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(Formerly NUREG-75/087)

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

9.3.2 PROCESS AND POST-ACCIDENT SAMPLING SYSTEMS

REVIEW RESPONSIBILITIES

Primary - Materials and Chemical Engineering Branch (~~CMEB~~EMCB)[1]

Secondary - ~~None~~Emergency Preparedness and Radiation Protection Branch (PERB)[2]
- Plant Systems Branch (SPLB)[3]

I. AREAS OF REVIEW

~~CMEB~~EMCB[4] reviews the following information in the applicant's safety analysis report (SAR):

1. The design objectives and design criteria for the process sampling system (PSS) and post-accident sampling system (PAS) are reviewed at the construction permit (CP) stage. During the operating license (OL) stage of review, CMEB 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 and PAS are reviewed at the operating license (OL) stage. The review includes (a) piping and instrumentation diagrams (P&IDs), (b) provisions for obtaining representative samples, (c) 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

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review includes design and expected temperatures and pressures and materials of construction of components of the system.

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

5. The design of the post-accident sampling system, the administrative program[5] and the operational procedures of post-accident sampling for the reactor coolant and containment atmosphere are reviewed to determine the capability of promptly obtaining samples, under accident conditions, for chemical and radiochemical analyses.

Review Interfaces:[6]

The EMCB will coordinate other branches' evaluations that interface with the overall review of the system as follows:

1. The Plant Systems Branch (SPLB) performs the following: (a) review of the ventilation systems which may be operating during post-accident sampling under SRP Section 11.3; (b) verifies, under SRP Section 3.11, that those valves which are inaccessible during an accident are environmentally qualified to ensure operability under accident conditions; (c) reviews, under SRP Section 3.6.1, 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.

2. The Emergency Preparedness and Radiation Protection Branch (PERB) performs the following: (a) reviews, under SRP Section 12.3-12.4, the potential personal radiation exposure during post-accident sampling; (b) reviews, under SRP Section 11.5, the sampling and monitoring systems for radwaste processing systems.

3. The Containment Systems and Severe Accident Branch (SCSB), under SRP Section 6.2.4, verifies that remotely operated containment isolation valves in the PSS and PAS are designed to close on a containment isolation signal (CIS) or safety injection signal (SIS).

4. The Instrumentation and Controls Branch (HICB), under SRP Section 7.3, verifies that an override capability exists for the containment isolation valves which will be used for post-accident sampling of the reactor coolant, containment sump water, and containment atmosphere without clearing the CIS or SIS.

5. The Electrical Engineering Branch (EELB), under SRP Section 8.3.1, ensures that power supplies are available to all remotely operated valves in the PAS, after detection of an accident which requires containment isolation, assuming a concurrent loss of offsite power.

~~CMEB coordinates reviews related to this system that are performed by the following branches as part of their primary review responsibilities: Radiological Assessment Branch (RAB), under~~

~~SRP Section 12.3, reviews the potential personal radiation exposure during post-accident sampling. Effluent Treatment Systems Branch (ETSB) reviews the ventilation systems which may be operating during post-accident sampling and the sampling and monitoring systems for radwaste processing systems under SRP Sections 11.3 and 11.5, respectively. Containment Systems Branch (CSB), under SRP Section 6.2.4, verifies that remotely operated containment isolation valves in the PSS and PAS are designed to close on a containment isolation signal (CIS) or safety injection signal (SIS). Instrumentation and Control Systems Branch (ICSB), under SRP Section 7.3, verifies that an override capability exists for the containment isolation valves which will be used for post-accident sampling of the reactor coolant, containment sump water, and containment atmosphere without clearing the CIS or SIS. Power Systems Branch (PSB), under SRP Section 8.3.1, ensures that power supplies shall be available to all remotely operated valves in the PAS, after detection of an accident which requires containment isolation, assuming a concurrent loss of offsite power. Equipment Qualification Branch (EQB), under SRP Section 3.11, verifies that those valves which are inaccessible during an accident are environmentally qualified to ensure operability under accident conditions. Auxiliary System Branch (ASB), under SRP Section 3.6.1, reviews 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.~~[7]

For those areas of review identified above as being ~~reviewed as~~ part of the ~~primary~~ review ~~responsibility of other branches~~under other SRP sections, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP sections ~~of the corresponding primary branch~~.[8]

II. ACCEPTANCE CRITERIA

~~CMEB~~EMCB[9] acceptance criteria are based on meeting the relevant requirements of the following regulations:

A. 10 CFR Part 20, § ~~20.1(c)~~20.1101(b)[10] as it relates to making every reasonable effort to maintain radiation exposures as low as is reasonably achievable.

B. General Design Criterion 1 as it relates to the design of the PSS and PAS and components to standards commensurate with the importance of their safety functions.

C. General Design Criterion 2 as it relates to the PSS and PAS being able to withstand the effects of natural phenomena.

D. General Design Criterion 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.

E. General Design Criterion 14 as it relates to assuring the integrity of the reactor coolant pressure boundary.

F. General Design Criterion 26 as it relates to controlling reliably the rate of reactivity changes.

G. General Design Criterion 41 as it relates to reducing the concentration and quality of fission products released to the environment following postulated accidents.

H. General Design Criterion 60 as it relates to capability of the PSS and PAS to control~~ling~~[11] the release of radioactive materials to the environment.

I. General Design Criterion 63 as it relates to detecting conditions that may result in excessive radiation levels in the fuel storage and radioactive waste systems.

J. General Design Criterion 64 as it relates to monitoring the containment atmosphere and plant environs for radioactivity.~~, and~~[12]

K. TMI Action Plan Item~~Clarifications of Section~~ II.B.3 ~~in~~of NUREG-0737 (Reference 12) as it relates to the capability to promptly obtain and analyze samples from the reactor coolant system and containment following an accident. 10 CFR 50.34(f)(2)(viii) provides equivalent requirements for those applicants subject to 10 CFR 50.34(f).[13]

L. TMI Action Plan Item III.D.1.1 of NUREG-0737 as it relates to the provisions for a leakage control program to minimize the leakage from those portions of the PSS and PAS 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).[14]

Specific criteria and guidelines of Regulatory Guides 1.21, 1.26, 1.29, 1.56, 1.97 and 8.8 are used to meet the relevant requirements of 10 CFR Part 20, §~~20.1(c)~~20.1101(b)[15], General Design Criteria (GDC) 1, 2, 13, 14, 26, 41, 60, 63, and 64~~ in Appendix A to 10 CFR Part 50~~[16], and 10 CFR 50.34(f)(2)(viii) including[17] the clarifications of item II.B.3 in NUREG-0737, as follows:

1. The applicant's design should be such that the PSS has the capability for sampling all normal process systems and principal components, including provisions for obtaining samples from at least the points indicated below. The guidelines of regulatory position C.2 in Regulatory Guide 1.21~~ (reference 1)~~[18] and regulatory positions C.1 and C.4.a in Regulatory Guide 1.56~~ (reference 2)~~[19] are used to meet the requirements of the relevant General Design Criteria (GDC).

