Low-Priority Generic Issues

From 1983 to 1999, the Generic Issues (GI) program consisted of six separate and distinct steps: identification, prioritization, resolution, imposition, implementation, and verification. During this time, four priority rankings were used in the prioritization step: HIGH, MEDIUM, LOW, and DROP. A LOW-priority ranking meant that no safety concerns demanding at least MEDIUM-priority attention were involved and that little or no prospect existed for safety improvements that were both substantial and worthwhile. When the prioritization process resulted in a LOW-priority ranking for an issue, approval of this ranking by the responsible Office Director signified that the issue had been eliminated from further pursuit. However, in accordance with Staff Requirements Memorandum 871021A, the staff has periodically conducted a review of the LOW-priority generic issues to determine whether any new information existed that would necessitate reassessment of the original prioritization evaluations.

Staff completed a final review of the LOW-priority issues in August 2010. For disposition of the LOW-priority issues, staff evaluated each of the remaining 13 LOW priority ranking issues to confirm that 1) these issues are addressed by current U.S. Nuclear Regulatory Commission's (NRC's) regulatory requirements, guidance, or oversight and 2) the operating experience has not indicated a change in the safety significance of these issues. Based on the reviews and evaluations of the LOW priority ranking issues documented in this report, staff recommends changing the status of remaining LOW-priority issues and dropping them from further pursuit. By changing the status of the LOW priority ranking issues, these issues will no longer be periodically assessed.

For each issue, a historical background of the identification and prioritization of the issue is presented. After the historical background, an overview of the NRC regulatory framework and any relevant operating experience related to the issues are discussed. Finally, a discussion is provided to demonstrate the application of the NRC regulatory framework to each issue and to support its disposition.

1. ITEM I.F.2: DEVELOP MORE DETAILED QA CRITERIA

1.1 Overview

Item I.F.2, “Develop More Detailed QA Criteria,” of the TMI Action Plan was proposed to improve the quality assurance (QA) program for plants’ design, construction, and operations. Item I.F.2 consists of 11 detailed QA criteria, which established 11 generic issues under Item I.F.2. Four of these issues were RESOLVED when new requirements were established with changes made in July 1981 to Chapter 17 of the Standard Review Plan (SRP) and the remaining seven items were given a LOW-priority ranking in the main report of NUREG-0933, published in November 1983. Staff conducted a review of the remaining seven issues in 2010 to determine whether any new information would necessitate reassessment of original prioritization evaluations. Staff determined that the operating experience has not indicated a change in the safety significance of these issues. In addition, staff verified that the current NRC regulatory requirements or guidance address these issues and identified applicable regulatory framework as presented below. Because these items have been addressed by the existing regulations and the operating experience has not raised the significance of these issues, the NRC staff recommends changing the status of these items and dropping them from further pursuit.

Item I.F.2 (1): Assure the Independence of the Organization Performing the Checking Function

> Related regulatory framework: 10 CFR 50.34(f)(3)(iii)(A) and Section 17.5 of the SRP.

Item I.F.2 (4): Establish Criteria for Determining QA Requirements for Specific Classes of Equipment

> Related regulatory framework: Criterion II, "Quality Assurance Program," of 10 CFR Part 50, Appendix B, and the following Subparts of ASME NQA-1-1994 Edition: Subpart 2.4, “Installation, Inspection, and Testing Requirements for Power, Instrumentation, and Control Equipment at Nuclear Facilities,” Subpart 2.5, “Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete, Structural Steel, Soils, and Foundations for Nuclear Power Plants,” Subpart 2.7, “Quality Assurance Requirements of Computer Software for Nuclear Facility Applications,” Subpart 2.8, “Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems for Nuclear Power Plants.”

Item I.F.2 (5): Establish Qualification Requirements for QA and QC Personnel

> Related regulatory framework: Criterion II, "Quality Assurance Program," of 10 CFR Part 50 Appendix B, Regulatory Guide (RG) 1.8, “Qualification and Training of Personnel for Nuclear Power Plants,” Revision 3, 10 CFR 50.34(f)(3)(iii)(E), and Section 17.5 of the SRP.

