U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

9.3.2 PROCESS AND POST-ACCIDENT SAMPLING SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of the functionality and capability of the process sampling systems (PSS)

Secondary - Organization responsible for the review of radiation protection

Organization responsible for the review of chemical engineering issues

Organization responsible for the review of thermal-hydraulic performance of safety systems in pressurized water reactors and boiling water reactors

Organization responsible for the review of systems associated with the balance of plant

Organization responsible for the review of instrumentation and control systems

Organization responsible for the review of emergency planning

I. AREAS OF REVIEW

This SRP section is applicable to construction permit (CP) and operating license (OL) applications submitted under 10 CFR Part 50 and design certification (DC) and combined license (COL) applications submitted under 10 CFR Part 52. The SRP was originally written for Part 50 license applications. For DC and COL applications submitted under 10 CFR Part 52,

Rev. 3 - [Month] 2007

USNRC STANDARD REVIEW PLAN

This Standard Review Plan, NUREG-0800, has been prepared to establish criteria that the U.S. Nuclear Regulatory Commission staff responsible for the review of applications to construct and operate nuclear power plants intends to use in evaluating whether an applicant/licensee meets the NRC's regulations. The Standard Review Plan is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required 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 an acceptable method of complying with the NRC regulations.

The standard review plan sections are numbered in accordance with corresponding sections in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)." Not all sections of Regulatory Guide 1.70 have a corresponding review plan section. The SRP sections applicable to a combined license application for a new light-water reactor (LWR) are based on Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)."

These documents are made available to the public as part of the NRC's policy to inform the nuclear industry and the general public of regulatory procedures and policies. Individual sections of NUREG-0800 will be revised periodically, as appropriate, to accommodate comments and to reflect new information and experience. Comments may be submitted electronically by email to NRR_SRP@nrc.gov.

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the level of information reviewed should be consistent with that of a final safety analysis report (FSAR) information submitted in an OL application. However, verification that the as-built facility conforms with the approved design is performed through the inspections, tests, analyses, and acceptance criteria (ITAAC) process.

The specific areas of review are as follows:

1. The design objectives and design criteria for the PSS are reviewed at the CP stage. During the OL stage, the review consists of confirming the design accepted at the CP stage and evaluating the adequacy of the applicant's Technical Specifications in these areas. The review includes identification of the process streams to be sampled and the parameters to be determined through sampling (e.g., gross beta-gamma concentration, boric acid concentration).

2. The system descriptions for the PSS are reviewed at the OL stage. The review includes (a) piping and instrumentation diagrams, (b) provisions for obtaining representative samples, (c) the location of sampling points and sample stations, and (d) provisions for purging sampling lines.

3. The seismic design and quality group classifications of piping and equipment, and the bases for the classifications chosen are reviewed at the CP stage. At the OL stage, the review includes design and expected temperatures and pressures and the materials of construction of system components.

4. The isolation provisions for the system and the means provided to limit radioactive releases by minimizing reactor coolant losses are reviewed at the CP stage.

5. The PSS operational procedures for sampling the reactor coolant and containment atmosphere to determine the capability to promptly obtain samples for chemical and radiochemical analyses are reviewed.

6. The post-accident sampling system (PASS) is not mandatory. In some designs, such as the certified design for the AP1000 plant, approved by the NRC, PASS has been eliminated (NRC letter to Combustion Engineering (CE) Owners Group dated May 16, 2000). Although the PSS in the designs such as the AP1000 plant does not have a specific post-accident sampling capability, their design allows for the collection and analysis of highly radioactive samples for boron, containment pH, and containment atmosphere for hydrogen and other fission products. Subject to meeting the following conditions, the PSS could collect highly radioactive samples. In lieu of the PASS, the following actions are required to qualify process sampling for taking radioactive samples without having a specific post-accident sampling capability:

    A. Establish the capability for classifying a fuel damage event at the Alert level threshold

    B. Develop contingency plans for obtaining highly radioactive samples of the reactor coolant, containment sump, and containment atmosphere

C. Determine for its own plant(s) that no decrease in the effectiveness of emergency plans will result from not having post-accident sampling system capability

D. Establish the capability to sample and analyze hydrogen in the containment atmosphere (recommended)

E. Maintain offsite capability to monitor radioactivity, including radioactive iodines

7. Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). For design certification (DC) and combined license (COL) reviews, the staff reviews the applicant's proposed ITAAC associated with the structures, systems, and components (SSCs) related to this SRP section in accordance with SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria." The staff recognizes that the review of ITAAC cannot be completed until after the rest of this portion of the application has been reviewed against acceptance criteria contained in this SRP section. Furthermore, the staff reviews the ITAAC to ensure that all SSCs in this area of review are identified and addressed as appropriate in accordance with SRP Section 14.3.

8. COL Action Items and Certification Requirements and Restrictions. 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

Other SRP sections interface with this section as follows:

1. Sections 3.2.1 and 3.2.2: review of the acceptability of the seismic and quality group classifications for system components.

2. Sections 3.3.1, 3.3.2, 3.4.2, 3.5.3, 3.7.1 through 3.7.4, 3.8.4, and 3.8.5: review of the acceptability of the design analyses, procedures, and criteria used to establish the ability of Seismic Category I structures housing the system and supporting systems to withstand the effects of natural phenomena such as the safe shutdown earthquake, the probable maximum flood, and tornadoes and tornado missiles.

