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1

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General Comment

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Attachments

Draft rev. 3 of 19.0 RCW comments submitted to the NRC

SUNSI Review Complete
Template = ADM-013

E-RIDS = ADM-03
Add = A. Cubbage (AEC)

consideration of

Subsequent to COL issuance, the staff may review the applicant's PRA (or portions thereof) in the context of licensing actions, following the guidance provided in Regulatory Guide (RG) 1.174 and RG 1.200, and SRP Sections 19.1 and 19.2 (previously SRP Chapter 19). Associated application-specific regulatory guidance and SRP sections should be consulted, while maintaining the validity of the staff findings associated with the licensing basis related to PRA and severe accidents.

the

The purpose of the staff's review is to ensure that the applicant has adequately addressed the Commission's objectives regarding ~~the appropriate way to address~~ severe accidents and ~~to~~ use PRA in the design and operation of facilities under review. These objectives are outlined in RG 1.206, Section C.I.19.2 and should be addressed in Section 19.1.1 of the applicant's Final Safety Analysis Report (FSAR).

The scope of a DC review is limited to the design-specific aspects within the scope of the design certification. The design-specific PRA developed during the DC stage may not identify site-specific information (e.g., local hazards, switchyard and offsite grid configuration, and ultimate heat sink) and may not explicitly model all aspects of the design (e.g., balance of plant). A seismic PRA cannot be performed without a site-specific probabilistic seismic hazard analysis (PSHA) and as-built information. Consequently, a seismic margin analysis (SMA) is acceptable.

This SRP provides guidance for reviewing PRA-based SMA submitted in support of a DC or COL application. DC/COL-ISG-20 (Reference 24) discusses post-DC activities to update the PRA-based SMA throughout the licensing process of new reactors, including COL action items and post-licensing activities, to ensure a coherent and consistent process for the quality of PRA-based SMA to adequately meet Title 10 of the *Code of Federal Regulations* (10 CFR) 52.47(a)(27), 10 CFR 52.79(a)(46), 10 CFR 52.79 (d)(1) and 10 CFR 50.71(h).

The applicant's design-specific PRA may include assumptions regarding site parameters and the interfaces with undeveloped aspects of the design. This is acceptable at the DC stage and results in the identification of PRA-based insights that include design, site, and operational assumptions. Although the staff has not published a format and content document specific to DC applications, RG 1.206, Section C.I.19, is intended to include all information needed for the staff to review a COL application that does not refer to a DC. Therefore, DC applicants are expected to provide the material in RG 1.206, Section C.I.19, except for those elements that require site-specific or plant-specific information not yet available.

As indicated above, format and content guidance for COL applications is provided in RG 1.206. COL applicants not referring to a DC should follow the guidance in RG 1.206, Section C.I.19. The staff will review the full scope of information requested by this guidance. Where the DC included generic analysis of external events, the COL applicant may demonstrate that the relevant parameters of the generic analysis bound the corresponding site-specific parameters. Alternatively, the COL applicant may show that a particular initiating event is too infrequent or inconsequential to affect core damage frequency (CDF) or large release frequency (LRF). Otherwise, the event must be included in the description of risk results and insights.

For a COL application that references a DC, the staff review of the PRA for the COL should focus on the plant-specific aspects of the PRA and site-specific design features that deviate assessment of the structural effects of postulated containment phenomenological challenges such as direct containment heating and ex-vessel explosions loads on the containment. The review and evaluation focus on the structural performance of the containment boundary as the ultimate barrier to radionuclide releases to the environment in a severe accident.

<u>COL Action Items and Certification Requirements and Restrictions</u>. For a DC application, the review will also address COL action items and requirements and restrictions (e.g., interface requirements and site parameters).

For a COL application referencing a DC, a COL applicant must address COL action items (referred to as COL license information in certain DCs) included in the referenced DC. Additionally, a COL applicant must address requirements and restrictions (e.g., interface requirements and site parameters) included in the referenced DC.