Item I.F.2 (7): Clarify That the QA Program Is a Condition of the Construction Permit and Operating License

Related regulatory framework: 10 CFR 50.54(a)(1), 10 CFR 52.17(a)(1)(xi), 10 CFR 52.47(a)(19), 10 CFR 52.79(a)(25), 10 CFR 50.54(a)(4), 10 CFR 50.54(a)(4)(i)-(iv), and Section 17.5 of the SRP.

Item I.F.2 (8): Compare NRC QA Requirements with Those of Other Agencies

SECY-03-0117, "Approaches for Adopting More Widely Accepted International Quality Standards," July 9, 2003.

Item I.F.2 (10): Clarify Requirements for Maintenance of "As-Built" Documentation

Related Regulatory Framework: Criterion VI, "Document Control," and Criterion XVII, "Quality Assurance Records," of 10 CFR Part 50, Appendix B, ANSI/ASME-NQA-1, 10 CFR 50.34(f)(3)(iii)(G), and Section 17.5 of the SRP.

Item I.F.2 (11): Define Role of QA in Design and Analysis Activities

Related Regulatory Framework: Criterion III, "Design Control," of 10 CFR Part 50, Appendix B, 10 CFR 50.34(f)(3)(iii)(H), and Section 17.5 of the SRP.

Section 1.2 describes a historical background of the identification and prioritization of these issues. Section 1.3 presents an overview of the NRC regulatory framework for QA. Finally, in Section 1.4 a discussion is provided to demonstrate the application of the NRC regulatory framework for QA to each issue and to support their disposition.

1.2 Background

1.2.1 Description

The overall objective of this Three Mile Island (TMI) Action Plan[1] item was the improvement of the quality assurance (QA) program for design, construction, and operations to provide greater assurance that plant design, construction, and operational activities were conducted in a manner commensurate with their importance to safety. Several systems important to the safety of TMI-2 were not designed, fabricated, and maintained at a level equivalent to their safety importance. This condition existed at other plants and resulted primarily from the lack of clarity in NRC guidance. This situation and other problems relating to the QA organization, authority, reporting, and inspection were identified by the various TMI accident investigations and inquiries.[1]

The intent of this item was to provide more explicit and detailed criteria concerning the elements that, in general, were found in well-conducted QA programs. Providing these more detailed criteria was expected to result in the establishment of QA programs of the caliber desired. As stated in NUREG-0933, "Resolution of Generic Safety Issues,"[2] implementation of such programs would result in the detection of deficiencies in design, construction, and operation.

1.2.2 Possible Solutions

In NUREG-0933[2], staff proposed more detailed QA criteria for design, construction, and operations with the following considerations:

(1) Assure the independence of the organization performing the checking functions from the organization responsible for performing the tasks. For the construction phase, consider options for increasing the independence of the QA function. Include an option to require that licensees perform the entire quality assurance/quality control (QA/QC) function at construction sites. Consider using the third-party concept for accomplishing the NRC review and audit and making the QA/QC personnel agents of NRC. Consider using the Institute of Nuclear Power Operations to enhance QA/QC independence.

(2) Include the QA personnel in the review and approval of plant operational maintenance and surveillance procedures and quality-related procedures associated with design, construction, and installation.

(3) Include the QA personnel in all activities involved in design, construction, installation, preoperational and startup testing, and operation.

(4) Establish criteria for determining QA requirements for specific classes of equipment such as instrumentation, mechanical equipment, and electrical equipment.

(5) Establish qualification requirements for QA and QC personnel.

(6) Increase the size of the licensees' QA staff.

(7) Clarify that the QA program is a condition of the construction permit and operating license and that substantive changes to an approved program must be submitted to NRC for review.

(8) Compare NRC QA requirements with those of other agencies (i.e., NASA, FAA, DOD) to improve NRC requirements.

(9) Clarify organizational reporting levels for the QA organization.

(10) Clarify requirements for maintenance of "as built" documentation.

(11) Define the role of QA in design and analysis activities. Obtain views on prevention of design errors from licensees, architect-engineers, and vendors.

