REVISION OF USNRC REVIEW GUIDANCE FOR PRA AND SEVERE ACCIDENTS

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Abstract

Section 19.0 of the USNRC Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants provides guidance for the NRC staff's review of the description and results of a design specific Probabilistic Risk Assessment (PRA) for a design certification (DC) and a plant-specific PRA for a combined license (COL) application. This SRP section also provides guidance for the NRC staff's review of an applicant's deterministic evaluation of design features for the prevention or mitigation of severe accidents. Section 19.0 of the SRP was last updated in June, 2007. Since that time the NRC staff has completed review of one amended application for design certification and two applications for a Combined License (COL). The staff is nearing completion of its review of another complete application for DC and in the process of reviewing additional DC applications or amendments and several more applications for a COL. As a result the NRC staff has gained a substantial amount of experience in its review of the information provided in Chapter 19 of applications for DC and COL. Some of this experience has been translated into revised and updated guidance and documented in several Interim Staff Guidance (ISG) documents that have been available to the public on the USNRC public website (www.nrc.gov). They include DC/COL-ISG-3 ("PRA Information to Support Design Certification and Combined License Applications"), DC/COL-ISG-20 ("Seismic Margin Analysis for New Reactors Based on Probabilistic Risk Assessment"), and DI&C-ISG-3 ("Review of New Reactor Digital Instrumentation and Control Probabilistic Risk Assessments"). The revision of SRP Section 19.0 has been released for public comment. It will incorporate this interim staff guidance and will include additional guidance based on experience gained in the review of new reactor applications. This includes review guidance for the following areas:

- *Technical adequacy of Level 1 PRA*
- *Level 2 PRA and severe accident evaluation*
- *PRA for non-power modes of operation*
- *Assessment of seismic risk*
- *Assessment of high winds in the PRA*
- *Assessment of internal fires in the PRA*
- *Specific issues associated with passive designs*
- *Topics specific to Integral Pressurized Water Reactors*
- *Modeling of digital instrumentation and control (I&C) systems in the PRA*

In this paper much of the new guidance being included in Section 19.0 of the SRP will be described. Emphasis will be placed on the guidance expected to be of most interest to developers and reviewers of PRA for new nuclear power plants.

1. Introduction and Summary

The Standard Review Plan (SRP) provides guidance to United States Nuclear Regulatory Commission (USNRC) staff in performing safety reviews of light-water nuclear reactor power plants. The SRP scope includes construction permit (CP) or operating license (OL) applications (including requests for amendments) submitted under Title 10 of the Code of Federal Regulations (10 CFR) Part 50 (USNRC, 2012a). The scope also includes applications for early site permits (ESP), design certifications (DC), combined licenses (COL), standard design approvals (SDA), or manufacturing licenses (ML) under 10 CFR Part 52 (USNRC, 2012b) (including requests for amendments). The SRP is contained in NUREG-0800 (USNRC, 2007a) and includes an introduction plus a large number of sections that cover the topics addressed in the USNRC's safety review.

The principal purpose of the SRP is to assure the quality and uniformity of USNRC safety reviews. It is also the intent of the SRP to make information about regulatory matters widely available and to improve communication between the USNRC, interested members of the public, and the nuclear power industry, thereby increasing understanding of the USNRC's review process.

The USNRC is currently updating a number of SRP Sections in preparation for the review of applications for design certification it expects to receive over the next few years. A few of the sections being updated relate to probabilistic risk assessment (PRA). They include the introduction to the SRP, SRP Section 19.0, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors", and SRP Section 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities."

The introduction to the SRP is being revised to include a description of the risk-informed, integrated review framework to be applied by the staff for new Integral Pressurized Water Reactor (iPWR) DC and COL applications under 10 CFR Part 52. This framework includes a graded approach to review of structures, systems and components (SSCs) in the plant design based in part on their risk significance as determined with results from an applicant's PRA. The framework is described in more detail in a paper presented at a conference held in 2011 and sponsored by the Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI) (NEA, 2012), and will not be discussed further in this paper.