	GDC
a. For a pressurized water reactor (PWR):	
Reactor coolant (e.g., letdown system, etc.)	13, 14, 26, 64
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
Secondary coolant (e.g., condensate hotwell)	13, 14
Pressurizer tank.	64
Steam generator blowdown (if applicable).	14, 64
Secondary coolant condensate treatment waste	64
Sumps inside containment.	64
Containment atmosphere	64
Gaseous radwaste storage tanks.	63, 64

b. For a boiling water reactor (BWR):

Main condenser evacuation system offgas.	64
Reactor coolant (inlet and outlet of reactor water cleanup system).	13, 14, 64
Standby liquid control system tank.	13, 26
Sumps inside containment.	64
Spent fuel pool.	63
Drywell atmosphere (Mark I & II).	64
Inlet and outlet of gaseous radwaste storage tank.	63, 64
Inlet and outlet of condensate polishing system.	13, 14

Other sample points that may be included in the PSS but do not require remote sampling are given in SRP Section 11.5.

2. The required analysis and frequencies should be given in the plant technical specifications.

3. ~~CMEB~~EMCB[20] will use the following guidelines for determining the acceptability of the PSS functional design:

   a. Provisions should be made to assure 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 Regulatory Guide 1.21,~~ (reference 1)~~[21] are used to meet these criteria.

   b. Provisions should be made to assure representative samples from gaseous process streams and tanks in accordance with ANSI N13.1~~-1969~~[22] (~~r~~Reference ~~3~~23)[23]. The guidelines of regulatory position C.6 in Regulatory Guide 1.21~~ (reference 1)~~[24] 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 Regulatory Guide 1.21~~ (reference 1)~~[25] are followed to meet this criterion.

d. Provisions should be made to purge and drain sample streams back to the system or origin or to an appropriate waste treatment system in accordance with the requirements of 10 CFR Part 20, §~~20.1(c)~~20.1101(b)[26] to keep radiation exposures as low as is reasonably achievable. The guidelines of regulatory positions 2.d.(2), 2.f.(3), and 2.f.(8) in Regulatory Guide 8.8~~ (reference 5)~~[27] are followed to meet this criterion.

e. Isolation valves should fail in the closed position, in accordance with the requirements of GDC 60~~ in Appendix A to 10 CFR Part 50~~[28] 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 Part 20, § ~~20.1(c)~~20.1101(b)[29] to keep radiation exposures to as low as reasonably achievable and the requirements of GDC 60~~ in Appendix A to 10 CFR Part 50~~[30] to control the release of radioactive materials to the environment. The guidelines of position 2.i.(6) in Regulatory Guide 8.8~~ (reference 5)~~[31] should be followed to meet this criterion. Passive flow restrictions in the sample lines may be replaced by redundant environmentally qualified, remotely operated isolation valves to limit potential leakage from sampling lines. The automatic containment isolation valves should close on containment isolation signals or safety injection signals.

4. To meet the requirements of GDC 1 and 2, the seismic design and quality group classification of sampling lines, components and instruments for both the PSS and PAS 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 Regulatory Guide 1.26~~ (reference 6)~~[32], regulatory positions C.1, C.2, C.3, and C.4 in Regulatory Guide 1.29~~ (reference 7)~~[33], and the guidelines of Regulatory Guide 1.97~~ (reference 8)~~[34]. 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 Regulatory Guide 1.26~~ (reference 6)~~[35].

5. The post-accident sampling system and operational procedures should meet the guidelines of item II.B.3 in NUREG-0737~~ (reference 9)~~[36], and of Regulatory Guide 1.97 ~~(reference 8)~~[37], and the following additional clarifications:

a. To meet the requirements of GDC 13 and 14~~ in Appendix A to 10 CFR Part 50~~[38], if chemical analyses show that chloride concentration in the reactor coolant exceeds the Technical Specification limits, then verification that the dissolved oxygen concentration is below the Technical Specification limits will be mandatory. Verification of hydrogen residual in excess of 10 cubic centimeters per kilogram[39] (at standard temperature and pressure)~~ per kilogram~~ of reactor coolant will be acceptable in lieu of direct analysis of dissolved oxygen for 20 days.

b. To meet the requirements of GDC 60~~ in Appendix A to 10 CFR Part 50~~[40], if on line gas chromatography is used for reactor coolant analyses, special provisions (e.g., pressure relief and purging) should be available to prevent high-pressure carrier gas from entering the reactor coolant.

c. To meet the requirements of GDC 60~~ in Appendix A to 10 CFR Part 50~~[41], passive flow restrictions in the sampling lines may be replaced by redundant, fully qualified, remotely operated isolation valves to limit potential leakage from the sample lines. The automatic containment isolation valves should close on containment isolation signals or safety injection signals. All remotely operated valves should have assured power supplies and control so that they can be reopened after an accident without clearing the isolation signal. Valves which are inaccessible during an accident should be environmentally qualified to ensure operability under accident conditions.

Technical Rationale:[42]

The application of the above acceptance criteria to the process and post-accident sampling systems is addressed in the following paragraphs.

1. 10 CFR Part 20, §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 as low as is reasonably achievable (ALARA). The radiation protection community has recognized that it is prudent to avoid unnecessary exposure to radiation and to maintain doses ALARA as it is assumed that there is a non-threshold linear relationship between dose and biological effects that is independent of the dose rate. The objective of the ALARA requirement in regard to the process and post-accident sampling systems 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 process and post-accident sampling systems. The safety benefit of implementing radiation protection goals in regard to the process and post-accident sampling systems 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 structures, systems and components important to safety be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety functions to be performed. The process and post-accident sampling systems are important to safety in that: 1) 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; 2) during

normal operation, the process sampling system provides information that allows the operator to assess integrity of the fuel cladding and to recognize conditions that could jeopardize integrity of the reactor coolant pressure boundary; 3) following an accident, the post-accident sampling system provides information that allows the operator to assess the success of safety systems and systems important to safety in mitigating the consequences of the accident and to determine appropriate actions for further mitigation or recovery from the accident; and 4) the sampling systems contain or may contain radioactive material. Meeting the requirements of GDC 1 ensures that the process and post-accident sampling systems will be designed, fabricated, erected and tested to generally accepted and recognized codes and standards that are sufficient to assure a quality system in keeping with the required safety functions.