1.2.3 Priority Determination

A priority determination was made of the benefit of the above 11 items in improving QA.

Staff stated in NUREG-0933[2] that "while the QA improvement program could result in the establishment of an improved QA organizational structure at many plants, the results depended heavily upon management acceptance. Lack of program implementation and management acceptance, rather than inadequate criteria as suggested by this issue, were the primary causes of deficiencies in QA. Increasing the detail of the QA criteria had little potential for improving the quality of design, construction, or operation and, therefore, risk." Items I.F.2(2), I.F.2(3), I.F.2(6), and I.F.2 (9), which addressed the concern stated above, were RESOLVED and included in the July 1981 revision to Chapter 17 of the SRP.[3]

It was believed that the issue of QA in nuclear power plants should be a high priority. However, the issue and solutions to QA deficiency as described herein (except for the completed issues I.F.2(2), I.F.2(3), I.F.2(6) and I.F.2(9)) failed to address the problem of management acceptance of QA programs. Hence, the residual items (I.F.2(1), I.F.2(4), I.F.2(5), I.F.2(7), I.F.2(8), I.F.2(10), I.F.2(11)) were given a low priority.

1.3 NRC Regulatory Framework

1.3.1 Regulatory Background

The regulatory framework for quality assurance is established by 10 CFR Part 50 Appendix B. [4] The 18 criteria of Appendix B[4] are implemented through quality assurance program descriptions, regulatory guides, and consensus standards such as ANSI N45.2,[5] "Quality Assurance Requirements for Nuclear Facilities," and ASME NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications." [6] Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)," [7] describes a method acceptable to the NRC staff for complying with the provisions of Appendix B[4] with regard to establishing and implementing the requisite quality program. It states that ASME/ANSI NQA-1-1983[6] is an acceptable method for complying with the pertinent quality requirements of Appendix B.[4]

Since the late 1980s, the staff has completed several initiatives to improve the efficiency and effectiveness of the regulatory framework for quality assurance. In 1989, the staff issued Generic Letter (GL) 89-02, "Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products," [8] and in 1991, GL 91-05, "Licensee Commercial Grade Procurement and Dedication Programs." [9] These generic letters documented the staff's position on the process for the procurement and dedication of commercial-grade items. In GL 89-02,[8] the staff conditionally endorsed the June 1988 EPRI NP-5652, "Guideline for the Utilization of Commercial-Grade Items in Safety Related Applications (NCIG-07)."[10] Historically, the commercial-grade dedication process has proven to be an effective method for procuring items from the commercial market and demonstrating their suitability for use in safety-related applications.

In the early 1990s, the staff facilitated the change-control process for administrative controls described in RG 1.33, "Quality Assurance Program Requirements (Operation),"[11] by allowing these controls to be relocated from the technical specifications to the quality assurance program. In 1998, the staff issued RG 1.176, "An Approach for Plant-Specific, Risk-Informed Decision Making: Graded Quality Assurance,"[12] that defines a method acceptable to the staff for grading the requirements of Appendix B.[4] Subsequently, the staff recommended in SECY-98-300, "Options for Risk-Informed Revisions to 10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities,"[13] that risk-informed approaches to the application of special treatment requirements be developed. In November 2004, NRC promulgated 10 CFR 50.69, ‘‘Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors,’’ to permit power reactor licensees and license applicants to implement an alternative regulatory framework with respect to ‘‘special treatment’’ where special treatment refers to those requirements that provide increased quality assurance beyond normal industrial practices that structures, systems, and components (SSCs) perform their design-basis functions. In support of 10 CFR 50.69, the staff issued RG 1.201, ‘‘Guidelines for Categorizing Structures, Systems and Components in Nuclear Power Plants According to Their Safety Significance,’’[14] in January 2006 for trial use. The staff withdrew RG 1.176 after publishing the new framework, consisting of the rule along with RG 1.201.[14]

