280 cal/g limit should be used for PWR's and BWR's."

The following should be appended to the above statement:

> However, recent experiments (reference 1, 2, 3, 4) indicate that RIA fuel rod incipient failure may not be violent and that violent failures occur at energy depositions significantly higher than 280 cal/g. Therefore, an alternative criterion to limit violent fuel expulsion may be used if it is justified.

2.7. Design Evaluation; Section II.C.3.(f):

In Section II.C.3.(f), the cautionary words about temperature and pressure conservatisms working at cross purposes presumably refers to effects such as large diametral gaps leading to high temperatures and gas release but simultaneously providing more volume to accommodate it. Please recognize that in some cases (such as the one mentioned above), trying to be conservative on all counts could result in analyzing conditions which cannot physically exist - such as a fuel rod in which small gaps and low densification were assumed to determine volume while large gaps and high densification were assumed for fission gas release.

We recommend that the last sentence in Section II.C.3.(f) be changed to read:

> The reviewer should ensure that conservatisms that were incorporated for calculating temperatures do not introduce nonconservatisms or mutually exclusive conditions with regard to fuel rod pressures.

2.8. Testing, Inspection and Surveillance Plans; Section II.D.3:

In Section II.D.3. the wording of the first paragraph is vague as to the definition of new design features.

We recommend that this section be reworded by adding the following sentence to the first paragraph:

> New design features are those changes which are shown to significantly change fuel rod performance based on design and or safety analyses.

3. Comments on Appendix A:

3.1. Background; Section A:

Section A of Appendix A states that:

"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."

However, rod insertion is not always required such as in the case for the large break LOCA.

Combustion Engineering suggests that the second sentence of Section A be changed to read:

"SRP Section 4.2 states that the fuel system coolability should be maintained and that damage should not be so severe as to prevent control rod insertion when required during these low probability accidents."

3.2. Analysis of Loads; Section B.1:

Section B.1 addresses input for the fuel assembly structural analysis. The primary coolant system structural analysis provides input to the reactor internals analysis. The reactor internals analysis is used to provide input to the detailed fuel assembly analysis. Since the reactor internals analysis includes input from the primary coolant system analysis it appears more correct to reference the reactor internals analysis.

Also, the third sentence of Section B.1 states:

"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)."

Combustion Engineering interprets this sentence to mean that if a non-linear fuel assembly response is predicted then non-linear type structural analysis should be used.

We suggest that Section B.1 be reworded to read as follows:

B. ANALYSIS OF LOADS

1. Input

Input for the fuel assembly structural analysis comes from results of the reactor internals structural analysis, which is reviewed by the Mechanical Engineering Branch. Input for the fuel assembly response to a LOCA should include (a) motions of the core plate; core shroud, fuel alignment plate, or other relevant structures; these motions should correspond to the break that produced the peak fuel assembly loadings in the reactor internals analysis, and (b) transient pressure differences that apply loads directly to the fuel assembly. If the earthquake loads are large enough to produce a non-linear fuel assembly response, the seismic analysis should use non-linear structural analysis methods; if a linear response is produced, a spectral analysis may be used (see Regulatory Guide 1.60).

3.2. Analysis of Loads; Section B.2:

The third paragraph of Section B.2 states:

"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)."

It is presumed that the standard problem is associated with the methods review of the first paragraph. As it would be redundant to rework this problem for all plants of a standard design or those employing generic analytical methods, C-E recommends adding the following sentence:

"This sample problem need be worked once for all plants employing generic analytical methods."

3.3. Analysis of Loads; Section B.3:

3.3.1 In Section B.3. part (a) of the first paragraph states:

"(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."

The multiplication factor of 1.3 to be applied to lateral forces on the fuel is conservative to the point of being unrealistic. The recommendation assumes (1) a double-ended break (as opposed to a mechanistic break), (2) boiling down the whole length of the channel, (3) maximum crossflow occurs at the exact time the maximum LOCA lateral loads peaked, (4) that the crossflow velocity is exactly normal to a solid plane which extends the entire length of the fuel. The combination of these conservatisms led to a calculated maximum factor of 1.15. Furthermore, it was additionally hypothesized that two adjacent assemblies were moving in equal and opposite directions upon impact (additional .15). C-E observes that flashing is minimal during the period of significant loads (0-250 msec.). During the first 250 msec., flashing remains somewhat localized near the top of the core and does not penetrate below the center. When this is considered together with realistic assumptions on crossflow impingement and assembly motion, the multiplication factor should be 1.0.