The NRC staff uses SRP 19.0 in its review of information provided in Chapter 19 of an applicant's DC Final Safety Analysis Report (FSAR) or COL FSAR. The purpose of this review is to ensure that an applicant has adequately addressed the Commission's objectives regarding the appropriate way to treat severe accidents and use PRA in the design and operation of facility. These objectives are outlined in Regulatory Guide (RG) 1.206, Section C.I.19.2 (USNRC, 2007b). Revision 3 of SRP Section 19.0 is being developed to (1) consolidate guidance for the review issued on an interim basis after Revision 2 of SRP Section 19.0 was issued and (2) incorporate new guidance for review of Chapter 19 of an applicant's DC FSAR or COL FSAR based on experience gained in the review of DC and COL applications after the interim staff guidance was issued. Revision 3 of SRP 19.0 includes new acceptance criteria and review procedures developed during NRC reviews of PRA information and severe accident assessments in DC and COL applications. The new guidance included in SRP 19.0 is the principal subject of this paper and is discussed further below.

Revision 3 of SRP Section 19.1(USNRC, 2012c) was recently updated and issued for use. This revision reflects requirements in §50.71(h) of the NRC's regulations (USNRC, 2012d) for holders of a COL to develop, maintain and update a plant-specific Level 1 and Level 2 PRA. The revision also reflects several important related documents that were either developed or updated since the last revision of the SRP section. These documents include Revision 2 of Regulatory Guide 1.200 on PRA Technical Adequacy (USNRC, 2009), the combined ASME/ANS PRA Standard and Addendum A (ASME/ANS, 2009), industry guidelines for fire probabilistic risk assessment peer review process (NEI, 2008a) and the revised industry guideline for performing PRA peer review using the ASME PRA Standard (NEI, 2008b). SRP 19.1 now provides appropriate guidance for review of the technical adequacy of a baseline PRA that supports a proposed license amendment from a holder of a COL under 10 CFR Part 52, as well as a holder of an OL under 10 CFR Part 50. SRP 19.1 will not be discussed further in this paper.

2. Revised Review Guidance in SRP 19.0 for PRA and Severe Accidents

A large number of changes have been made to the review guidance in SRP 19.0. These changes include new or revised acceptance criteria, review procedures and expectations regarding the scope and level of detail of information submitted for review. This paper focuses on a subset of the changes believed to be of most interest to people working on the development or review of PRA for new nuclear power plants. This subset of changes includes new or revised guidance for conducting reviews of information in the following subject areas each of which is summarized below.

- Technical adequacy of Level 1 PRA
- Level 2 PRA and severe accident evaluation
- PRA for non-power modes of operation
- Assessment of seismic risk
- Assessment of high winds in the PRA
- Assessment of internal fires in the PRA
- Specific issues associated with passive designs
- Topics specific to iPWRs
- Modeling of digital I&C systems in the PRA

The following specific review procedures have been included in Revision 3 of SRP 19.0 for completing the reviews in the above areas.

2.1 Level 1 PRA Technical Adequacy

The reviewer confirms that the applicant has: (1) identified those high-level requirements or attributes of the applicable PRA standard that the PRA did not embody, (2) addressed the impact on the qualitative and quantitative results of the PRA when excluding those high-level requirements or attributes of the standard that are applicable but have not been incorporated. RG 1.200 contains the staff's guidance concerning PRA technical adequacy and peer review. Peer review of the DC or COL PRA is not required prior to receipt of an application. However, if a peer review or self-assessment was conducted prior to receipt of the application, the staff should examine the documented results. If a peer review has not been performed, and the applicant 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 PRA against the technical elements described in RG 1.200 to confirm that the PRA is technically adequate to support the application.

2.2 Level 2 PRA and Severe Accident Evaluation

For DC applications, and COL applications not referencing the Level 2 PRA in the DC, the reviewer carries out an independent assessment of the plant response to selected severe accident scenarios using an acceptable NRC evaluation tool. 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.

The reviewer evaluates the extent to which the applicant has satisfied the USNRC's deterministic containment performance goals. The reviewer does this by comparing the applicant's evaluation of containment performance under loading conditions associated with beyond design basis accidents with the approach described in RG 1.212 (USNRC, 2010). If the applicant uses an alternate approach, the reviewer issues requests for additional information as necessary to confirm that the applicant has provided adequate technical justification for an alternate approach.