In 1999, the Commission amended 50.54(a) to allow licensees to make certain changes to their quality assurance programs without prior NRC review. This includes changes such as the use of a QA standard approved by NRC that is more recent that the QA standard in the licensee's current QA program, the use of a quality assurance alternative or exception approved by an NRC safety evaluation (provided that the basis of the NRC approval is applicable to the licensee's facility), and generic organizational changes. The number of license amendments and changes to QA programs has declined as a result of these initiatives. In a Nuclear Energy Institute (NEI) August 15, 2000, letter to the staff, NEI stated, "The direct final rule was promulgated 13 months prior to the workshop, providing adequate time for the industry to ascertain the short-term worth of the rule in reducing unnecessary burden while maintaining the integrity of a comprehensive QA program. It was evident to the industry participants during the course of the workshop that the direct final rule has been beneficial. A separate rulemaking on 10 CFR 50.54(a) is not needed since QA special treatment requirements are being addressed under the Risk-Informing Part 50, Option 2 initiative."

The NRC staff has reviewed risk-informed applications in many areas. In this respect, the staff has been successful in developing and implementing a regulatory means for factoring risk insights into the current regulatory framework. In addition, the staff has taken steps to reduce the scope of equipment subject to the requirements of Appendix B.[4] Appendix B[4] contains provisions for applying a graded approach to quality assurance according to a component's importance to safety. The process explained in 10 CFR 50.69 recognizes that components may differ in importance and implements a graded approach based on a risk-informed categorization process. This approach significantly reduces the scope of SSCs subject to special treatment requirements including QA programmatic requirements.

1.3.2 QA Program Commitments (RG and GLs)

As stated in Section 17.5 of the SRP,[3] “Quality Assurance Program Description - Design Certification, Early Site Permit and New License Applicants,” applicants or holders commit to the most recent revision of the RGs and GLs listed below.

a. RG 1.26, “Quality Group Classifications and Standards for Water-Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants” (03/2007).[15]

b. RG 1.29, “Seismic Design Classification.”[16]

c. RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," Revision 1 (03/2007).[17]

d. GL 89-02, "Actions to Improve the Detection of Counterfeit and Fraudulently Marked Products."[8]

e. GL 91-05, "Licensee Commercial-Grade Dedication Programs"[9]

Exceptions or alternatives to the specific criteria in any of these RGs and GLs may be proposed by applicants or holders provided adequate justification is provided.

1.3.3 QA Program Commitments (Standards)

In addition to RGs and GLs listed above, applicants or license holders commit to the standards listed below. Exceptions or alternatives to the specific criteria in any of these standards may be proposed by applicants or license holders provided adequate justification is provided.

a. Subpart 2.2, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Nuclear Power Plants," ASME NQA-1-1994 Edition.[6]

b. Subpart 2.4, "Installation, Inspection, and Testing Requirements for Power, Instrumentation, and Control Equipment at Nuclear Facilities," ASME NQA-1-1994 Edition.[6]

c. Subpart 2.5, "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete, Structural Steel, Soils, and Foundations for Nuclear Power Plants," ASME NQA-1-1994 Edition.[6]

d. Subpart 2.7, "Quality Assurance Requirements of Computer Software for Nuclear Facility Applications," ASME NQA-1-1994 Edition.[6]

e. Subpart 2.8, "Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems for Nuclear Power Plants," ASME NQA-1-1994 Edition.[6]

f. Subpart 2.15, "Quality Assurance Requirements for Hoisting, Rigging, and Transporting Items for Nuclear Power Plants," ASME NQA-1-1994 Edition.[6]

g. Subpart 2.20, "Quality Assurance Requirements for Subsurface Investigations for Nuclear Power Plants," ASME NQA-1-1994 Edition.[6]

h. Nuclear Information and Records Management Association, Inc. (NIRMA) Technical Guide (TG) 11-1998, "Authentication of Records and Media."[18]

i. NIRMA TG 15-1998, "Management of Electronic Records."[19]

j. NIRMA TG 16-1998, “Software Configuration Management and Quality Assurance.”[20]

k. NIRMA TG 21-1998, “Electronic Records Protection and Restoration.”[21]

l. Section 4, “Storage, Preservation, and Safekeeping,” of Supplement 17S-1, “Supplementary Requirements for Quality Assurance Records,” NQA-1-1994 Edition.[6]

1.3.4 Publications

The following tables provide a list of NRC publications related to the QA program.