Review Interfaces

The organization responsible for structural engineering supports the review of the PRA and severe accident evaluation in two main areas: the applicant's evaluation of seismic contributors (specifically the seismic hazard analysis and estimation of seismic capacities (acceleration at which there is high confidence in low probability of failure) (HCLPF value) and the applicant's analysis of containment performance. This organization provides written input to the SER. Acceptance criteria for these sections are outlined below. , fire protection

Other organizations that use the PRA and severe accident evaluation results and insights in their programs, processes, and reviews (e.g., human factors, emergency preparedness, security, inspection, technical specifications (~~TS~~), regulatory treatment of non-safety systems (RTNSS), maintenance rule implementation) may need to interface with the PRA staff in evaluating these areas. PRA staff should be prepared to discuss the prioritization of structures, systems, and components (SSC) based on risk significance, as well as PRA-based insights related to the design. This information will help reviewers of other areas focus their review on safety-significant issues. In addition, PRA staff reviews Tier 1 to ensure appropriate treatment of important insights and assumptions from the PRA as described in Section C.II.1 of RG 1.206 and SRP Section 14.3.

The organizations that are responsible for the review of the design of the plant for external natural hazards (e.g., earthquakes, high winds, external fires, external flooding), hazards related to human activities (e.g., transportation and local industry) and in-plant area hazards (internal fire and flooding) may need to support the PRA staff in reviewing these hazards. The PRA staff may also request support from the organizations that review the systems and thermal-hydraulic (T-H) analyses to ensure that the applicant's PRA properly considers and addresses important issues (e.g., failure mechanisms, system interactions, and T-H modeling and uncertainties).

The organizations responsible for the review of severe accident issues, including severe accident management alternatives, in Sections 7.2 and 7.3 of the Environmental Report (ER) need to maintain coordination with the PRA staff to assure consistency in the review of severe accident information given in the ER and the review of severe accident evaluations in Chapter 19 of the FSAR.

5. 10 CFR 52.79(a)(17) states that a COL application for a LWR design must contain an FSAR that provides the information with respect to compliance with ~~a~~ the technically relevant positions of the Three Mile Island (TMI) requirements in 10 CFR 50.34(f), with the exception of 10 CFR 50.34(f)(1)(xii), 10 CFR 50.34(f)(2)(ix), and 10 CFR 50.34(f)(3)(v).

6. 10 CFR 52.79(a)(18) states that a COL application must contain the information required by 10 CFR 50.69(b)(2), if the applicant seeks to use risk-informed treatment of SSCs in accordance with 10 CFR 50.69.

7. 10 CFR 52.79(a)(38) states that a COL application for a LWR design must contain an FSAR that includes a description and analysis of design features for the prevention and mitigation of severe accidents, for example, challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure core melt ejection, hydrogen combustion, and containment bypass.

8. 10 CFR 52.79(a)(46) states that a COL application must contain an FSAR that includes a description of the plant-specific PRA and its results. With respect to this regulation, the following items are noted:

   A. The Statement of Consideration (72 FR 49387) for the revised Part 52 states the understanding that the complete PRA (e.g., codes) would be available for NRC inspection at the applicant's offices, if needed. The NRC expects that, generally, the information that it needs to perform its review of the COL application from a PRA perspective is that information that will be contained in applicants' FSAR Chapter 19.

   B. RG 1.206 provides guidance on reporting PRA-related information. As discussed in the Statement of Consideration (72 FR 49387) for the revised Part 52 the guidance focuses on qualitative description of insights and uses, but also acknowledges that some quantitative PRA results should be submitted.

   C. In accordance with the requirements in 10 CFR 52.79(d)(1), COL applicants must provide the basis for determining that the SMA of the DCD is applicable to the proposed plant or perform a plant-specific SMA. In any case, a plant-specific supplement must identify any SSCs outside the scope of the DCD that are relied upon for safe shutdown after an earthquake.