3. Section 3.4.1: review of internal flood protection for piping, tank, and vessel failures, operation of the fire protection system, and water leakage into safety-related areas.

4. Sections 3.5.1.1 through 3.5.1.6, and 3.5.2: review of the protection against internally and externally generated missiles.

5. Section 3.6.1: review of the design with respect to the effects of externally or internally generated missiles, pipe whip, and jet impingement forces associated with postulated pipe breaks in high-energy fluid systems or leakage cracks in moderate-energy fluid systems.

6. Section 3.6.2: review of possible break locations in high- and moderate-energy systems during normal plant operation and the review of dynamic effects (e.g. pipe whip, jet impingement) of pipe breaks.

7. Section 3.10: review of the seismic qualification of category I instrumentation.

8. Section 3.11: verification that those valves that are inaccessible during an accident are environmentally qualified to ensure operability under accident conditions.

9. Section 6.2.1.3 - 6.2.1.4: review of the environmental effects of piping failures inside containment.

10. Section 6.2.4: verification that remotely operated containment isolation valves in the PSS are designed to close on a containment isolation signal or safety injection signal.

11. Section 6.6: verification that in-service inspection requirements are met for system components.

12. Section 7.3: verification that override capability exists for containment isolation valves that will be used for process sampling of the reactor coolant, containment sump water, and containment atmosphere without clearing the containment isolation signal or safety injection signal.

13. Section 7.5: review of the types of instruments needed for flood protection, including the adequacy of detectors and alarms necessary to detect rising water levels within structures, and the review of the consequences of flooding on other safety-related instrumentation and electrical equipment.

14. Section 8.3.1: verification that power supplies are available to all remotely operated containment isolation valves in the sampling system, after detection of an accident that requires containment isolation and assuming a concurrent loss of offsite power.

15. Section 11.2: review of liquid radwaste processing.

16. Section 11.3: review of the ventilation systems that may operate during sampling of radioactive materials.

17. Section 11.5: review of the sampling and monitoring systems for radwaste processing systems.

18. Sections 12.3-12.4: review of potential personal radiation exposure during process sampling of radioactive material.

19. Section 13.3: review of contingency plans and procedures for post-accident sampling.

20. Section 16.0: review of Technical Specifications.

21. Chapter 17: review of quality assurance.

The specific acceptance criteria and review procedures are contained in the referenced SRP sections.

II. ACCEPTANCE CRITERIA

Requirements

Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:

1. 10 CFR 20.1101(b), as it relates to providing engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public as low as is reasonably achievable (ALARA)

2. General Design Criterion (GDC) 1, found in Appendix A to 10 CFR Part 50, as it relates to the design of the PSS and components in accordance with standards commensurate with the importance of their safety functions

3. GDC 2, as it relates to the ability of the PSS to withstand the effects of natural phenomena

4. GDC 13, as it relates to monitoring variables that can affect the fission process, the integrity of the reactor core, and the reactor coolant pressure boundary

5. GDC 14, as it relates to assuring the integrity of the reactor coolant pressure boundary by sampling for chemical species that can affect the reactor coolant pressure boundary

6. GDC 26, as it relates to reliably controlling the rate of reactivity changes by sampling boron concentration

7. GDC 41, as it relates to reducing the concentration and quality of fission products released to the environment following postulated accidents by sampling the chemical additive tank for chemical additive concentrations to ensure an adequate supply of chemicals for meeting the material compatibility requirements and the elemental iodine removal requirements of the containment spray and recirculation solutions following a postulated accident

8. GDC 60, as it relates to the capability of the PSS to control the release of radioactive materials to the environment

9. GDC 63, as it relates to detecting conditions that may result in excessive radiation levels in the fuel storage and radioactive waste systems

10. GDC 64, as it relates to monitoring the containment atmosphere and plant environs for radioactivity

11. Three Mile Island (TMI) Action Plan Item III.D.1.1 in NUREG-0737, as it relates to the provisions for a leakage control program to minimize the leakage from those portions of the PSS outside of the containment that contain or may contain radioactive material following an accident. 10 CFR 50.34(f)(2)(xxvi) provides equivalent requirements for those applicants subject to 10 CFR 50.34(f).

12. 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations;

13. 10 CFR 52.80(a), which requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license, the provisions of the Atomic Energy Act, and the NRC's regulations.

SRP Acceptance Criteria

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 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.

1. The applicant's design is such that the PSS has the capability to sample all normal process systems and principal components, including provisions for obtaining samples from at least the points indicated below. The guidelines of Regulatory Guide (RG) 1.21, Position C.2, the Electric Power Research Institute (EPRI) BWR Water Chemistry Guidelines, and the Electric Power Research Institute (EPRI) PWR Water Chemistry Guidelines are used to meet the requirements of the relevant GDC.