Table 1.1. List of Regulatory Guides Related to the QA Program

Guide Number	Title	Rev.	Publish Date	Last Evaluation
1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants	4	03/2007	03/2007
1.28	Quality Assurance Program Requirements (Design and Construction)	3	08/1985	06/2009
1.29	Seismic Design Classification	4	03/2007	03/2007
1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment (Safety Guide 30)	--	08/1972	06/2008
1.33	Quality Assurance Program Requirements (Operation)	2	02/1978	—
1.37	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants	1	03/2007	03/2007
1.38	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants	2	05/1977	04/2008
1.94	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants	1	04/1976	04/2008
1.116	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems	0-R	05/1977	03/2008

Table 1.2. List of the SRP Sections Related to the QA Program

Section	Title	Rev.	Date Updated
17.1	Quality Assurance During the Design and Construction Phases		
		Rev. 2	07/1981
		Rev. 1	02/1979
		Rev. 0	11/1975
17.2	Quality Assurance During the Operations Phase		
		Rev. 2	07/1981
		Rev. 1	02/1979
		Rev. 0	11/1975
17.3	Quality Assurance Program Description		
		Rev. 0	08/1990
17.4	Reliability Assurance Program (RAP)	Initial Issuance	03/2007
		Draft Rev. 0	06/1996
17.5	Quality Assurance Program Description - Design Certification, Early Site Permit and New License Applicants	Initial Issuance	03/2007
		Draft Rev. 0	01/2006
17.6	Maintenance Rule	Rev. 1	08/2007
		Initial Issuance	03/2007

1.4 Assessment and Conclusion

1.4.1 Item I.F.2 (1): Assure the Independence of the Organization Performing the Checking Function

This item was evaluated in Item I.F.2 and was determined to be a LOW-priority issue in the main report of NUREG-0933,[2] published in November 1983. In 1998, consideration of new information on the lack of independence in the checking function submitted by Region IV in April 1997 did not change this conclusion.[22]

According to 10 CFR 50.34(f)(3)(iii)(A), "each applicant for a light-water-reactor construction permit or manufacturing license whose application was pending as of February 16, 1982" in addition to "each applicant for a design certification, design approval, combined license, or manufacturing license under part 52" of 10 CFR needs to "establish a quality assurance (QA) program based on consideration of: (A) Ensuring independence of the organization performing checking functions from the organization responsible for performing the functions." In addition, Section 17.5 of the SRP[3] states that "the QA program requires independence between the organization performing checking functions from the organization responsible for performing the functions. (This provision applies to DC applicant, ESP, and construction QA programs. This provision is not applicable to design reviews/verifications.)"

The NRC staff concludes that this item has been adequately addressed and therefore recommends changing the status of this issue and dropping this item from further pursuit.

1.4.2 Item I.F.2 (4): Establish Criteria for Determining QA Requirements for Specific Classes of Equipment

This item was evaluated in Item I.F.2 above and was determined to be a LOW-priority issue in the main report of NUREG-0933[2] published in November 1983.

Criterion II, "Quality Assurance Program," of 10 CFR Part 50, Appendix B,[4] states that "The applicant shall identify the structures, systems, and components to be covered by the quality assurance program and the major organizations participating in the program, together with the designated functions of these organizations. The quality assurance program shall provide control over activities affecting the quality of the identified structures, systems, and components, to an extent consistent with their importance to safety." In addition, as discussed earlier, applicants or license holders commit to the standards below that identify requirements for specific classes of equipment.

- Subpart 2.4, "Installation, Inspection, and Testing Requirements for Power, Instrumentation, and Control Equipment at Nuclear Facilities," ASME NQA-1-1994 Edition.
- Subpart 2.5, "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete, Structural Steel, Soils, and Foundations for Nuclear Power Plants," ASME NQA-1-1994 Edition.
- Subpart 2.7, "Quality Assurance Requirements of Computer Software for Nuclear Facility Applications," ASME NQA-1-1994 Edition.
- Subpart 2.8, "Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems for Nuclear Power Plants," ASME NQA-1-1994 Edition.