3. GDC 2 requires that structures, systems and components important to safety be designed to withstand the effects of natural phenomena without the loss of capability to perform their safety functions. The process and post-accident sampling systems connect to systems, such as the reactor coolant pressure boundary, that are designed to Seismic Category I requirements. Those portions of the process and post-accident sampling systems or components that form interfaces between Seismic Category I and non-Seismic Category I features should be designed to Seismic Category I requirements. Meeting the requirements of GDC 2 for those portions of the process and post-accident sampling systems 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 assure 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 process sampling system is relied upon to provide water and gaseous samples from the reactor coolant system and associated auxiliary systems during all normal modes of operation. The post-accident sampling system provides the capability to obtain and analyze reactor coolant and containment atmosphere samples to determine the extent of core degradation following an accident. Satisfying the requirements of GDC 13 for the process and post-accident sampling systems ensures 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, of rapidly propagating failure, and of gross rupture. The process and post-accident sampling systems are 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 due to degradation from corrosive chemical attack.

6. GDC 26 establishes requirements regarding reliably controlling the rate of reactivity changes. In PWRs, the sampling systems are utilized 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. In BWRs, the sampling systems are used 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 RCS, enhancing plant safety by: 1) meeting the combined reactivity control system redundancy and capability requirements, and 2) 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 process sampling system 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 process and post-accident sampling systems are 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 and post-accident samples are 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 process and post accident sampling systems, 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 process and post-accident sampling systems contain or may contain radioactive material that must be properly controlled. Examples of the controls used to prevent release of radioactive material from the process and post-accident sampling systems to the environment include: 1) redundant automatic isolation valves that will close on a containment isolation signal or safety injection signal and will fail in the closed position; 2) purging and draining the sample lines back to the system being sampled or to the radwaste treatment system; and 3) 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 process and post-accident sampling systems, enhances safety by

providing a means in the design to control the release of radioactive material to the environment. Application of criterion 60 provides reasonable assurance that the process and post-accident sampling systems are 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 process sampling system, 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 process sampling system provides information necessary for the control of water chemistry to maintain the material properties of the fuel assembly cladding, structural members and cooling systems of the spent fuel pool. The process sampling system, through the ability to sample the gaseous radwaste storage tanks, 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 process sampling systems, 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 as low as reasonably achievable.

10. GDC 64 requires means be available for monitoring the containment atmosphere, spaces containing components used for post-LOCA recirculation, effluent discharge paths and the plant environs for radioactivity that may be released during normal operations, anticipated operational occurrences, and postulated accidents. The process and post-accident sampling system designs provide the means for monitoring for radioactivity that may be released during normal operations, anticipated operational occurrences and postulated accidents. The process and post-accident sampling systems provide 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 process and post-accident sampling systems provide 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 process and post-accident sampling systems, ensures that a means is provided to monitor the release of radioactive materials providing the plant operator with 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 II.B.3 of NUREG-0737 require provision of the capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain radioactive materials without exceeding specified radiation exposures. 10 CFR 50.34(f)(2)(viii) establishes equivalent requirements for those applicants subject to 10 CFR 50.34(f). Materials to be analyzed and quantified include samples that are indicators of core damage, hydrogen in the containment atmosphere, dissolved gasses, chloride, and boron concentrations. Prompt sampling and analysis of the reactor coolant system and the containment atmosphere following an accident provides important information necessary to the operators efforts to assess and control the plant during the course of an accident. Critical information regarding core damage and coolant

characteristics can be obtained through radiological and chemical analysis of primary coolant liquid and gas samples. Additional information concerning core damage and the potential for a hydrogen reaction in containment can be obtained from an analysis of a containment atmosphere sample. Meeting the guidance of TMI Action Plan Item II.B.3 as clarified in NUREG-0737, as it relates to the design of the post-accident sampling systems, ensures that important sample information can be obtained and analyzed promoting rapid accident diagnosis, improving operator control and enhancing public safety.

12. TMI Action Plan Item III.D.1.1 of NUREG-0737 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.
10 CFR 50.34(f)(2)(xxvi) establishes equivalent requirements for those applicants subject to 10 CFR 50.34(f). The process and post accident sampling systems provide a means for sampling the reactor coolant systems and containment atmosphere to provide information necessary to assess and control the plant under accident conditions. Because these systems draw samples directly from the reactor coolant systems or from the containment atmosphere they have a 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 these systems to as low as practical levels. Meeting the guidance of TMI Action Plan Item III.D.1.1 as clarified in NUREG-0737, as it relates to the process and post accident sampling systems, 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 and emphasize material from this SRP section, as may be appropriate for a particular case.

1. In the review of the process sampling system, ~~CMEB~~EMCB[43] compares the list of process sampling points contained in the SAR with the sampling points identified in Subsection II.1, above, to assure that the required process sampling points have been provided.

2. ~~CMEB~~EMCB[44] compares 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 Regulatory Guide 1.21~~ (reference 1)~~[45] and with the principles for obtaining representative samples of gases contained in ANSI N13.1~~-1969 (reference 3)~~[4647].

3. ~~CMEB~~EMCB[48] compares the seismic design and quality group classifications of the PSS and PAS to the classifications of the fluid systems to which the sampling system is connected.

4. ~~CMEB~~EMCB[49] reviews the technical specifications for process sampling to determine that the content and intent of the technical specifications are in agreement with the requirements developed as a result of the staff's review.[50]

5. ~~CMEB~~EMCB[51] verifies that provisions have been made to limit the potential for reactor coolant loss from the rupture of a sample line and provides ~~Accident Evaluation Branch~~the PERB[52] with estimates of RCS fluid losses that would result from sample line rupture.

6. ~~CMEB~~EMCB[53] reviews the post-accident sampling system and operational procedures to determine compliance with the guidelines of item II.B.3 in NUREG-0737 ~~(reference 9)~~[54]and of Regulatory Guide 1.97 ~~(reference 8)~~[55]and the additional clarifications in specific acceptance criterion II.5[56] above. ~~CMEB~~EMCB[57] verifies that dilution, mixing and sample collection steps will not introduce excessive analytical errors. ~~CMEB~~EMCB[58] verifies that the frequency of system functional testing will be adequate to ensure operability of the system and that operators are trained and kept proficient in the use of the system. ~~CMEB~~EMCB[59] reviews the procedure for relating radionuclide concentrations to the extent of core damage. ~~CMEB~~EMCB[60] reviews information pertaining to accuracy and sensitivity of chemical analysis procedures and on-line instrumentation under post-accident chemistry conditions (high specific sample activity and possible analytical interference due to high levels of fission products-iodine, iodide, bromide, cesium, rubidium).