For a Pressurized Water Reactor (PWR)	GDC
Reactor coolant (e.g., letdown system)	13, 14, 26, 64, 60
Refueling (borated) water storage tank	13, 26
ECCS core flooding tank	13
Boric acid mix tank	13, 26
Boron injection tank	13
Chemical additive tank	13, 14, 41

Spent fuel pool	63, 60
Secondary coolant (e.g., condensate hotwell)	13, 14
Pressurizer tank	64, 60
Steam generator blowdown (if applicable)	14, 64, 60
Secondary coolant condensate treatment waste	64, 60
Sumps inside containment	64, 60
Containment atmosphere	64, 60
Gaseous radwaste storage tanks	63, 64, 60
<u>For a Boiling Water Reactor (BWR)</u>	<u>GDC</u>
Main condenser evacuation system offgas, and charcoal delay or decay beds	64, 60
Reactor coolant (inlet and outlet of reactor water cleanup system)	13, 14, 64, 60
Standby liquid control system tank	13, 26
Sumps inside containment	64, 60
Spent fuel pool	63, 60
Drywell atmosphere (Mark I & II)	64, 60
Inlet and outlet of gaseous radwaste storage tank	63, 64, 60
Inlet and outlet of condensate polishing system	13, 14

SRP Section 11.5 gives other sample points that may be included in the PSS but do not require remote sampling.

2. The plant Technical Specifications include the required analysis and frequencies.

3. The following guidelines should be used to determine the acceptability of the PSS functional design:

   A. Provisions should be made to ensure representative samples from liquid process streams and tanks. For tanks, provisions should be made to sample the bulk volume of the tank and to avoid sampling from low points or from potential sediment traps. For process stream samples, sample points should be located in turbulent flow zones. The guidelines of Regulatory Position C.6 in RG 1.21 are followed to meet these criteria.

B. Provisions should be made to ensure representative samples from gaseous process streams and tanks in accordance with American National Standards Institute/Health Physics Society (ANSI/HPS) Standard N13.1-1999. The guidelines of Regulatory Position C.6 in RG 1.21 are followed to meet this criterion.

C. Provisions should be made for purging sampling lines and for reducing plateout in sample lines (e.g., heat tracing). The guidelines of Regulatory Position C.7 in RG 1.21 are followed to meet this criterion.

D. Provisions should be made to purge and drain sample streams back to the system of origin or to an appropriate waste treatment system in accordance with the requirements of 10 CFR 20.1101(b) to keep radiation exposures at ALARA levels. The guidelines of Regulatory Positions 2.d.(2), 2.f.(3), and 2.f.(8) in RG 8.8 are followed to meet this criterion.

E. Isolation valves should fail in the closed position, in accordance with the requirements of GDC 60 to control the release of radioactive materials to the environment.

F. Passive flow restrictions to limit reactor coolant loss from a rupture of the sample line should be provided in accordance with the requirements of 10 CFR 20.1101(b) to keep radiation exposures to ALARA levels and the requirements of GDC 60 to control the release of radioactive materials to the environment. The guidelines of Regulatory Position 2.i.(6) in RG 8.8 should be followed to meet this criterion. Redundant environmentally qualified, remotely operated isolation valves may replace passive flow restrictions in the sample lines to limit potential leakage. The automatic containment isolation valves should close on containment isolation signals or safety injection signals.

4. To meet the requirements of GDCs 1 and 2, the applicant's seismic design and quality group classification of sampling lines, components, and instruments for the PSS should conform to the classification of the system to which each sampling line and component is connected (e.g., a sampling line connected to a Quality Group A and seismic Category I system should be designed to Quality Group A and seismic Category I classification), in accordance with Regulatory Positions C.1, C.2, and C.3 in RG 1.26; Regulatory Positions C.1, C.2, C.3, and C.4 in RG 1.29, and the guidelines of RG 1.97. Components and piping downstream of the second isolation valve may be designed to Quality Group D and nonseismic Category I requirements, in accordance with Regulatory Position C.3 in RG 1.26.

Technical Rationale

The technical rationale for application of these acceptance criteria to the areas of review addressed by this SRP section is discussed in the following paragraphs:

1. 10 CFR 20.1101(b) requires that licensees use, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve doses that are ALARA. The radiation protection community has recognized that it is prudent to avoid unnecessary exposure to radiation and to maintain doses at ALARA

levels based on the assumption that a nonthreshold linear relationship exists between dose and biological effects that is independent of the dose rate. The objective of the ALARA requirement for the PSS is to ensure that licensees make every reasonable effort in planning, design, and operation of the system to maintain exposures to radiation as far below the limits of 10 CFR Part 20 as is reasonably achievable. Appropriate station layout and design features should be provided to reduce the potential doses to personnel who must operate, service, or inspect the station PSSs. The safety benefit of implementing radiation protection goals for the PSSs is to reduce doses wherever and whenever reasonably achievable, thereby reducing the risk that is assumed (for radiation protection purposes) to be proportional to the dose.

2. GDC 1 requires that SSCs important to safety be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety functions to be performed. The PSS is important to safety in that (a) through connections to systems, such as the reactor coolant system, that are designed and classified in accordance with recognized quality standards, their failure could adversely affect the integrity of these systems, (b) during normal operation, the PSS provides information that allows the operator to assess the integrity of the fuel cladding and to recognize conditions that could jeopardize the integrity of the reactor coolant pressure boundary, (c) although the PSS does not have a specific post-accident sampling capability, its design allows for the collection and analysis of highly radioactive samples that may be present following an accident. Meeting the requirements of GDC 1 ensures that the PSSs will be designed, fabricated, erected, and tested to generally accepted and recognized codes and standards that are sufficient to ensure a quality system in keeping with the required safety functions.