2.3 PRA for Non-Power Modes of Operation

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. The insights obtained from the shutdown PRA may no longer be valid if, in the future, licensee practices deviate dramatically from these assumptions. It is the COL applicant's responsibility to confirm the assumptions made at the DC stage and should capture any significant differences.

The staff reviews the applicant's assumptions related to equipment availability and compares them to technical specification (TS) requirements. Risk-significant equipment should be evaluated with respect to 10 CFR 50.36(c)(2)(ii)(D) (USNRC, 2012a) to determine whether additional TS requirements are needed. The staff may also review the results of sensitivity studies performed to demonstrate the risk benefit of equipment that is only controlled administratively.

The staff reviews the applicant's implementation of any applicable "expeditious actions" outlined in NRC Generic Letter (GL) 88-17 (USNRC, 1988). The staff needs to ensure that the applicant is meeting the applicable 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.

The staff reviews the applicant's implementation of applicable industry guidance for safety during outages provided in NUMARC-91-06 (NUMARC, 1991). 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. 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 Reactors (BWRs).

2.4 PRA-Based Seismic Margins Analysis (SMA)

The staff reviews the design-specific plant system and accident sequence analysis in the SMA in accordance with the acceptance criteria in the SRP. They verify 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.

2.5 Treatment of Internal Fires

The staff review considers the extent to which an applicant's assessment of the risk associated with internal fires conforms to the guidance in NUREG/CR-6850 (USNRC, 2005). This document describes an acceptable method for performing a fire PRA to support applications for a DC or COL. The staff may find that a DC applicant uses an approach to implementing the analysis tasks in NUREG/CR-6850 that is simpler than that suggested in NUREG/CR-6850. This can occur when the specifics of cable routings, ignition sources and target locations in each fire zone of the plant are not known at the time the DC application is submitted. Such an approach may be acceptable if conservative assumptions are used such that it is reasonable to conclude that the simplified results bound those expected with the more detailed approach described in NUREG/CR-6850 with respect to core damage frequency (CDF) and large release frequency (LRF). Examples of conservative assumptions that have been accepted by the staff in previous reviews are listed in SRP 19.0.

The staff review confirms that the fire risk analysis uses the same system and accident sequence models as the internal events evaluation.

The staff review confirms that the applicant has determined the appropriate internal event sequences based on the specific fire location and correctly modified these sequences to consider the effects of specific fires and include the possibility of fire propagation through potentially failed fire barriers.

The staff review confirms that the applicant reports the CDF and LRF derived from the fire PRA in the FSAR and provides a characterization of the dominant accident sequences and associated major contributors to CDF for each sequence in the FSAR.

2.6 Treatment of High Winds

The staff review evaluates the applicant's methodology and use of data for estimating initiating event frequencies and assumptions in its high winds risk assessment and verifies that the methodology is consistent with the state-of-the-art and that the assumptions are reasonable for estimating the CDF associated with high wind events that could damage the plant.

2.7 Treatment of Digital Instrumentation and Controls

Guidance for reviewing system models for digital I&C systems was developed and originally documented in NRC Interim Staff Guidance document DI&C-ISG-3 (USNRC, 2008b). This guidance has now been fully incorporated into SRP 19.0. This is detailed guidance that assumes the complete PRA documentation is available for review. Since the NRC does not require applicants to submit their PRA for review with their application, full use of this guidance is reserved for inspection or audit of an applicant's or licensee's PRA. However, some of the guidance is also useful in the review of the PRA information required to be submitted to the NRC as part of an application for a DC or COL.

3. Conclusions

The NRC staff is updating a number of SRP Sections that relate to PRA. Guidance for evaluating the technical adequacy of PRAs that support risk-informed changes to the licensing basis has been updated. A new risk-informed approach for the NRC staff review of small modular iPWR design applications is being developed that involves using information from the applicant's reliability assurance program to assess the significance of safety-related and non-safety related SSCs with respect to risk. SRP Section 19.0, which is used to guide the NRC's review of PRA information submitted by an applicant for a DC or COL, is being updated to include guidance issued on an interim basis since the last revision of SRP 19.0 and to reflect review procedures developed during recent reviews of applications to design and build new large light water reactors. The NRC staff believes that the improvements being made to this review guidance will lead to an increase in both the effectiveness and efficiency of its review of PRA information it receives from new nuclear power plant applicants and licensees.