9. 10 CFR 52.79(c)(1), (d)(1), and (e)(1) state that if a COL application references a standard design approval, standard DC, or the use of one or more manufactured nuclear power reactors licensed under Subpart F of 10 CFR Part 52, then the plant-specific PRA information must use the PRA information for the design approval, design certification, or manufactured reactor, respectively, and must be updated to account for site-specific design information and any design changes or departures. The Statement of Consideration (72 FR 49388) for the revised Part 52 states in the case where a COL application is referencing a DC, the NRC only expects the design changes and

differences in the modeling (or its uses) pertinent to the PRA information to be addressed to meet the submittal requirement of 10 CFR 52.79(d)(1).

10. Section IV.A.2.a of each DCR states that a COL application which references a DC must include a plant-specific FSAR containing the same type of information and using the same organization and numbering as the generic DCD for the certified design, as modified and supplemented by the applicant's exemptions and departures.

<u>SRP Acceptance Criteria</u>

Background

Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for the review described in this SRP section. The SRP is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is ~~required~~ expected to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with the NRC regulations.

The SRP acceptance criteria are derived from Commission direction and staff guidance published in multiple documents, including the following:

1. Policy Statement, "Severe Reactor Accidents Regarding Future Designs and Existing Plants," 50 FR 32138, August 8, 1985.

2. Policy Statement, "Safety Goals for the Operations of Nuclear Power Plants," 51 FR 28044, August 4, 1986.

3. Policy Statement, "Nuclear Power Plant Standardization," 52 FR 34884, September 15, 1987.

4. Policy Statement, "Regulation of Advanced Nuclear Power Plants," 59 FR 35461, July 12, 1994.

5. Policy Statement, "The Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities," 60 FR 42622, August 16, 1995.

6. SECY-90-016, "Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," Agencywide Documents Access and Management System (ADAMS) Accession No. ML003707849, dated January 12, 1990, and the related staff requirements memorandum (SRM), ADAMS Accession No. ML003707885, dated June 26, 1990.

7. SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs," ADAMS Accession No. ML003708021, dated April 2, 1993, and the related SRM, ADAMS Accession No. ML003708056, dated July 21, 1993.

Page 19.0-25 implies that future fire PRA's need to be developed in accordance with NUREG 6850. Paragraph 3 below indicates that use fo the FIVE methodology is still acceptable.

3. The Commission approved the use of simplified probabilistic methods, such as but not limited to the Electric Power Research Institute (EPRI) Fire-Induced Vulnerability Evaluation (FIVE) methodology, to evaluate fire risk.

4. The Commission approved the staff's position that advanced LWR vendors should perform bounding analyses of site-specific external events likely to be a challenge to the plant (such as river flooding, storm surge, tsunami, volcanism, high winds, and hurricanes). When a site is chosen, its characteristics should be compared to those assumed in the bounding analyses to ensure that the site is enveloped. If the site is enveloped, the COL applicant need not perform further PRA evaluations for these external events. The COL applicant should perform site-specific PRA evaluations to address any site-specific hazards for which a bounding analysis was not performed or which are not enveloped by the bounding analyses to ensure that no vulnerabilities due to siting exist.

In addition, Regulatory Issue Summary (RIS) 07-06, "Regulatory Guide 1.200 Implementation," dated March 22, 2007, states that PRAs required under 10 CFR Part 52 should use NRC-endorsed consensus standards to the extent practicable.

Acceptance Criteria

Based on these guidance documents and the major objectives stated in Subsection I, the staff has established the following acceptance criteria for its review. These acceptance criteria apply to the PRA and severe accident evaluation in general. Specific subsets of the criteria apply to individual elements of the applicant's analyses (e.g., Level 1 shutdown PRA, severe accident management).

1. The staff will determine that the applicant has used the PRA to do the following:

   A. Identify and address potential design features and plant operational vulnerabilities; for example, vulnerabilities in which a small number of failures could lead to core damage, containment failure, or large releases that could drive plant risk to unacceptable levels with respect to the Commission's goals

   B. Reduce or eliminate the significant risk contributors of existing operating plants applicable to the new design by introducing appropriate features and requirements

   C. Select among alternative features, operational strategies, and des

should this be 1E-5 instead of 1E-6?

2. The staff will determine that the applicant has adequately demonstrated that:

   A. The risk associated with the design compares favorably against the Commission's goals of less than 1×10^{-4} per year (/yr) for CDF and less than 1×10^{-6}/yr for LRF.