3. GDC 2 requires that SSCs important to safety be designed to withstand the effects of natural phenomena without the loss of the capability to perform their safety functions. The PSS connects to systems, such as the reactor coolant pressure boundary, that are designed to seismic Category I requirements. Those portions of the PSSs or components that form interfaces between seismic Category I and nonseismic Category I features should be designed to seismic Category I requirements. Meeting the requirements of GDC 2 for those portions of the PSS that interface with seismic Category I systems will enhance plant safety by ensuring the integrity of seismic Category I systems, such as the reactor coolant pressure boundary, during the design-basis seismic event.

4. GDC 13 requires that instrumentation be provided to monitor variables and systems to ensure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, and the reactor coolant pressure boundary. The PSS is relied upon to provide water and gaseous samples from the reactor coolant system and associated auxiliary systems during all normal modes of operation. Satisfying the requirements of GDC 13 for the PSS ensures that important information is provided for evaluating whether safety systems and other systems important to safety are performing their intended safety functions (i.e., reactivity control, fuel cladding integrity, maintaining reactor coolant system integrity, and maintaining containment integrity).

5. GDC 14 requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage,

rapidly propagating failure, and gross rupture. The PSS is relied upon during normal operating, transient, and postulated accident conditions to provide primary and secondary water chemistry data. Verification that key chemistry parameters, such as chloride, hydrogen, and oxygen concentrations, are within prescribed limits and that impurities are properly controlled provides assurance that the many mechanisms for corrosive attack will be mitigated and will not adversely affect the reactor coolant pressure boundary. Minimizing the potential for corrosive chemical attack increases plant safety by decreasing the probability that the reactor coolant pressure boundary will be compromised because of degradation from corrosive chemical attack.

6. GDC 26 establishes requirements regarding the reliable control of the rate of reactivity changes. PWRs use the sampling systems to verify the boron concentration, necessary for the control of the rate of reactivity changes, by sampling the reactor coolant, the boric acid storage tanks, and the refueling water storage tank. BWRs use the sampling systems to verify the boron concentration in the standby liquid control system, which may be used to maintain the reactor subcritical under cold conditions in the event that the control rod system is inoperable. Meeting the requirements of GDC 26, as it relates to the sampling systems, ensures sampling and evaluation of boron concentrations used to control the rate of reactivity changes through the injection of borated water into the reactor coolant system, enhancing plant safety by (a) meeting the combined reactivity control system redundancy and capability requirements and (b) not exceeding acceptable fuel design limits.

7. GDC 41 requires that systems to control fission products, hydrogen, oxygen, and other substances that may be released into the reactor containment be provided as necessary to reduce the concentration and quality of fission products released to the environment following postulated accidents. This requirement is met, in part, by using the PSS to determine the chemical concentration in the containment spray chemical additive tank. Determination that the chemical additive concentration is within limits ensures that a sufficient chemical supply is available during postulated accidents to meet elemental iodine removal requirements and material compatibility requirements. In addition, the PSS is relied upon to sample and evaluate the conditions inside containment resulting from the metal-water reactions, radiolysis, and corrosion following a postulated accident. The information obtained from the process samples is used to verify the safety functions of engineered safety features, including the atmospheric cleanup systems and the containment spray system, to mitigate the consequences of postulated accidents by removing from the containment atmosphere radioactive material that may be released in an accident. Meeting the requirements of GDC 41, as it relates to the PSS, ensures that sufficient sample information can be provided to verify the safety function of engineered safety features to reduce the concentration and quality of fission products that may be released to the environment following postulated accidents.

8. GDC 60 requires that means be provided to control the release of radioactive materials to the environment. The PSS contains or may contain radioactive material that must be properly controlled. Examples of the controls used to prevent release of radioactive material from the PSS to the environment include (a) redundant automatic isolation valves that will close on a containment isolation signal or safety injection signal and will fail in the closed position, (b) purging and draining the sample lines back to the system being sampled or to the radwaste treatment system, and (c) passive flow restrictions to limit reactor coolant loss from a rupture of a sample line. Meeting the requirements of

GDC 60, as it relates to the PSS, enhances safety by providing a means in the design to control the release of radioactive material to the environment. Application of GDC 60 provides reasonable assurance that the PSS is designed, constructed, installed, and operated on a level commensurate with the need to protect the health and safety of the public and plant operating personnel.

9. GDC 63 requires that systems be provided to monitor the fuel storage and radioactive waste systems to detect conditions that may result in excessive radiation levels. The PSS, through sampling of the spent fuel pool water and the gaseous radwaste storage tanks, should be capable of detecting conditions that could result in excessive radiation levels and excessive personnel exposure. The PSS provides information necessary for the control of water chemistry to maintain the material properties of the fuel assembly cladding and structural members and cooling systems of the spent fuel pool. The PSS, through the ability to sample the gaseous radwaste storage tanks or offgas charcoal delay or decay beds, is also used to detect abnormal levels of radioactivity in the radwaste processing facilities. Meeting the requirements of GDC 63, as it relates to the PSS, ensures that sampling methods are available to monitor the spent fuel pool and the gaseous radwaste storage tank radioactivity levels such that personnel exposures are maintained at ALARA levels.