4. References

ASME/ANS (2009), "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, Addendum A to ASME/ANS RA-S-2008," ASME/ANS RA-Sa-2009, American Society of Mechanical Engineers/American Nuclear Society, February 2009.

NEA (2012), "Applying Risk Insights in USNRC Reviews of Integral Pressurized Water Reactor Designs," Paper presented at Workshop on PSA for New and Advanced Reactors June 20-24, 2011, NEA/CSNI/R(R2012)2, Organization for Economic Co-operation and Development, Nuclear Energy Agency, June 19, 2012.

NEI (1991), "Guidelines for Industry Actions to Assess Shutdown Management," NUMARC 91-06, Nuclear Energy Institute, December 1991.

NEI (2008a), "Process for Performing Follow-On PRA Peer Reviews Using the ASME PRA Standard," Revision 2, NEI 05-04, Nuclear Energy Institute, November 2008.

NEI (2008b), "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines," Draft Version H, Revision 0, Nuclear Energy Institute, November 2008.

USNRC (1988), "Loss of Decay Heat Removal (Generic Letter 88-17) 10 CFR 50.54(f)," Letter from USNRC to All Holders of Operating Licenses or Construction Permits for Pressurized Water Reactors, October 17, 1988 (Available at www.nrc.gov).

USNRC (2005), "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," NUREG/CR-6850 & EPRI 1011989, U.S. Nuclear Regulatory Commission and Electric Power Research Institute, September 2005 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML043490032).

USNRC (2007a), "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," NUREG-0800, U.S. Nuclear Regulatory Commission, March 2007 (ADAMS Accession No. ML070810350).

USNRC (2007b), "Combined License Applications for Nuclear Power Plants (LWR Edition)," Regulatory Guide 1.206, U.S. Nuclear Regulatory Commission, June 2007 (ADAMS Accession No. ML071450387).

USNRC (2008a), "PRA Information to Support Design Certification and Combined License Applications," DC/COL-ISG-3, U.S. Nuclear Regulatory Commission, June 11, 2008 (ADAMS Accession No. ML081430087).

USNRC (2008b), "Review of New Reactor Digital Instrumentation and Control Probabilistic Risk Assessments", DI&C-ISG-3, U.S. Nuclear Regulatory Commission, August 11, 2008 (ADAMS Accession No. ML080570048).

USNRC (2009), "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-informed Activities," Revision 2, Regulatory Guide 1.200, U.S. Nuclear Regulatory Commission, March 2009 (ADAMS Accession No. ML090410014).

USNRC (2010a), "Seismic Margin Analysis for New Reactors Based on Probabilistic Risk Assessment", DC/COL-ISG-20, U.S. Nuclear Regulatory Commission, March 15, 2010 (ADAMS Accession No. ML100491233).

USNRC, (2010b), “Containment Structural Integrity Evaluation for Internal Pressure Loadings Above Design Basis Pressure”, USNRC Regulatory Guide 1.216, U.S. Nuclear Regulatory Commission, August 2010 (ADAMS Accession No. ML093200703).

USNRC, (2012a), “Domestic Licensing of Production and Utilization Facilities” United States Code of Federal Regulations, Title 10, Part 50, Revised as of January 1, 2012.

USNRC, (2012b), “Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants” United States Code of Federal Regulations, Title 10, Part 52, Revised as of January 1, 2012.

USNRC, (2012c), “Determining the Technical Adequacy of Probabilistic Risk Assessment for Risk-Informed License Amendment Requests after Initial Fuel Load”, NUREG-0800, Section 19.1, U.S. Nuclear Regulatory Commission, May 2012. (ADAMS Accession No. ML112990771)

USNRC, (2012d), “Licenses, Certifications and Approvals for Nuclear Power Plants” United States Code of Federal Regulations, Title 10, Part 52, Revised as of January 1, 2012.