   B. The design compares favorably against the Commission's approved use of a CPG, which includes (1) a deterministic goal that containment integrity be

Regulatory Treatment of Non-safety Systems (RTNSS) process (See SRP Section 19.3 for additional information).

9. Consistent with the guidance in Section 2.5 of RG 1.174, the staff expects that the applicant will have subjected its PRA to quality control. In accordance with the Statement of Consideration for the revised Part 52 (72 FR 49365), the PRA is not part of the design-basis information, therefore, the PRA is not subject to the quality assurance requirements of 10 CFR Part 50, Appendix B. However, the applicant should address the following methods of quality control:

   A. Use

   General comment regarding the decision to include some level of PRA detail in the FSAR Chapter 19 going forward. PRA models are never complete and are updated routinely. If we put this detailed information in the FSAR, the FSAR change processes will have to be followed to update that PRA information. I agree PRA should not be subject to Appendix B requirements but as soon as we include that level of detail in the FSAR we are subject to maintaining that PRA information current under FSAR update requirements. My personnel opinion is that the information in Chapter 19 needs to be left at a high/programmatic level and the modelling details and insights controlled under a licensee controlled document not subject to FSAR revision requirements.

   B. Use ... prov ... info ... d

   C. Doc ... well ... s

   D. Use ... are ... or d

   E. Pee

10. The staff will determine that the technical adequacy of the PRA is sufficient to justify the specific results and risk insights that are used to support the DC or COL application. Toward this end, the applicant's PRA submittal should be consistent with prevailing PRA standards, guidance, and good practices as needed to support its uses and applications and as endorsed by the NRC (e.g., RG 1.200 and SRP Section 19.1). As discussed in RGs 1.174 and 1.200, the quality of a PRA is measured in terms of its appropriateness with respect to scope, level of detail, and technical adequacy. The applicant's adherence to the recommendations provided in RGs 1.174 and 1.200 pertaining to quality and technical adequacy will result in a more efficient and consistent NRC staff review process. With respect to PRA quality, the following items are noted:

    A. There are no regulatory requirements in 10 CFR Part 52 that specifically pertain to PRA quality with respect to DC or COL applications.

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1 Peer review of the DC PRA is not required prior to application. However, if a peer review was conducted prior to the application; the staff should examine the peer review report. If a certain aspect of the PRA deviates from accepted good practices, the applicant/holder should justify that this deficiency does not impact the PRA results or risk insights. Otherwise, applicants/holders need to correct the deficiency and resubmit the PRA results and risk insights. If a peer review has not been performed, the applicants/holders should justify why their PRAs are adequate in terms of scope, level of detail, and technical acceptability. If the applicant's/holder's justification fails to provide the staff with an appropriate level of confidence in the models, results, and insights, the staff should conduct an audit of the applicant's/holder's PRA against the technical elements described in RG 1.200 to determine the PRA technical adequacy.

key

14. The staff will determine that FSAR Chapter 19 includes PRA qualitative results, including the identification of PRA assumptions, the identification of PRA-based insights, and discussion of the results and insights from importance, sensitivity, and uncertainty analyses.

15. The staff will determine that the internal events PRA quantitative results includes internal fires and floods and their contributions.

16. It is acceptable for applicants to report significant risk contributors by separate hazard groups (i.e., provide separate lists of the contributors for internal events, the contributors for internal floods, the contributors for seismic events, the contributors for internal fires, etc.). Applicants may also elect to develop an integrated list of significant risk contributors that summarize the results across all hazard groups.

17. In the context of the PRA results and insights, the term "significant" is intended to be consistent with its definition provided in RG 1.200. The definitions of "significant accident sequence" and "significant contributor" are suitable for both LERF and LRF. Using any other definition of "significant" inconsistent with the definitions provided by RG 1.200 shall be subject to additional staff review and approval.