Based on the review of NRC's regulations related to this issue presented above, staff concludes that Item I.F.2 (4) is adequately addressed by the existing regulations. Therefore, staff recommends changing the status of Item I.F.2 (4) and dropping this item from further pursuit.

1.4.3 Item I.F.2 (5): Establish Qualification Requirements for QA and QC Personnel

This item was evaluated in Item I.F.2 above and was determined to be a LOW-priority issue in the main report of NUREG-0933[2] published in November 1983.

Criterion II, "Quality Assurance Program," of 10 CFR Part 50 Appendix B[4] establishes requirements for training of the personnel: "The program shall provide for indoctrination and training of personnel performing activities affecting quality as necessary to assure that suitable proficiency is achieved and maintained." In addition, RG 1.8, "Qualification and Training of Personnel for Nuclear Power Plants,"[23] Revision 3, provides guidance that is acceptable to the NRC staff regarding qualifications and training for nuclear power plant personnel. This RG endorses ANSI/ANS-3.1-1993, "Selection, Qualification, and

Training of Personnel for Nuclear Power Plants,"[24] with certain clarifications, additions, and exceptions.

Moreover, 10 CFR 50.34(f)(3)(iii)(E) states that "each applicant for a light-water-reactor construction permit or manufacturing license whose application was pending as of February 16, 1982" in addition to "each applicant for a design certification, design approval, combined license, or manufacturing license under part 52" of 10 CFR needs to "establish a quality assurance (QA) program based on consideration of: … (E) establishing qualification requirements for QA and QC personnel." Finally, Section 17.5 of the SRP[3] describes the SRP acceptance criteria for "Training and Qualification Criteria - Quality Assurance."

Based on the review of NRC's regulations related to this issue presented above, the staff concludes that Item I.F.2 (5) is adequately addressed by the existing regulations. Therefore, the staff recommends changing the status of Item I.F.2 (5) and dropping this item from further pursuit.

1.4.4 Item I.F.2 (7): Clarify That the QA Program Is a Condition of the Construction Permit and Operating License

This item was evaluated in Item I.F.2 above and was determined to be a LOW-priority issue in the main report of NUREG-0933[2] published in November 1983.

10 CFR 50.54(a)(1) clearly states implementation of the QA program as a condition in every nuclear power reactor operating license issued under 10 CFR 50: "Each nuclear power plant or fuel reprocessing plant licensee subject to the quality assurance criteria in appendix B[4] of this part shall implement, under § 50.34(b)(6)(ii) or § 52.79 of this chapter, the quality assurance program described or referenced in the safety analysis report, including changes to that report. However, a holder of a combined license under part 52 of this chapter shall implement the quality assurance program described or referenced in the safety analysis report applicable to operation 30 days prior to the scheduled date for the initial loading of fuel." In addition, 10 CFR 50.54(a)(1) is also a condition in every combined license issued under 10 CFR 52. Finally, 10 CFR 52.17(a)(1)(xi), 10 CFR 52.47(a)(19), and 10 CFR 52.79(a)(25) outline the QA program requirements for applicants of Early Site Permits (ESP), Standard Design Certifications and Combined Licenses, respectively. SRP[3] Section 17.5 outlines a standardized QA program for DC, ESP, CP, OL and COL applicants and holders.

Moreover, this issue specifies that "substantive changes to an approved program must be submitted to NRC for review." This part of the issue is also addressed by 10 CFR 50.54(a)(4) that states "Changes to the quality assurance program description that do reduce the commitments must be submitted to the NRC and receive NRC approval prior to implementation." 10 CFR 50.54(a)(4)(i)-(iv) outlines the process to make these changes.

Based on the review of NRC's regulations related to this issue presented above, staff concludes that Item I.F.2 (7) is adequately addressed by the existing regulations. Therefore, staff recommends changing the status of Item I.F.2 (7) and dropping this item from further pursuit.