In the FSERs for the ABWR and System 80+ design certifications the staff accepted exemptions from certain 10 CFR 50.34(f)(2)(viii) and NUREG 0737 post accident sampling system (PASS) requirements. Details of these exemptions are outlined in the following table:

APPLICABILITY	REQUIRED SAMPLE	SOURCE	EXEMPTION	BASIS
-System 80+ -ABWR	Reactor coolant system (RCS) boron concentration within 3 hours	Item II.B.3 of NUREG 0737	Sample for boron within 8 hours	Sample used only to confirm mitigation measures. Core condition obtained by other methods (neutron flux monitoring).
-System 80+ -ABWR	RCS activity within 3 hours	Item II.B.3 of NUREG 0737	Sample for activity within 24 hours	Sample used only to confirm mitigation measures. Core condition obtained by other methods (cont. high range area monitor, reactor vessel water level, core exit thermocouples).
-System 80+ -ABWR	Containment atmosphere for hydrogen within 3 hours	-10 CFR 50.34(f)(2)(viii) -Item II.B.3 of NUREG 0737	Sample for cont. hydrogen within 24 hours	Containment hydrogen monitor will be used during initial accident phase.
System 80+	RCS dissolved gasses within 3 hours	-10 CFR 50.34(f)(2)(viii) -Item II.B.3 of NUREG 0737	Sample for RCS dissolved gasses within 24 hours	Venting will remove gasses accumulated in reactor vessel. Vented gasses will be sampled.
System 80+	RCS chloride within 3 hours	-10 CFR 50.34(f)(2)(viii) -Item II.B.3 of NUREG 0737	Sample for RCS chloride within 24 hours	Chloride corrosion is a secondary consideration and can be addressed long term.
ABWR	RCS dissolved gasses and chloride within 3 hours	-10 CFR 50.34(f)(2)(viii) -Item II.B.3 of NUREG 0737	RCS dissolved gasses and chloride samples not required	When core uncovering is expected reactor vessel is depressurized. Chloride corrosion is a secondary consideration and can be addressed long term.

These exemptions meet the staff recommendations from SECY 93-087 which were approved by the Commission in a Staff Requirements Memorandum (References 19 and 20).[61]

7. EMCB determines that a post-accident sampling system administrative program has been established, implemented and will be maintained to ensure the plant has the capability to

obtain and analyze reactor coolant samples, containment atmosphere samples, radioactive iodides and particulates in plant gaseous effluents under accident conditions (References 21 and 22). The administrative program should include:

a) training of personnel
b) procedures for sampling and analysis, and
c) provisions for maintenance of sampling and analysis equipment.

Referencing the program in the administrative controls section of the Technical Specifications and including a detailed description of the program in the plant operation manuals is acceptable. A copy of the program should be readily available to the operating staff during accident and transient conditions.[62]

8. EMCB verifies that those portions of the process sampling and post-accident sampling systems located outside of containment that contain or may contain radioactive material following an accident are included in a leakage control program. The leakage control program should include periodic leak testing and measures to minimize leakage from the systems.[63]

For standard design certification reviews under 10 CFR Part 52, the procedures above should be followed, as modified by the procedures in SRP Section 14.3 (proposed), to verify that the design set forth in the standard safety analysis report, including inspections, tests, analysis, and acceptance criteria (ITAAC), site interface requirements and combined license action items, meet the acceptance criteria given in subsection II. SRP Section 14.3 (proposed) contains procedures for the review of certified design material (CDM) for the standard design, including the site parameters, interface criteria, and ITAAC.[64]

IV. EVALUATION FINDINGS

~~CMEB~~EMCB[65] verifies that sufficient information has been provided and that the review supports conclusions of the following type, to be included in the staff's safety evaluation report:

> The process and post-accident sampling systems include piping, valves, heat exchangers, and other components associated with the systems 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 design of these systems. The review has included descriptive information for the process and post-accident sampling systems 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 and post-accident sampling systems to applicable regulations, guides, and industry standards.

The staff concludes that the design of the process and post-accident sampling systems are acceptable and that the process sampling system meets the relevant requirements of 10 CFR Part 20, § ~~20.1(c)~~20.1101(b)[66] and General Design Criteria 1, 2, 13, 14, 26, 41 (for PWR only), 60, 63, and 64 ~~in Appendix A to 10 CFR Part 50~~[67], and the post-accident sampling system meets the relevant requirements of General Design Criteria 1, 2, 13, 14, and 60~~ in Appendix A to 10 CFR Part 50~~[68], 10 CFR 50.34(f)(2)(viii),[69] Item II.B.3 of NUREG-0737, 10 CFR 50.34(f)(2)(xxvi), and Item III.D.1.1 of NUREG-0737.[70] This conclusion is based on the following:

For PWR

The staff has determined that the proposed process sampling system meets (1) the requirements of GDC 13~~ in Appendix A to 10 CFR Part 50~~[71] 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~~ in Appendix A to 10 CFR Part 50~~[72], 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~~ in Appendix A to 10 CFR Part 50~~[73] 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~~ in Appendix A to 10 CFR Part 50~~[74] 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 chemical 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~~ in Appendix A to 10 CFR Part 50~~[75] 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~~ in Appendix A to 10 CFR Part 50~~[76] 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 13 and 26~~ in Appendix A to 10 CFR Part 50~~[77] 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 ~~in Appendix A to 10 CFR Part 50~~[78], 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 gaseous radwaste storage tank for radioactivity.

For both PWR and BWR

The staff has further determined that the proposed process sampling system meets (a) the requirements of 10 CFR Part 20, § ~~20.1(c)~~20.1101(b)[79] to keep radiation exposures to as low as is reasonably achievable and of GDC 60 ~~in Appendix A to 10 CFR Part 50~~[80] to control the release of radioactive materials to the environment, by purging and draining sample streams back to the system or 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; and (b) the requirements of GDC 63 ~~in Appendix A to 10 CFR Part 50~~[81], 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.

The staff also has determined that the proposed process sampling and post-accident sampling systems meet the quality standards requirements of GDC 1 and the seismic requirements of GDC 2 ~~in Appendix A to 10 CFR Part 50~~[82], by designing the sampling lines and components of the process and post-accident sampling systems to conform to the classification of the system to which each sampling line and component is connected, in accordance with the regulatory positions C.1, C.2, and C.3 of Regulatory Guide 1.26, the regulatory positions C.1, C.2, C.3, and C.4 of Regulatory Guide 1.29, and the guidelines of Regulatory Guide 1.97.