3.3.2 The fourth sentence of paragraph 2 Section B.3 states:

"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."

C-E analyses have not exhibited this sensitivity and in consideration of this and Comment 3.3.1 above, C-E recommends rewording Section B.3 to state:

"3. Uncertainty Allowances

The fuel assembly structure models and analytical methods are probably conservative and input parameters are also conservative. Thus no additional conservatisms need be evaluated. However, if any part of the analysis (PWR or BWR) exhibits pronounced sensitivity to input variations, conservative margin should be added."

3.4. Audit; Section B.4:

Section B.4 calls for independent audit calculations by the reviewer. Combustion Engineering feels that conducting a review of methods, a standard problem, and structural representations, the requirement for an audit analysis becomes superfluous, especially if the standard problem is representative of the vendor's design.

We recommend that a standard problem be done once in conjunction with the generic methods review, and that the requirement for an audit analysis be eliminated.

3.5. Combination of Loads; Section B.5:

Section B.5 calls for the combination of LOCA and seismic loads. C-E considers that the combination of LOCA and seismic loads is unrealistic for fuel and reactor internals. The fuel loads resulting from the required analysis of unrealistic break sizes already provide an overly conservative basis for the structural evaluation of the fuel.

C-E recommends deleting this section.

3.6. Determination of Strength; Section C.1:

The last sentence in Section C.1 states:

"The extra margin in P_{crit} for irradiated grids is thus assumed to offset the unknown deformation behavior of irradiated grids beyond P_{crit}."

For clarity we recommend rewording the last sentence to read:

The increased P_{crit} for irradiated grids reflects the conservatism inherent in the selection of P_{crit} from data obtained with unirradiated grids.

3.7. Acceptance Criteria; Section D.1:

3.7.1 The first paragraph of Section D.1 calls for combined loads to be considered for the Loss of Coolant Accident. As mentioned in Comment 3.5 above, C-E considers the combination of LOCA and Seismic loads to be unrealistic.

We recommend that the word LOCA be substituted for the word combined throughout Section D.1.

3.7.2 The fifth sentence of paragraph 1 of Section D.1 states:

"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."

For the same reason as mentioned in Comment 3.5 and to be consistent with item (b) of paragraph 1 of Section D.1, C-E suggests the fifth sentence of paragraph 1 of Section D.1 be changed to read:

If LOCA grid loads exceed P_{crit} then grid deformation must be assumed and the effects of distorted grids on the peak clad temperature and oxidation must be assessed.

3.7.3 The second sentence of the second paragraph of Section D.1 states:

"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."

As mentioned in Comment 3.5 and reiterated in Comment 3.7.1 C-E considers the combination of LOCA and seismic loads to be unrealistic.

We recommend the second sentence of the second paragraph of Section D.1 be reworded as follows:

Control rod insertability must be demonstrated for the loads resulting from the most severe break that requires insertability.

3.8 Safe Shutdown Earthquake; Section D.2:

The third sentence of Section D.2 states:

"The second criterion must be satisfied for SSE loads alone if no analysis for combined loads is required by Paragraph 1."

For the reasons discussed in Comment 3.5, C-E recommends this sentence be reworded as follows:

The second criteria must be satisfied for SSE loads.

References

1. Z. R. Martinson, et al., "Reactivity Initiated Accident Test Series Test RIA 1-1 Quick Look Report," TFBP-TR-300, EG&G Idaho Inc., December 1978.

2. Z. R. Martinson, et al., "Reactivity Initiated Accident Test Series Test RIA 1-2 Quick Look Report," TFBP-TR-303, EG&G Idaho Inc., December 1978.

3. "Quarterly Progress Report on the Nuclear Safety Research Reactor (NSRR) Experiments October 1975 - March 1976," Translation Series NUREG/TR-0025, Japanese Atomic Research Institute, June 1976.

4. R. VanHouten, Progress of Japanese Experiments of the Nuclear Safety Research Reactor, NRC letter, dated December 21, 1978.