1.4.5 Item I.F.2 (8): Compare NRC QA Requirements with Those of Other Agencies

This item was evaluated in Item I.F.2 above and was determined to be a LOW-priority issue in the main report of NUREG-0933[2] published in November 1983. In July 9, 2009, results of the staff's effort to review international quality assurance standards against the existing 10 CFR Part 50 Appendix B[4] framework were reported by issuance of SECY-03-0117.[25] In addition, approaches for adopting international quality standards for safety-related components in nuclear power plants into the existing regulatory framework were assessed. SECY-03-0117[25] also reviewed existing NRC quality assurance requirements and efforts to improve their effectiveness and efficiency. The staff concluded in SECY-03-0117[25] that considerable actions had already been taken or were in progress to reduce unnecessary regulatory burden on licensees resulting from compliance with Appendix B[4] requirements. In addition, the proposed 50.69 risk-informed rulemaking would provide a more efficient and effective regulatory process while continuing to maintain safety. The staff evaluation of the differences between Appendix B[4] and ISO 9001 is summarized in the attachment to SECY-03-0117.[25]

The staff concludes that the analysis presented in SECY-03-0117[25] has addressed Item I.F.2 (8) adequately. Therefore, the staff recommends changing the status of Item I.F.2 (8) and dropping this item from further pursuit.

1.4.6 Item I.F.2 (10): Clarify Requirements for Maintenance of "As-Built" Documentation

This item was evaluated in Item I.F.2 above and was determined to be a LOW-priority issue in the main report of NUREG-0933[2] published in November 1983.

Criterion VI, "Document Control," and Criterion XVII, "Quality Assurance Records," of 10 CFR Part 50, Appendix B,[4] establish requirements for issuing, identifying, and retrieving QA records. In addition, NRC-accepted practices for the collection, storage, and maintenance of QA records for nuclear power plants, independent storage of spent nuclear fuel and high-level radioactive waste facilities, special nuclear materials, packaging and transportation of radioactive materials, and gaseous diffusion plants are described in ANSI/ASME-NQA-1.[6]

Criterion VI, "Document Control," of 10 CFR Part 50, Appendix B,[4] describes the requirements to control changes in documents: “Measures shall be established to control the issuance of documents, such as instructions, procedures, and drawings, including changes thereto, which prescribe all activities affecting quality. These measures shall assure that documents, including changes, are reviewed for adequacy and approved for release by authorized personnel and are distributed to and used at the location where the prescribed activity is performed. Changes to documents shall be reviewed and approved by the same organizations that performed the original review and approval unless the applicant designates another responsible organization.”

Moreover, 10 CFR 50.34(f)(3)(iii)(G) states that “each applicant for a light-water-reactor construction permit or manufacturing license whose application was pending as of February 16, 1982” in addition to “each applicant for a design certification, design approval, combined license, or manufacturing license under part 52” of 10 CFR needs to “establish a quality assurance (QA) program based on consideration of: … (G) establishing procedures for maintenance of "as-built" documentation." Finally, Section 17.5 of the SRP[3] states that a "program is required to be established to control the development, review, approval, issue, use, and revision of documents." This section includes as-built drawings as one of the examples of controlled documents: "Examples of controlled documents include design drawings, as-built drawings, engineering calculations …"

Based on the review of NRC's regulations related to this issue presented above, the staff concludes that Item I.F.2 (10) is adequately addressed by the existing regulations. Therefore, the staff recommends changing the status of Item I.F.2 (10) and dropping this item from further pursuit.

1.4.7 Item I.F.2 (11): Define Role of QA in Design and Analysis Activities

This item was evaluated in Item I.F.2 above and was determined to be a LOW-priority issue in the main report of NUREG-0933[2] published in November 1983.

Criterion III, "Design Control," of 10 CFR Part 50, Appendix B,[4] describes the requirements of the program for the design of items. As explained in this criterion, measures should be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. In addition, these measures should include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. The design control measures provide for verifying or checking the adequacy of design and are applied to items such as the reactor physics, stress, thermal, hydraulic, and accident analyses; compatibility of materials; accessibility for inservice inspection, maintenance, and repair; and delineation of acceptance criteria for inspections and tests.