In addition, the staff has determined that the proposed post-accident sampling system meets (1) the requirements of 10 CFR 50.34(f)(2)(viii) and related[83] clarifications of I~~i~~tem II.B.3 in NUREG-0737 by providing a sampling program to obtain and analyze promptly samples from the reactor coolant and the containment atmosphere, for radionuclides which may be indicators of the degree of core damage and for dissolved gases, chloride and boron concentrations in liquids, following a postulated accident, in accordance with the guidelines of Regulatory Guide 1.97; (2) the requirements of GDC 13 and 14 ~~in Appendix A to 10 CFR Part 50~~[84], to monitor variables which can affect the integrity of the reactor core and the reactor coolant pressure boundary for accident conditions, and to assure a low probability of abnormal leakage, rapidly propagating failure, and gross rupture, respectively, by keeping the oxygen concentration in the reactor coolant below the Technical Specification limits when the chloride concentration exceeds the Technical Specification limits, or by having excessive hydrogen dissolved in the reactor coolant; ~~and~~ (3) the requirements of GDC 60 ~~in Appendix A to 10 CFR Part 50~~[85] to control the release of radioactive materials to the environment, by preventing high pressure carrier gas used in gas chromatographic analysis of reactor coolant from entering the reactor coolant, and by providing passive flow restrictions or fully qualified, remotely operated isolation valves in the sampling lines, to limit potential leakage from the sampling lines; and (4) 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.[86]

For design certification reviews, the findings will also summarize, to the extent that the review is not discussed in other safety evaluation report sections, the staff's evaluation of inspections, tests analyses, and acceptance criteria (ITAAC), including design acceptance criteria (DAC), site interface requirements, and combined license action items that are relevant to this SRP section.[87]

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

This SRP section will be used by the staff when performing safety evaluations of license applications submitted by applicants pursuant to 10 CFR 50 or 10 CFR 52.[88] Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of 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.[89]

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides and NUREG.

VI. REFERENCES

1. 10 CFR Part 20, §20.1101(b), "Radiation Protection Programs."[90]

2. 10 CFR Part 50, §50.34(f), "Additional TMI-related requirements."[91]

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

4~~11~~. 10 CFR Part 50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

5~~12~~. 10 CFR Part 50, Appendix A, General Design Criterion 13, "Instrumentation and Control."

6~~13~~. 10 CFR Part 50, Appendix A, General Design Criterion 14, "Reactor Coolant Pressure Boundary."

7~~14~~. 10 CFR Part 50, Appendix A, General Design Criterion 26, "Reactivity Control System Redundancy and Capability."

8~~15~~. 10 CFR Part 50, Appendix A, General Design Criterion 41, "Containment Atmosphere Cleanup."

9~~16~~. 10 CFR Part 50, Appendix A, General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment."

10~~17~~. 10 CFR Part 50, Appendix A, General Design Criterion 63, "Monitoring Fuel and Waste Storage."

11~~18~~. 10 CFR Part 50, Appendix A, General Design Criterion 64, "Monitoring Radioactivity Releases."

12~~9~~. NUREG-0737, "Clarification~~s~~[92] of TMI Action Plan Requirements."

13~~1~~. 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~~6~~. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants."

15~~7~~. Regulatory Guide 1.29, "Seismic Design Classification."

16~~2~~. Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors."~~-~~

~~4. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," LWR Edition.~~[93]

17~~8~~. Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions during and following ~~A~~an[94] Accident."

18~~5~~. Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable."

19. SECY 93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated April 2, 1993.[95]

20. Staff Requirements Memorandum, "SECY 93-087 - Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated July 21, 1993.[96]

21. NRC Letter to All Pressurized Water Reactor Licensees, "NUREG-0737 Technical Specifications (Generic Letter 83-37)," November 1, 1983.[97]

22. NRC Letter to All Boiling Water Reactor Licensees, "NUREG-0737 Technical Specifications (Generic Letter 83-36)," November 1, 1983.[98]

23~~3~~. ANSI N13.1-1969(R93), "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities," American National Standards Institute (1969)(reaffirmed 1993)[99].

SRP Draft Section 9.3.2
Attachment A - Proposed Changes in Order of Occurrence

Item numbers in the following table correspond to superscript numbers in the redline/strikeout copy of the draft SRP section.

Item	Source	Description
1.	Current PRB Names and Abbreviations.	Editorial change made to reflect current PRB names and responsibilities for this SRP section.
2.	Current PRB names and abbreviations.	Added the Emergency Preparedness and Radiation Protection Branch (PERB) as a PRB branch with secondary review responsibilities for this section in accordance with the current PRB organizational responsibilities.
3.	Current PRB names and abbreviations.	Added the Plant Systems Branch (SPLB) as a PRB branch with secondary review responsibilities for this section in accordance with the current PRB organizational responsibilities.
4.	Current PRB Names and Abbreviations.	Editorial change made to reflect current PRB names and responsibilities for this SRP section.
5.	**Integrated Impact # 243.**	Step 5 of the Areas of Review was modified to include a reference to the administrative program for the post-accident sampling system. Including the reference to an administrative program in the Areas of Review for this SRP section is consistent with the staff positions on the post-accident sampling system and its associated technical specifications contained in Generic Letter 83-36.
6.	SRP-UDP format item.	Revised the review interface section of Areas of Review to be consistent with SRP-UDP required format that uses a number/paragraph format to distinguish individual reviews and supporting reviews performed by other PRBs.
7.	SRP-UDP Format Item.	Deleted the old single paragraph format used for review interfaces, all of the listed review interfaces were included in the new review interface paragraphs.
8.	SRP-UDP format item	Revised to reflect applicability of interfaces with respect to other SRP sections rather than other branches.
9.	Current PRB Names and Abbreviations.	Editorial change made to reflect current PRB names and responsibilities for this SRP section.
10.	Reference Verification.	This is an administrative change to reference the revised section of 10 CFR Part 20 § 20.1101(b) that corresponds to the old § 20.1(c) for implementation of ALARA. For documentation of the changes to 10 CFR Part 20 see PI-23735 and Federal Register notice 56 FR 23396.
11.	Editorial Correction.	The use of "to controlling" is incorrect grammar and was replaced with "to control."

Item	Source	Description
12.	Editorial.	Deleted the "and" as an additional item was added to the list of Acceptance Criteria.
13.	**Integrated Impacts # 242 and # 244.**	Citation in the Acceptance Criteria of both the 10 CFR 50.34(f)(2)(viii) requirements and the clarifications contained in NUREG-0737 Item II.B.3 was done to bound the applicability of this issue to the necessary license applicants without the need to discuss applicability issues in the Acceptance Criteria. Existing operating licenses are bounded by their individual responses and commitments to NUREG-0737 Item II.B.3. CP/ML license applicants are subject to 10 CFR 50.34(f)(2)(viii) and DC/COL applicants, via 10 CFR 52, are subject to the requirements contained in 10 CFR 50.34(f)(2)(viii). Modifications to this requirement for DC/COL applicants consistent with the ABWR and ABB CE80+ FSERs and associated SECY papers are discussed under the specific acceptance criteria. In addition, a descriptive sentence consistent with the regulations of 10 CFR 50.34(f)(2)(viii) was added to provide clarity and consistency for the reviewer.
14.	**Integrated Impact # 854**	10 CFR 50.34(f)(2)(xxvi) including the clarification of Item III.D.1.1 of NUREG-0737 was added to the Acceptance Criteria. This criteria establishes the provisions for a leakage control program for those systems outside of containment that contain or may contain radioactive materials following an accident. Citation in the Acceptance Criteria of both the 10 CFR 50.34(f)(2)(xxvi) requirements and the clarifications contained in NUREG-0737 Item III.D.1.1 was done to bound the applicability of this issue to the necessary license applicants without the need to discuss applicability issues in the Acceptance Criteria. Existing operating licenses are bounded by their individual responses and commitments to NUREG-0737 Item III.D.1.1. CP/ML license applicants are subject to 10 CFR 50.34(f)(2)(xxvi) and DC/COL applicants, via 10 CFR 52, are subject to the requirements contained in 10 CFR 50.34(f)(2)(xxvi).
15.	Reference Verification.	This is an administrative change to reference the revised section of 10 CFR Part 20 § 20.1101(b) that corresponds to the old § 20.1(c) for implementation of ALARA. For documentation of the changes to 10 CFR Part 20 see PI-23735 and Federal Register notice 56 FR 23396.
16.	Editorial.	This change deleted the unnecessary phrase "in Appendix A to 10 CFR Part 50" following GDC 64. Referencing the GDC is sufficient and use of the phrase is redundant and unnecessary.