18. PRA maintenance should commence at the time of application for both DC and COL applicants. Once the certification is issued, the generic PRA would not need to be updated except as appropriate in connection with a DC amendment request. The PRA performed in support of a COL application should be updated to reflect plant modifications if there are changes to the design during the design, construction and operation phases of the facility. COL applicants should describe their PRA maintenance process in FSAR Chapter 19, including planned implementation of the program during design, construction and operation phases of the facility. The NRC expects COL applicants to describe their approach for maintaining and periodically upgrading the PRA in accordance with RG 1.206, Section C.I.19.7 and RG 1.200. For purposes of reporting the effects of plant modifications and changes to the NRC in accordance with the requirements of 10 CFR 50.71(e), the NRC expects the following when changes affect the PRA:

    A. PRA numerical changes should be reported when the cumulative risk impact of the changes resulting from the plant modifications, design changes or departures from the DC is more than a 10% change (either positive or negative) in the total core-damage frequency or total LRF from what was previously reported.

    B. All changes in key assumptions per RG 1.200 and all changes in risk insights as defined in RG 1.206 including differences between the updated risk insights and the certified design risk insights should also be reported to the NRC in accordance with the guidance in Section C.III of RG 1.206.

    C. All changes or departures from the design that result in a revision of PRA-based qualitative results should also be reported to the NRC.

I think this was first time use of this acronym. May want to define as Common Cause Failure (CCF)

19. 10 CFR 50.71(h)(2) states that each COL holder must maintain and upgrade the PRA required by 10 CFR 50.71(h)(1). This means that COL holders, in accordance with 10 CFR 50.71(h), must upgrade the PRA used to support the COL to cover those initiating events and modes of operation contained in NRC-endorsed consensus standards that exist one year prior to each required upgrade. The ASME PRA Standard describes "PRA upgrade" as the incorporation into a PRA model of a new methodology or significant changes in the scope or capability. This could include items such as new human error analysis methodology, new data, updated methods, new approaches to quantification or truncation, or new treatment of CCF.

20. RG 1.200 describes the elements of a PRA maintenance and update program that is acceptable to the staff. If the staff can confirm that the applicant's proposed program includes the key elements described in RG 1.200, it may conclude that such a program is acceptable.

21. In the analysis of high winds, tornado frequencies developed with methods and data in NUREG/CR-4461, Revision 1, and based on data for the central region of the United States will normally be acceptable because the central region of the country has the highest occurrence rate of tornadoes and the highest tornado intensities.

22. In the analysis of high winds, tornados may not always be bounding with respect to the damaging effects of missiles. In coastal regions prone to severe hurricanes a missile generated by hurricane force winds may be more damaging than one created by a tornado (See Interim Staff Guidance DC/COL-ISG-024 pertaining to RG 1.221 for treatment of hurricane missiles).

23. The staff will determine that the applicant has performed an evaluation of containment structural integrity for internal pressure loadings above design-basis pressure in accordance with guidelines in RG 1.216 or an acceptable alternative.

Acceptance Criteria for a PRA-based SMA

24. The staff will determine that the applicant has performed a PRA-based SMA to determine the seismic capacity of the plant and for each sequence that may lead to core damage or large release.

25. The design-specific plant system and accident sequence analysis for a PRA-based SMA is performed in accordance with, at a minimum, the Capability Category I requirements of Section 5-2.3 of Part 5 of the ASME/ANS PRA standard (Reference 1), with the exceptions that the analysis should not be based on site-specific and plant-specific information and should not rely on an as-built and as-operated plant.

26. Screening of rugged SSCs may be performed in a PRA-based SMA based on the DC's Certified Seismic Design Response Spectra (CSDRS) with its peak ground acceleration (PGA) scaled by a factor of 1.67. The basis for the screening should be adequately documented and ensure that the so-called "super element," as described in Note 3 of Section 5-2.3 of Part 5 of the ASME/ANS PRA standard, will not control the plant seismic margin capacity.

substantive change need to be defined. Possibly by using paragraph #18 guidance.

35. The Min-Max method[2] is acceptable for computing sequence-level HCLPF values.

36. The staff will determine that the design-specific plant-level HCLPF value has been demonstrated to be equal to or greater than 1.67 times the CSDRS PGA.