10. GDC 64 requires that means be available for monitoring the containment atmosphere, spaces containing components used for recirculation after a loss-of-coolant accident, effluent discharge paths, and the plant environs for radioactivity that may be released during normal operations, anticipated operational occurrences, and postulated accidents. The PSS design provides the means for monitoring for radioactivity that may be released during normal operations, anticipated operational occurrences, and postulated accidents. The PSS provides information to indicate the potential for being breached or the actual breach of the barriers to fission product release (i.e., fuel cladding, primary coolant pressure boundary, and containment). The PSS provides information regarding the release of radioactive materials, which allows for early indication of the need to initiate other protective actions. Meeting the requirements of GDC 64, as it relates to the PSS, ensures that a means is provided to monitor the release of radioactive materials, giving the plant operator the indications needed to initiate actions when necessary to protect the health and safety of plant personnel and the general public.

11. TMI Action Plan Item III.D.1.1 in NUREG-0737, or 10 CFR 50.34(f)(2)(xxvi) for those applicants subject to 10 CFR 50.34(f), requires a program and provisions for leakage control and detection for systems outside containment that contain (or might contain) source term radioactive materials following an accident. The PSS provides a means for sampling the reactor coolant systems and containment atmosphere to provide the information necessary to assess and control the plant under accident conditions. Because this system draw samples directly from the reactor coolant systems or from the containment atmosphere, it has the potential for containing source term radioactive material during the course of an accident. To prevent unnecessarily high exposures to workers and the public and to maintain control and use of the systems during an accident, a program should be implemented to minimize leakage from this system to as low as practical levels. Meeting the guidance of TMI Action Plan Item III.D.1.1 in

NUREG-0737, or 10 CFR 50.34(f)(2)(xxvi) for those applicants subject to 10 CFR 50.34(f), as they relate to the PSS, enhances safety by minimizing the leakage from these systems and thereby minimizing the potential exposures to workers and the public, and by providing reasonable assurance that excessive leakage will not prevent the use of the systems under accident conditions.

III. REVIEW PROCEDURES

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. For deviations from these acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives provide an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

1. In the review of the PSS, the primary review organization and the organization responsible for the review of chemical engineering issues review the process sampling points as shown on piping and instrumentation diagrams, and compare the list of process sampling points contained in the safety analysis report with the sampling points identified in Item 1 of the SRP Acceptance Criteria, above, to ensure that the required process sampling points have been provided.

2. The primary review organization and the organization responsible for the review of radiation protection compare the capability of the system to obtain representative samples of process fluids and the locations of sampling points with the guidelines for obtaining representative samples of fluids contained in Regulatory Position C.6 of RG 1.21 and with the principles for obtaining representative samples of gases contained in ANSI/HPS N13.1-1999.

3. The primary review organization and the organization responsible for the review of radiation protection verify that provisions made for purging sampling lines and for reducing plateout in sample lines (e.g., heat tracing) conform with the guidelines of Regulatory Position C.7 in RG 1.21.

4. The primary review organization, the organization responsible for the review of systems associated with the balance of plant, and the organization responsible for the review of radiation protection verify that provisions have been made to purge and drain sample streams back to the system of origin or to an appropriate waste treatment system to keep radiation exposures at ALARA levels, and that these provisions conform with the guidelines of Regulatory Positions 2.d.(2), 2.f.(3), and 2.f.(8) in RG 8.8.

5. The primary review organization, with the applicable secondary review organization, as necessary, verifies that isolation valves fail in the closed position to control the release of radioactive materials to the environment.

6. The primary review organization, the applicable secondary review organization, as necessary, and the organization responsible for the review of radiation protection verify that passive flow restrictions to limit reactor coolant loss from a rupture of the sample line are provided to keep radiation exposures to ALARA levels and to control the release

of radioactive materials to the environment. The guidelines of Regulatory Position 2.i.(6) in RG 8.8 should be followed to meet this criterion. Redundant environmentally qualified, remotely operated isolation valves may replace passive flow restrictions in the sample lines to limit potential leakage. The automatic containment isolation valves should close on containment isolation signals or safety injection signals.

7. The primary review organization compares the seismic design and quality group classifications of the PSS to the classifications of the fluid systems to which the sampling system is connected and confirms that the PSS satisfies the acceptance criteria of Sections 3.2.1 and 3.2.2. The organization responsible for the review of instrumentation and control systems compares the seismic design and quality group classifications of the instruments for the PSS to the classifications of the fluid systems to which the instruments of the sampling system is connected and confirms that the instrumentation satisfies the acceptance criteria of Section 3.10.

8. The organization responsible for the review of chemical engineering issues evaluates the Technical Specifications for process sampling to determine that their content and intent agree with the requirements developed as a result of the staff's review.

9. The primary review organization and the organizations responsible for the review of thermal-hydraulic performance of safety systems in pressurized and boiling water reactors verify that provisions have been made to limit the potential for reactor coolant loss from the rupture of a sample line and provide the estimates of reactor coolant system fluid losses that would result from sample line rupture.

10. In lieu of the PASS, the organizations responsible for the review of emergency planning and for the review of chemical engineering issues should examine the following actions required to qualify process sampling for taking radioactive samples without having a specific post-accident sampling capability:

    A. Establish the capability for classifying a fuel damage event at the Alert level threshold

    B. Develop contingency plans for obtaining highly radioactive samples of the reactor coolant, containment sump, and containment atmosphere

    C. Determine for its own plant(s) that no decrease in the effectiveness of emergency plans will result from not having post-accident system capability

    D. Establish the capability to sample and analyze hydrogen in the containment atmosphere (recommended)

    E. Maintain offsite capability to monitor radioactivity, including radioactive iodines

11. The primary review organization and the organization responsible for the review of radiation protection verify that a leakage control program includes those portions of the PSS located outside of containment that contain or may contain radioactive material following an accident. The leakage control program should include periodic leak testing and measures to minimize leakage from the PSS.