Item	Source	Description
17.	**Integrated Impact #242** and editorial.	Added 10 CFR 50.34(f)(2)(viii) to the list of criteria that is addressed under the specific criteria section. This is an administrative change to ensure consistency between the listing of criteria and those acceptance criteria that are discussed in greater detail under the specific criteria section.
18.	SRP-UDP format item	Removed identification of Regulatory Guides by reference number consistent with SRP-UDP format requirements.
19.	SRP-UDP format item	Removed identification of Regulatory Guides by reference number consistent with SRP-UDP format requirements.
20.	Current PRB Names and Abbreviations.	Editorial change made to reflect current PRB names and responsibilities for this SRP section.
21.	SRP-UDP Format Item, Reformat Reference Citations.	The parenthetical notation for a reference is only used for the first occurrence, all subsequent citations of the reference do not require use of a parenthetical notation.
22.	**Integrated Impact # 671.**	ANSI standard ANSI N 13.1 was reaffirmed in 1993. The dates for those codes and standards cited in the SRP that have been reaffirmed will be indicated in the references subsection. Since the version specific information will be contained in the reference subsection it has been deleted from the body of the text in accordance with SRP-UDP formatting requirements.
23.	Editorial.	Renumbered the references to be consistent with the re-ordering of the reference section. In addition, the word reference was capitalized in accordance with the SRP-UDP format.
24.	SRP-UDP Format Item, Reformat Reference Citations.	The parenthetical notation for a reference is only used for the first occurrence, all subsequent citations of the reference do not require use of a parenthetical notation.
25.	SRP-UDP Format Item, Reformat Reference Citations.	The parenthetical notation for a reference is only used for the first occurrence, all subsequent citations of the reference do not require use of a parenthetical notation.
26.	Reference Verification.	This is an administrative change to reference the revised section of 10 CFR Part 20 § 20.1101(b) that corresponds to the old § 20.1(c) for implementation of ALARA. For documentation of the changes to 10 CFR Part 20 see PI-23735 and Federal Register notice 56 FR 23396.

Item	Source	Description
27.	SRP-UDP format item	Removed identification of Regulatory Guides by reference number consistent with SRP-UDP format requirements.
28.	Editorial.	This change deleted the unnecessary phrase "in Appendix A to 10 CFR Part 50" following GDC 60. Referencing the GDC is sufficient and use of the phrase is redundant and unnecessary.
29.	Reference Verification.	This is an administrative change to reference the revised section of 10 CFR Part 20 § 20.1101(b) that corresponds to the old § 20.1(c) for implementation of ALARA. For documentation of the changes to 10 CFR Part 20 see PI-23735 and Federal Register notice 56 FR 23396.
30.	Editorial.	This change deleted the unnecessary phrase "in Appendix A to 10 CFR Part 50" following GDC 60. Referencing the GDC is sufficient and use of the phrase is redundant and unnecessary.
31.	SRP-UDP Format Item, Reformat Reference Citations.	The parenthetical notation for a reference is only used for the first occurrence, all subsequent citations of the reference do not require use of a parenthetical notation.
32.	SRP-UDP format item	Removed identification of Regulatory Guides by reference number consistent with SRP-UDP format requirements.
33.	SRP-UDP format item	Removed identification of Regulatory Guides by reference number consistent with SRP-UDP format requirements.
34.	SRP-UDP format item	Removed identification of Regulatory Guides by reference number consistent with SRP-UDP format requirements.
35.	SRP-UDP Format Item, Reformat Reference Citations.	The parenthetical notation for a reference is only used for the first occurrence, all subsequent citations of the reference do not require use of a parenthetical notation.
36.	SRP-UDP Format Item, Reformat Reference Citations.	The parenthetical notation for a reference is only used for the first occurrence, all subsequent citations of the reference do not require use of a parenthetical notation.
37.	SRP-UDP Format Item, Reformat Reference Citations.	The parenthetical notation for a reference is only used for the first occurrence, all subsequent citations of the reference do not require use of a parenthetical notation.

Item	Source	Description
38.	Editorial.	This change deleted the unnecessary phrase "in Appendix A to 10 CFR Part 50" following GDC 13 and 14. Referencing the GDC is sufficient and use of the phrase is redundant and unnecessary.
39.	Editorial.	Moved the parenthetical reference regarding standard temperature and pressure to immediately follow the presentation of the units.
40.	Editorial.	This change deleted the unnecessary phrase "in Appendix A to 10 CFR Part 50" following GDC 60. Referencing the GDC is sufficient and use of the phrase is redundant and unnecessary.
41.	Editorial.	This change deleted the unnecessary phrase "in Appendix A to 10 CFR Part 50" following GDC 60. Referencing the GDC is sufficient and use of the phrase is redundant and unnecessary.
42.	SRP-UDP format item, adding technical rationale.	Technical rationale were developed and added for the following Acceptance Criteria: 10 CFR Part 20 § 20.1101(b), GDCs 1,2,13,14,26,41,60,63,64, 10 CFR 50.34(f)(2)(viii) and 10 CFR 50.34(f)(2)(xxvi).
43.	Current PRB Names and Abbreviations.	Editorial change made to reflect current PRB names and responsibilities for this SRP section.
44.	Current PRB Names and Abbreviations.	Editorial change made to reflect current PRB names and responsibilities for this SRP section.
45.	SRP-UDP Format Item, Reformat Reference Citations.	The parenthetical notation for a reference is only used for the first occurrence, all subsequent citations of the reference do not require use of a parenthetical notation.
46.	**Integrated Impact # 671.**	ANSI standard ANSI N 13.1 was reaffirmed in 1993. The dates for those codes and standards cited in the SRP that have been reaffirmed will be indicated in the references subsection. Since the version specific information will be contained in the reference subsection it has been deleted from the body of the text in accordance with SRP-UDP formatting requirements.
47.	SRP-UDP Format Item, Reformat Reference Citations.	The parenthetical notation for a reference is only used for the first occurrence, all subsequent citations of the reference do not require use of a parenthetical notation.
48.	Current PRB Names and Abbreviations.	Editorial change made to reflect current PRB names and responsibilities for this SRP section.
49.	Current PRB Names and Abbreviations.	Editorial change made to reflect current PRB names and responsibilities for this SRP section.