III. REVIEW PROCEDURES

General Principles

The reviewer will select material from the procedures described below, as may be appropriate for a particular case. These review procedures are based on the identified SRP acceptance criteria. If the application deviates from these acceptance criteria, the staff should determine that the proposed alternative provides an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

The staff will review the description of the PRA and its results in order to make the evaluation findings described in this SRP section. In addition to a qualitative description, the staff will review some quantitative results (e.g., mean core-damage frequencies, mean large release frequencies, and importance measures). The NRC should review this information to ensure that it is able to conclude that the applicant has performed sufficiently complete and scrutable analyses, the results and insights support the application, and the applicant has in place programs and processes that will enable it to maintain an up-to-date PRA for these uses and applications.

RG 1.206, Section C.I.19, provides detailed guidance on the information expected to be included in Chapter 19 of the applicant's FSAR. Specific application content that the staff should review is not reproduced in this SRP section, only review guidance on specific topics. Instead, the staff should use RG 1.206 in parallel with the SRP to determine that the appropriate topics have been addressed by the applicant. Although Chapter 19 of the applicant's FSAR should include the information needed by the staff to determine that the relevant acceptance criteria have been met, some staff audits of the PRA and supporting analyses may be necessary to fully understand, review, and confirm the PRA results, insights, and associated analytical bases. The staff will refer to the summary reports from these audits in the SER. For instances in which additional information is needed to complete the staff's review of the FSAR, the staff will use the RAI process. Reviewers utilizing the RAI process should make it clear to the applicant that if an RAI results in substantive change to information in the FSAR or DCD, these documents must be revised to reflect the new information.

For additional information, the staff should consider the information provided in Chapter 19 of past SERs for advanced LWRs. The NRC issued the Final Safety Evaluation Report (FSER) for

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[2] A method used in the determination of the functional and accident sequence level of fragility. The overall fragility of a group of inputs combined using OR logic (i.e., seismic event tree nodal fault tree) is determined by the lowest (minimum) input. Conversely, the overall fragility of a group of inputs combined using AND logic (i.e., seismic event tree sequence) is determined by the highest (maximum) input.

non power modes

review will be needed to assure the validity of the PRA model. Reviewers should confirm that applicants identify and describe all the specific plant operating states (POS) in a refueling outage between the time the output breaker to the grid is opened for plant shutdown and when it is closed to resume power operation after the outage. The reasonableness of the PRA model cannot be judged without a description of each POS that includes an estimate of the expected time in the POS, a description of the expected changes in configuration of the nuclear steam supply system, a description of the methods of removing heat from the fuel during each POS, a description of the automatic and human actions expected to occur during each POS and an assessment of the potential upset conditions and human errors during each POS that could contribute to a loss of decay heat removal.

Design-Specific PRA (Level II PRA)

For DC applications and COL applications not referencing the Level II PRA in the DC, the reviewer[4] carries out an independent assessment of the plant response to selected severe accident scenarios using the latest version of the MELCOR computer code. The assessment should examine accident scenarios from the PRA, which are chosen based on a combination of frequency, consequence, and dominant risk. Some of these scenarios should be similar or identical to sequences analyzed by the applicant and reported in the PRA. The reviewer compares the results of corresponding sequences and release categories in the two studies. If the results of the assessment do not support and confirm the applicant's simulation of the accident progression, analysis methodology, and interpretations of its analyses of the reactor, containment, and system response to severe accidents, the reviewer engages with the applicant to resolve the differences in results.

Design-Specific PRA (PRA for Non-Power Modes of Operation)

1. Given that shutdown risk may be highly outage-specific, the staff reviews the shutdown PRA insights to confirm that operational assumptions used to develop an average shutdown model (e.g., use of nozzle dams, outage schedule, containment status, procedural requirements) have been clearly documented in the FSAR. If licensee practices deviate dramatically from these assumptions in the future, the insights obtained from the shutdown PRA may no longer be valid. It is the COL applicant's responsibility to confirm the assumptions made at the DC stage, and if done properly should capture any significant differences.