12. The organization responsible for the review of instrumentation and control systems reviews the instrumentation provided to ensure that information is provided for evaluating whether safety systems and other systems important to safety are performing their intended functions (i.e., reactivity control, fuel cladding integrity, maintaining reactor coolant system integrity, and maintaining containment integrity).

13. The organization responsible for the review of instrumentation and control systems reviews the sampling system related instrumentation that is recommended by RG 1.97 to ensure that this instrumentation meets the guidance of RG 1.97.

14. For review of a DC application, the reviewer should follow the above procedures to verify that the design, including requirements and restrictions (e.g., interface requirements and site parameters), set forth in the final safety analysis report (FSAR) meets the acceptance criteria. DCs have referred to the FSAR as the design control document (DCD). The reviewer should also consider the appropriateness of identified COL action items. The reviewer may identify additional COL action items; however, to ensure these COL action items are addressed during a COL application, they should be added to the DC FSAR.

    For review of a COL application, the scope of the review is dependent on whether the COL applicant references a DC, an early site permit (ESP) or other NRC approvals (e.g., manufacturing license, site suitability report or topical report).

    For review of both DC and COL applications, SRP Section 14.3 should be followed for the review of ITAAC. The review of ITAAC cannot be completed until after the completion of this section.

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the review and calculations (if applicable) support conclusions of the following type to be included in the staff's safety evaluation report. The reviewer also states the bases for those conclusions.

> The process sampling system includes piping, valves, heat exchangers, and other components from the point of sample withdrawal from a fluid system up to the analyzing station, sampling station, or local sampling point. Our review included the provisions proposed to sample all principal fluid process streams associated with plant operation and the applicant's proposed PSS design. The review has included descriptive information for the process sampling system and the location of sampling points, as shown on piping and instrumentation diagrams. The basis for acceptance in our review has been conformance of the applicant's design for the process sampling system to applicable regulations, guides, and industry standards.
>
> The staff concludes that the design of the process sampling system is acceptable and that the process sampling system meets the relevant requirements of 10 CFR 20.1101(b) and GDC 1, 2, 13, 14, 26, 41 (for PWR only), 60, 63, and 64. The following paragraphs discuss the bases for this conclusion for each of the requirements, as applicable to PWRs, BWRs, or both.

For some designs, which may not contain the listed systems or locations for sampling, sampling of systems or locations found to be equivalent is acceptable in meeting the requirements of this SRP section.

For PWR

The staff has determined that the proposed process sampling system meets (1) the requirements of GDC 13 to monitor variables that can affect the fission process for normal operation, anticipated operational occurrences, and accident conditions by sampling the reactor coolant, the ECCS core flooding tank, the refueling water storage tank, the boric acid mix tank, and the boron injection tank for boron concentrations; (2) the requirements of GDC 13 and 14 to monitor variables that can affect the reactor coolant pressure boundary and to assure a low probability of abnormal leakage, rapidly propagating failure, and gross rupture, respectively, by sampling the reactor coolant and the secondary coolant for chemical impurities that can affect the reactor coolant pressure boundary; (3) the requirements of GDC 26 to control the rate of reactivity changes by sampling the reactor coolant, the refueling water storage tank, and the boric acid mix tank for boron concentration; (4) the requirements of GDC 13, 14, and 41 to monitor variables that can affect the integrity of the reactor core and reactor coolant pressure boundary and to reduce the concentration and quality of fission products released to the environment following postulated accidents, respectively, by sampling the chemical additive tank for chemical additive concentrations to ensure an adequate supply of chemicals for meeting the material compatibility requirements and the elemental iodine removal requirements of the containment spray and recirculation solutions following a postulated accident; and (5) the requirements of GDC 64 to monitor for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents by sampling the reactor coolant, the pressurizer tank, the steam generator blowdown, the secondary coolant condensate treatment waste, the sump inside containment, the containment atmosphere, and the gaseous radwaste storage tank for radioactivity.

For BWR

The staff has determined that the proposed process sampling system meets (1) the requirements of GDC 13 and 14 to monitor variables that can affect the reactor coolant pressure boundary and to assure a low probability of abnormal leakage, rapidly propagating failure, and gross rupture, respectively, by sampling the reactor coolant and the condensate for chemical impurities that can affect the reactor coolant pressure boundary; (2) the requirements of GDC 26 to maintain the reactor core subcritical under cold conditions in the event that control rod system is inoperable by sampling the standby liquid control system tank for boron concentration; and (3) the requirements of GDC 64 to monitor for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents by sampling the reactor coolant, the main condenser evacuation system offgas, the sump inside containment, the drywell atmosphere, and the offgas charcoal delay or decay beds for radioactivity.