SRP Draft Section 9.3.2

Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
50.	Editorial Correction.	Added a period at the end of the sentence.
51.	Current PRB Names and Abbreviations.	Editorial change made to reflect current PRB names and responsibilities for this SRP section.
52.	Current PRB names and abbreviations.	Replaced "Accident Evaluation Branch" with "PERB" This review procedure provides estimates of the RCS fluid losses that would result from a sample line rupture for use during the reviews performed under SRP section 15.6.2 by the Emergency Preparedness and Radiation Protection Branch.
53.	Current PRB Names and Abbreviations.	Editorial change made to reflect current PRB names and responsibilities for this SRP section.
54.	SRP-UDP Format Item, Reformat Reference Citations.	The parenthetical notation for a reference is only used for the first occurrence, all subsequent citations of the reference do not require use of a parenthetical notation.
55.	SRP-UDP Format Item, Reformat Reference Citations.	The parenthetical notation for a reference is only used for the first occurrence, all subsequent citations of the reference do not require use of a parenthetical notation.
56.	Editorial.	Added the word "specific" and modified the identification of the associated step number. These changes ensure a clear reference for use by the Reviewer in locating the additional clarifications.
57.	Current PRB Names and Abbreviations.	Editorial change made to reflect current PRB names and responsibilities for this SRP section.
58.	Current PRB Names and Abbreviations.	Editorial change made to reflect current PRB names and responsibilities for this SRP section.
59.	Current PRB Names and Abbreviations.	Editorial change made to reflect current PRB names and responsibilities for this SRP section.
60.	Current PRB Names and Abbreviations.	Editorial change made to reflect current PRB names and responsibilities for this SRP section.
61.	**Integrated Impact 242**	Added a table which illustrates the PASS exemptions allowed in the evolutionary FSERs. All the exemptions meet the positions of SECY 93-087.
62.	**Integrated Impact # 243.**	A new Review Procedure was added to address the NRC staff guidance contained in Generic Letters 83-36 and 83-37 regarding an administrative program for the post-accident sampling systems. The new Review Procedure is consistent with the NRC staff guidance contained in Enclosure 1 of the Generic Letters.

Item	Source	Description
63.	**Integrated Impact # 854**	Added a new Review Procedure to address the review of the leakage control program necessary to meet the requirements contained in 10 CFR 50.34(f)(2)(xxvi) and as clarified under item III.D.1.1 of NUREG-0737. This review procedure is consistent with section 20.5.38 of the ABWR FSER which documents staff reviews of a leakage control program including periodic leak testing and measures to minimize the leakage from the systems.
64.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard paragraph to address application of Review Procedures in design certification reviews.
65.	Current PRB Names and Abbreviations.	Editorial change made to reflect current PRB names and responsibilities for this SRP section.
66.	Reference Verification.	This is an administrative change to reference the revised section of 10 CFR Part 20 § 20.1101(b) that corresponds to the old § 20.1(c) for implementation of ALARA. For documentation of the changes to 10 CFR Part 20 see PI-23735 and Federal Register notice 56 FR 23396.
67.	Editorial.	This change deleted the unnecessary phrase "in Appendix A to 10 CFR Part 50" following GDC 64. Referencing the GDC is sufficient and use of the phrase is redundant and unnecessary.
68.	Editorial.	This change deleted the unnecessary phrase "in Appendix A to 10 CFR Part 50" following GDC 60. Referencing the GDC is sufficient and use of the phrase is redundant and unnecessary.
69.	**Integrated Impacts # 242 and # 244.**	10 CFR 50.34(f)(2)(viii) was added to the introductory paragraph of the Evaluation Findings covering Item II.B.3 of NUREG-0737. The modifications to this Acceptance Criteria for the Evolutionary and Advanced Light-Water Reactors contained in SECY 93-087 and its associated SRM are addressed in a separate Evaluation Finding specific to the Evolutionary and Advanced Light-Water Reactors. 10 CFR 50.34(f)(2)(viii) contains the post accident sampling requirements related to NUREG-0737 Item II.B.3.
70.	**Integrated Impact # 854**	Added 10 CFR 50.34(f)(2)(xxvi) and item III.D.1.1 of NUREG-0737 to the list of relevant requirements. In addition, adding these requirements to the listing necessitated minor editorial changes to the sentence to accommodate the new requirements.
71.	Editorial.	This change deleted the unnecessary phrase "in Appendix A to 10 CFR Part 50" following GDC 13. Referencing the GDC is sufficient and use of the phrase is redundant and unnecessary.

Item	Source	Description
72.	Editorial.	This change deleted the unnecessary phrase "in Appendix A to 10 CFR Part 50" following GDC 14. Referencing the GDC is sufficient and use of the phrase is redundant and unnecessary.
73.	Editorial.	This change deleted the unnecessary phrase "in Appendix A to 10 CFR Part 50" following GDC 26. Referencing the GDC is sufficient and use of the phrase is redundant and unnecessary.
74.	Editorial.	This change deleted the unnecessary phrase "in Appendix A to 10 CFR Part 50" following GDC 41. Referencing the GDC is sufficient and use of the phrase is redundant and unnecessary.
75.	Editorial.	This change deleted the unnecessary phrase "in Appendix A to 10 CFR Part 50" following GDC 64. Referencing the GDC is sufficient and use of the phrase is redundant and unnecessary.
76.	Editorial.	This change deleted the unnecessary phrase "in Appendix A to 10 CFR Part 50" following GDC 14. Referencing the GDC is sufficient and use of the phrase is redundant and unnecessary.
77.	Editorial.	This change deleted the unnecessary phrase "in Appendix A to 10 CFR Part 50" following GDC 26. Referencing the GDC is sufficient and use of the phrase is redundant and unnecessary.
78.	Editorial.	This change deleted the unnecessary phrase "in Appendix A to 10 CFR Part 50" following GDC 64. Referencing the GDC is sufficient and use of the phrase is redundant and unnecessary.
79.	Reference Verification.	This is an administrative change to reference the revised section of 10 CFR Part 20 § 20.1101(b) that corresponds to the old § 20.1(c) for implementation of ALARA. For documentation of the changes to 10 CFR Part 20 see PI-23735 and Federal Register notice 56 FR 23396.
80.	Editorial.	This change deleted the unnecessary phrase "in Appendix A to 10 CFR Part 50" following GDC 60. Referencing the GDC is sufficient and use of the phrase is redundant and unnecessary.
81.	Editorial.	This change deleted the unnecessary phrase "in Appendix A to 10 CFR Part 50" following GDC 60. Referencing the GDC is sufficient and use of the phrase is redundant and unnecessary.