2. The staff reviews the applicant's assumptions related to equipment availability and compares them to TS requirements. Risk-significant equipment should be evaluated with respect to 10 CFR 50.36(c)(2)(ii)(D) to determine whether additional TS requirements are needed. The staff may also review the results of sensitivity studies

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[4] Support from an independent contractor or staff in the Office of Research may be necessary.

generally

performed to demonstrate the risk benefit of equipment that is controlled only by voluntary administrative controls (e.g., maintenance rule implementation).

3. The staff reviews the applicant's implementation of the applicable expeditious actions outlined in NRC Generic Letter (GL) 88-17 (Reference 23). The staff needs to ensure that the applicant is meeting the expeditious actions consistent with the guidance for meeting the guidelines in GL 88-17 which are described in detail in enclosures 1 and 2 of the GL. Deviation from GL 88-17 guidance could lead to configurations where cold leg penetrations are permitted to be open in Pressurized Water Reactors (PWRs) without the appropriate steam generator (SG) manway open or when nozzle dams are installed in the wrong order. Such configurations may invalidate PRA results. Staff may review NRC Information Notice No. 88-36 (Reference 26) to understand the risks associated with these configurations.

4. The staff reviews the applicant's implementation of industry guidance for safety during outages provided in NUMARC-91-06 (Reference 4). In particular, the staff should assure that, if the applicant plans to use freeze seals, the potential for loss-of-coolant accidents due to failed freeze seals has been considered in the PRA. The potential for such accidents is discussed in NRC Information Notice No. 91-41 (Reference 25). Reviewers should also confirm the existence of an adequate means to control reactor vessel level and an adequate means to control reactor vessel temperature and pressure during shutdown in Boiling Water Reators (BWRs).

5. Accidents during non-power modes of operation are not part of the design bases of the facility. Consequently, non-power operations, associated accident sequences and specific accident phenomanolgy are not considered in the review of the accident analyses provided in Chapter 15 of the FSAR. Indeed, the staff's review of the level of safety during non-power modes of operation provided by the design of the facility and operating procedures and controls in place is limited to the review of the PRA for non-power modes of operation. This puts additional burden on the PRA reviewer to pursue issues, as necessary, to assure that the PRA model has fidelity and the assumptions in the risk analyses are justified. In some cases the reviewer may need to engage reviewers from other technical branches that have expertise in a particular areas (e.g., systems operation, T-H performace, operating experience). Reviewers should therefore be aware of the following issues related to safety during non-power modes of operation:

   A. Based on previous PRAs, studies by the EPRI and studies performed by the staff, roughly 80 percent of risk for traditional PWR designs occurs during periods when the reactor coolant system is drained and open (midloop operation is a subset of this condition).

   B. The time it takes to reach boiling in the reactor vessel following loss of the decay heat removal function can be very short during PWR midloop operation (e.g. 12 minutes). Steaming into the containment will lead to intolerable conditions that could seriously affect the ability of personnel to close the containment.

C. During reduced inventory operation in a PWR a large vent for the reactor coolant system (RCS), such as a hot leg SG plenum man way, is necessary before opening a cold leg penetration to prevent expelling water from the core following a loss of residual heat removal. RCS piping penetrations may exist below the active fuel and pathways may exist via connected systems that could lead to draining the reactor vessel. In these cases reviewers should identify the isolation functions available and operable and assure that they are treated accurately in the PRA model.

Design-Specific PRA (PRA-Based Seismic Margins Analysis (SMA)

1. Staff responsible for the review of the description and results of the applicant's PRA review the design-specific plant system and accident sequence analysis in accordance with the acceptance criteria given in Section II of this SRP.

2. Staff responsible for the review of the seismic and structural design of the facility review (1) the applicant's evaluation of seismic fragilities, and (2) the applicant's determination of plant-level HCLPF in accordance with the acceptance criteria given in Section II of this SRP.

The staff reviewing the plant system and accident sequence analysis verifies that the applicant has considered random equipment failures, seismic interactions, as well as operator actions in the plant system and accident sequence analysis as applicable. It is important that the plant systems analysis focus on those sequences leading to core damage or containment failures, including applicable sequences leading to the following containment failures: (1) loss of containment integrity, (2) loss of containment isolation, and (3) loss of function for prevention of containment bypass. The applicant should address the following operating modes in the analysis: (1) at power (full power), (2) low power, and (3) shutdown.