For both PWR and BWR

The staff has further determined that the proposed process sampling system meets (1) the requirements of 10 CFR 20.1101(b) to keep radiation exposures as low as is reasonably achievable and of GDC 60 to control the release of radioactive materials to the environment by purging and draining sample streams back to the system of origin or to an appropriate radwaste treatment system, and by providing either redundant isolation valves that fail in the closed position or passive flow restrictions in the sampling lines; (2) the requirements of GDC 63 to detect conditions that may result in excessive radiation levels in fuel storage and radioactive waste systems by sampling the spent fuel pool water and the gaseous radwaste storage tank for radioactivity; and (3) the requirements of 10 CFR 50.34(f)(2)(xxvi) and related clarifications of Item III.D.1.1 in NUREG-0737 by inclusion of applicable portions of the systems in a leakage control program that contains periodic leak testing and measures to minimize the leakage from the systems.

The staff also has determined that the proposed process sampling system meets the quality standards requirements of GDC 1 and the seismic requirements of GDC 2 by designing the sampling lines and components of the process sampling system to conform to the classification of the system to which each sampling line and component is connected, in accordance with Regulatory Positions C.1, C.2, and C.3 of RG 1.26, Regulatory Positions C.1, C.2, C.3, and C.4 of RG 1.29, and the guidelines of RG 1.97.

For DC and COL reviews, the findings will also summarize the staff's evaluation of requirements and restrictions (e.g., interface requirements and site parameters) and COL action items relevant to this SRP section.

In addition, to the extent that the review is not discussed in other SER sections, the findings will summarize the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable.

V. IMPLEMENTATION

The staff will use this SRP section in performing safety evaluations of DC applications and license applications submitted by applicants pursuant to 10 CFR Part 50 or 10 CFR Part 52. Except when the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the staff will use the method described herein to evaluate conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications docketed six months or more after the date of issuance of this SRP section, unless superseded by a later revision.

VI. REFERENCES

1. 10 CFR 20.1101(b), "Radiation Protection Programs."

2. 10 CFR 50.34(f)(2)(xxvi), "Additional TMI-Related Requirements."

3. 10 CFR Part 50, Appendix A, General Design Criterion 1, "Quality Standards and Records."

4. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

5. 10 CFR Part 50, Appendix A, General Design Criterion 13, "Instrumentation and Control."

6. 10 CFR Part 50, Appendix A, General Design Criterion 14, "Reactor Coolant Pressure Boundary."

7. 10 CFR Part 50, Appendix A, General Design Criterion 26, "Reactivity Control System Redundancy and Capability."

8. 10 CFR Part 50, Appendix A, General Design Criterion 41, "Containment Atmosphere Cleanup."

9. 10 CFR Part 50, Appendix A, General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment."

10. 10 CFR Part 50, Appendix A, General Design Criterion 63, "Monitoring Fuel and Waste Storage."

11. 10 CFR Part 50, Appendix A, General Design Criterion 64, "Monitoring Radioactivity Releases."

12. NUREG-0737, "Clarification of TMI Action Plan Requirements."

13. Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants."

14. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants."

15. Regulatory Guide 1.29, "Seismic Design Classification."

16. Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident."

17. Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable."

18. NRC Letter to CE Owners Group, "Acceptance for Referencing of the Combustion Engineering Joint Applications Report, CE NPSD-1157, Revision 1, 'Technical Justification for the Elimination of the Post-Accident Sampling System from the Plant Design and Licensing Bases for CEOG Utilities,'" May 16, 2000 (ADAMS Accession No. ML003715250).

19. NRC Letter to Westinghouse Owners Group, "Safety Evaluation Related to Topical Report WCAP-14986, Revision 1, 'Westinghouse Owners Group Post Accident Sampling System Requirements,'" June 14, 2000 (ADAMS Accession No. ML003723268).

20. ANSI/HPS N13.1-1999, "Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities."

21. NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard, Vol. 1, issued September 2004.

22. "BWR Water Chemistry Guidelines - Revision 2." EPRI Report TR-103515-Revision 2, Electric Power Research Institute, February 2000.

23. "Pressurized Water Reactor Secondary Water Chemistry Guidelines - Revision 6," EPRI Report TR-1008224-Revision 6, Electric Power Research Institute, December 2004.

24. "Pressurized Water Reactor Primary Water Chemistry Guidelines: Volume 1, Revision 5," EPRI Report TR-1002884-Revision 5, Electric Power Research Institute, October 2003.

25. "PWR Primary Water Chemistry Guidelines: Vol. 2: Revision 4 Volume 2," EPRI Report TR-105714-V2R4, Electric Power Research Institute, March 1999.

PAPERWORK REDUCTION ACT STATEMENT

The information collections contained in the Standard Review Plan are covered by the requirements of 10 CFR Part 50 and 10 CFR Part 52, and were approved by the Office of Management and Budget, approval number 3150-0011 and 3150-0151.

PUBLIC PROTECTION NOTIFICATION

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

SRP Section 9.3.2
Description of Changes

This SRP section affirms the technical accuracy and adequacy of the guidance previously provided in Draft Revision 3, dated June 1996 of this SRP. See ADAMS accession number ML052070526.

In addition this SRP section was administratively updated in accordance with NRR Office Instruction, LIC-200, Revision 1, "Standard Review Plan (SRP) Process." The revision also adds standard paragraphs to extend application of the updated SRP section to prospective submittals by applicants pursuant to 10 CFR Part 52.