Item	Source	Description
82.	Editorial.	This change deleted the unnecessary phrase "in Appendix A to 10 CFR Part 50" following GDC 2. Referencing the GDC is sufficient and use of the phrase is redundant and unnecessary.
83.	**Integrated Impacts # 242.**	10 CFR 50.34(f)(2)(viii) was added to the Evaluation Findings covering Item II.B.3 of NUREG-0737 for both PWRs and BWRs. The modifications to this acceptance criteria for the Evolutionary and Advanced Light-Water Reactors contained in SECY 93-087 and its associated SRM are addressed in a separate Evaluation Finding specific to the Evolutionary and Advanced Light-Water Reactors. 10 CFR 50.34(f)(2)(viii) contains the post accident sampling requirements related to NUREG-0737 Item II.B.3.
84.	Editorial.	This change deleted the unnecessary phrase "in Appendix A to 10 CFR Part 50" following GDC 14. Referencing the GDC is sufficient and use of the phrase is redundant and unnecessary.
85.	Editorial.	This change deleted the unnecessary phrase "in Appendix A to 10 CFR Part 50" following GDC 60. Referencing the GDC is sufficient and use of the phrase is redundant and unnecessary.
86.	**Integrated Impact # 854.**	Added an evaluation finding to address the requirements contained in 10 CFR 50.34(f)(2)(xxvi) as clarified by item III.D.1.1 of NUREG-0737. As is consistent with the findings documented in the ABWR FSER these requirements are met by verification that applicable portions of the system are included in a leakage control program that contains periodic leak testing and actions to minimize leakage from the system.
87.	10 CFR 52 applicability issue.	Added a discussion paragraph in the evaluation findings addressing the applicability of the procedures to design certification (DC) and combined license applications.
88.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard sentence to address application of the SRP section to reviews of applications filed under 10 CFR Part 52, as well as Part 50.
89.	SRP-UDP Guidance	Added standard paragraph to indicate applicability of this section to reviews of future applications.
90.	Reference Verification.	This is an administrative change to include a reference to the revised section of 10 CFR Part 20 § 20.1101(b) that corresponds to the old § 20.1(c) for implementation of ALARA. For documentation of the changes to 10 CFR Part 20 see PI-23735 and Federal Register notice 56 FR 23396.

Item	Source	Description
91.	**Integrated Impact #854.**	Added a reference to 10 CFR 50.34(f) to address the addition of and provide a reference to the sections of this part of the CFRs added to this SRP section.
92.	Reference Verification.	The title for NUREG-0737 uses "clarification" not "clarifications".
93.	Reference Citation Verification.	The reference to Regulatory Guide 1.70 was deleted because it was not cited in the balance of the SRP and is not used for specific criteria or guidance. Subsequent references have been renumbered accordingly.
94.	Reference Verification and Editorial.	When using "an" in a title it is normally not capitalized.
95.	**Integrated Impact # 242**.	Added a new reference for SECY 93-087 which contains current guidance and staff positions regarding the PASS design for Evolutionary and Advanced Light Water Reactors.
96.	**Integrated Impact # 242.**	Added a new reference to the SRM for SECY 93-087 which contains current guidance and staff positions regarding the PASS design for Evolutionary and Advanced Light Water Reactors.
97.	**Integrated Impact # 243.**	A reference to Generic Letter 83-37, "NUREG-0737 Technical Specifications," was added. This generic letter addresses the staff guidance regarding an administrative program for the post-accident sampling systems.
98.	**Integrated Impact # 243.**	A reference to Generic Letter 83-36, "NUREG-0737 Technical Specifications," was added. This generic letter addresses the staff guidance regarding an administrative program for the post-accident sampling systems.
99.	**Integrated Impact # 671.**	ANSI standard ANSI N 13.1 was reaffirmed in 1993. The dates for those codes and standards cited in the reference subsection that have been reaffirmed will be indicated using the reaffirmed notation.

SRP Draft Section 9.3.2

Attachment B - Cross Reference of Integrated Impacts

Integrated Impact No.	Issue	SRP Subsections Affected
242	Revise the SRP to address the post-accident sampling requirements and guidance for the Evolutionary Advanced Light Water Reactors (ALWRs). Modifications to the existing post-accident sampling system criteria for the Evolutionary ALWRs is contained in SECY 93-087 and its associated SRM. The FSERs for the ABB-CE 80+ and the ABWR contain review findings that address the modifications described in SECY 93-087 and the associated SRM.	ACCEPTANCE CRITERIA: Revise item II.K, modify the list for the specific criteria. REVIEW PROCEDURES: Revise step 6. EVALUATION FINDINGS: Modify third paragraph. Revise the third paragraph for the section on both PWRs and BWRs. REFERENCES: Add two new references addressing SECY 93-087 and its SRM.
243	Revise the SRP to include a review of an applicants administrative program for the post-accident sampling system. Generic Letters 83-36 and 83-37 contain staff guidance on developing and maintaining an administrative program for the post-accident sampling system.	AREAS OF REVIEW: Revise step 5 to include the administrative program. REVIEW PROCEDURES: Add a new step 7. REFERENCES: Add two new references addressing Generic Letters 83-36 and 83-37.
244	Revise the SRP to address the requirements of 10 CFR 50.34(f)(2)(viii) which include individual radiation exposure limits.	ACCEPTANCE CRITERIA: Revise item II.K. REVIEW PROCEDURES: Revise step 6. EVALUATION FINDINGS: Modify the first sentence of the third paragraph.
671	Cite the reaffirmed version of ANSI N13.1-1969 which was reaffirmed in 1993.	ACCEPTANCE CRITERIA: Modify step 3.b of the specific criteria. REVIEW PROCEDURES: Modify step 2. REFERENCES: Revise reference to ANSI N13.1-1969(R93).
854	Revise the SRP to address the requirements of 10 CFR 50.34(f)(2)(xxvi) as clarified in NUREG-0737 item III.D.1.1. This issue addresses the review of provisions for leakage control and detection in the design of systems outside containment that contain (or might contain) radioactive materials following an accident.	ACCEPTANCE CRITERIA: Add a new step L. REVIEW PROCEDURES: Add a new step 8. EVALUATION FINDINGS: Modify the third paragraph. Add a new item 4 to the third paragraph for the section on both PWRs and BWRs.