Design-Specific PRA (Treatment of Internal Fires)

this paragraph implies that the only acceptable FPRA is one developed per NUREG 6850 which was intended for plants transitioning to NFPA 805. New plant fire protection programs may wish to adopt NFPA 806. Will it be the expectation that new plants adopt NFPA 806 performance based risk informed fire protection programs?

1. The reviewer considers the extent to which applicant's
with internal fires conforms to the guidance in NUREG/
PRA Methodology for Nuclear Power Facilities," issued
considers the NUREG/CR-6850 approach to be a state
approach because methodological issues raised in pas
individual plant examination of fire analyses, have beer
to the extent allowed by the current state-of-the-art. N
acceptable method for performing fire PRA to support a
or a COL. Reviewers my find that applicants for desigr
implementing the analysis tasks in NUREG/CR-6850 tl
in NUREG/CR-6850. This can occur when the specific
sources, and target locations in each fire zone of the pl
design certification application is submitted. Such an approach may be acceptable if conservative assumptions are used such that it is reasonable to conclude that the results bound those expected with the more detailed approach described in NUREG/CR-6850

A. The applicant's analyses should be comprehensive in scope and address all applicable internal and external events and all plant operating modes. Since some aspects of the applicant's approach may involve non-PRA techniques to address specific events (e.g., PRA-based seismic margins), the PRA staff review should ensure that the scope of the applicant's analyses is appropriate for the identified uses and applications of the PRA.

B. The level of detail of the applicant's PRA should be commensurate with the identified uses and applications of the PRA (e.g., sufficient to gain risk-informed insights and use such insights, in conjunction with assumptions made in the PRA, to identify and support requirements important to the design and plant operation). The PRA should reasonably reflect the actual plant design, construction, operational practices, and relevant operational experience of the applicant and the industry. The burden is on the applicant to justify that the PRA approach, methods, and data, as well as the requisite level of detail necessary for the NRC staff's review and assessment, are appropriate. RGs 1.174 and 1.200 provide additional guidance on the level of detail that should be included in the PRA. If detailed design information (e.g., regarding cable and pipe routing) is not available or if it can be shown that detailed modeling does not provide significant additional information, it is acceptable to make bounding-type assumptions consistent with the guidelines in RG 1.200. However, the risk models should still be able to identify vulnerabilities as well as design and operational requirements such as ITAAC and COL action items. In addition, the bounding assumptions should not mask any risk-significant information about the design and its operation.

2. The staff will determine that the applicant has performed sensitivity studies sufficient to gain insights about the impact of uncertainties (and the potential lack of detailed models) on the estimated risk. The objectives of the sensitivity studies should include (1) determining the sensitivity of the estimated risk to potential biases in numerical values, such as initiating event frequencies, failure probabilities, and equipment unavailabilities, (2) determining the impact of the potential lack of modeling details on the estimated risk, and (3) determining the sensitivity of the estimated risk to previously raised issues (e.g., motor-operated valve reliability). As noted in Element 1.1 of Table A-1 in Appendix A to RG 1.200, special emphasis should be placed on PRA modeling of novel and passive features in the design, as well as addressing issues related to those features, such as digital instrumentation and control, explosive (squib) valves, and the issue of T-H uncertainties.[6]

this was already covered on page 19.0-21. I don't see the value added in keeping this footnote.

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[6] ~~The issue of T-H uncertainties arises from the "passive" nature of safety-related systems used for accident mitigation. Passive safety systems rely on natural forces, such as gravity, to perform their functions. Such driving forces are small compared to those of pumped systems, and the uncertainty in their values, as predicted by a "best estimate" T-H analysis, can be of comparable magnitude to the predicted values themselves. Therefore, some accident sequences with a frequency high enough to impact results, but which are not predicted to lead to core damage by a best estimate T-H analysis, may actually lead to core damage when T-H uncertainties are considered in the PRA models.~~