The technical changes are incorporated in Revision 3, dated [Month] 2007:

Review Responsibilities - Reflects changes in review branches resulting from reorganization and branch consolidation. Change is reflected throughout the SRP.

I. AREAS OF REVIEW

A. Eliminated reference to the post-accident sampling system (PASS).

B. Added that the PASS is not mandatory and operational procedures for taking chemical and radiochemical samples in lieu of the PASS .

C. Added required actions to qualify process sampling for radioactive samples without a specific post-accident sampling capability

D. Added ITAAC as an area of review.

E. Added review interfaces for Sections 3.2.1, 3.2.2, 3.3.1, 3.3.2, 3.4.2, 3.5.3, 3.7.1 through 3.7.4, 3.8.4, 3.8.5, 3.4.1, 3.5.1.1 through 3.5.1.6, 3.5.2, 3.6.2, 3.10, 6.2, 6.6, 7.5, 11.2, and 16.0, and Chapter 17.

II. ACCEPTANCE CRITERIA

A. Eliminated reference to the post-accident sampling system.

B. Added ITAAC as an area of review.

C. Eliminated references to the documents containing guidelines for the post-accident sampling system.

D. Added the capability of the PSS to collect and analyze highly radioactive samples.

E. Eliminated references to the documents providing guidance for the post-accident sampling system (TMI Action Plan Item II.B.3 in NUREG-0737).

F. Added reference to 10 CFR 50.34(f)(2)(xxvi) requirements for those applicants that are subject to 10 CFR 50.34(f).

G. Added EPRI PWR Water Chemistry Guidelines.

H. Added GDC 60 to the table in the SRP Acceptance Criteria Subsection.

I. Modified the Technical Rationale for GDC 63 to include offgas charcoal delay or decay beds.

J. Modified the Technical Rationale for TMI Action Plan Item III.D.1.1 by adding reference to 10 CFR 50.34(f)(2)(xxvi).

K. Updated the reference of ANSI N13.1 to ANSI/HPS N13.1-1999.

III. REVIEW PROCEDURES

A. Eliminated reference to the post-accident sampling system.

B. Eliminated the requirement for compliance of the post-accident sampling system and its operational procedures with NUREG-0737 and Regulatory Guide 1.97.

C. Eliminated certain exemptions from the post-accident sampling requirements included in the advanced BWR and System 80+ DCs.

D. Added the review of provisions for purging and draining sample lines, and for limiting the release of radiactive material through the provision of isolation valves or passive flow restrictors.

E. Added the PSS in the DC of the AP1000 plant.

F. Added actions taken to qualify the PSS for taking radioactive samples without having a specific post-accident sampling capability.

G. Eliminated the need for establishing and implementing a post-accident sampling administrative program for obtaining and analyzing the post-accident samples.

H. Added additional review procedures on instrumentation and controls.

IV. EVALUATION FINDINGS

A. Eliminated reference to the post-accident sampling system.

B. Eliminated descriptions of the functions performed by the post-accident sampling system.

C. Added additional information to the introduction to make the findings more generic.

D. For the evaluation finding for GDC 64 for BWRs, the gaseous radwaste storage tank was changed to the offgas charcoal delay or decay beds.

E. Added an evaluation finding for the requirements of 10 CFR 50.34(f)(2)(xxvi) and related clarifications of Item III.D.1.1 in NUREG-0737 to the subsection covering requirements applicable to both BWR sand PWRs.

V. IMPLEMENTATION

No changes.

VI. REFERENCES

A. Eliminated the following references:

4. Regulatory Guide 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition).

16. Regulatory Guide1.56, “Maintenance of Water Purity in Boiling Water Reactors.”

19. SECY 93-87, “Policy, Technical and Licensing Issues Pertaining to ALWR Designs,” April 2,1993.

20. Staff Requirements Memorandum, SECY 93-87, July 21, 1993.

21. Generic Letter 83-37, “NUREG-0737 Technical Specifications,” November 1, 1983.

22. Generic Letter 83-36, “NUREG-0737 Technical Specifications,“ November 1, 1983.

23. ANSI N13.1-1969(R93), “Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities,” American National Standards Institute, issued 1969 (reaffirmed 1993).

B. Added the following references:

18. NRC Letter to CE Owners Group, Containing NRC Approval to Eliminate the Post-accident Sampling System, May 16, 2004 (ADAMS Accession No. ML003715250).

19. NRC Letter to Westinghouse Owners Group, Containing NRC Approval to Eliminate the Post-accident Sampling System, June 14, 2000 (ADAMS Accession No. ML00373268).

20. ANSI/HPS N13.1-1999, “Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities.”

21. NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design,” Vol. 1, issued September 2004.

22. “BWR Water Chemistry Guidelines - Revision 2.” EPRI Report TR-103515-Revision 2, Electric Power Research Institute, February 2000.

23. “Pressurized Water Reactor Secondary Water Chemistry Guidelines - Revision 6," EPRI Report TR-1008224-Revision 6, Electric Power Research Institute, December 2004.

24. “Pressurized Water Reactor Primary Water Chemistry Guidelines: Volume 1, Revision 5," EPRI Report TR-1002884-Revision 5, Electric Power Research Institute, October 2003.

25. “PWR Primary Water Chemistry Guidelines: Vol. 2: Revision 4 Volume 2," EPRI Report TR-105714-V2R4, Electric Power Research Institute, March 1999.