Moreover, 10 CFR 50.34(f)(3)(iii)(H) states that "each applicant for a light-water-reactor construction permit or manufacturing license whose application was pending as of February 16, 1982" in addition to "each applicant for a design certification, design approval, combined license, or manufacturing license under part 52" of 10 CFR needs to "establish a quality assurance (QA) program based on consideration of: … (H) providing a QA role in design and analysis activities." Finally, Section 17.5 of the SRP[3] states that "The QA role in design and analysis activities is defined. Design documents are reviewed by individuals knowledgeable and qualified in QA to ensure the documents contain the necessary QA requirements. (This applies to DC applicants, ESP, and construction QA programs.)"

Based on the review of the NRC's regulations related to this issue presented above, the staff concludes that Item I.F.2 (11) is adequately addressed by the existing regulations. Therefore, the staff recommends changing the status of Item I.F.2 (11) and dropping this item from further pursuit.

2. ITEM III.D.2.1: RADIOLOGICAL MONITORING OF EFFLUENTS

2.1 Overview

Item III.D.2.1, "Radiological Monitoring of Effluents," of TMI Action Plan was proposed to improve public radiation protection by providing assurance that all possible accident effluent-release pathways are monitored and that monitors will perform properly under accident conditions. Item III.D.2.1 consists of the following three issues, which were given a LOW-priority ranking in the main report of NUREG-0933 published in 1983. Staff conducted a review of these issues in 2010 to determine whether any new information would necessitate reassessment of original prioritization evaluations. Staff determined that the operating experience has not indicated a change in the safety significance of these issues. In addition, based on the review of NRC's regulations, staff determined that although some specific requirements that were proposed by these issues have not been established, the overall objectives of these issues are met by the existing regulations. Moreover, the low safety significance of the issue does not warrant further actions to evaluate and implement some of the proposed solutions. Based on the review of NRC's regulations related to these issues presented below and the fact that the operating experience has not raised the significance of these issues, staff recommends changing the status of these issues and dropping them from further pursuit.

Item III.D.2.1(1): Evaluate the Feasibility and Perform a Value-Impact Analysis of Modifying Effluent-Monitoring Design Criteria

> Related regulatory frame: Criterion 64, "Monitoring radioactivity releases," of 10 CFR Part 50, Appendix A, 10 CFR 50.34(f)(2)(xvii)(E), 10 CFR 50.34(f)(2)(xxviii), 10 CFR 50.34(f)(2)(xxvii), and Section 11.5 of the SRP.

Item III.D.2.1(2): Study the Feasibility of Requiring the Development of Effective Means for Monitoring and Sampling Noble Gases and Radioiodine Released To the Atmosphere

> Related regulatory frame: Criterion 64, "Monitoring radioactivity releases," of 10 CFR Part 50, Appendix A, and 10 CFR 50.34(f)(2)(xvii)(E).

Item III.D.2.1(3): Revise Regulatory Guides

> RG 1.21, "Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste," Rev. 2 (June 2009), Section 11.5 of the SRP (March 2007), and RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Rev. 4 (June 2006)

Section 2.2 describes a historical background of the identification and prioritization of radiological monitoring of effluents issues. Section 2.3 presents an overview of the NRC regulatory framework for the radiological effluent control and monitoring. Finally, in

Section 2.4 a discussion is provided to demonstrate the application of the NRC regulatory framework to these issues and to support their disposition.

2.2 Background

2.2.1 Description

The objective of Task III.D.2 was to improve public radiation protection in the event of a nuclear power plant accident by improving: (1) radioactive effluent monitoring; (2) the dose analysis for accidental releases of radioiodine, tritium, and carbon-14; (3) the control of radioactivity released into the liquid pathway; (4) the measurement of offsite radiation doses; and (5) the ability to rapidly determine offsite doses from radioactivity release by meteorological and hydrological measurements so that population-protection decisions can be made appropriately.[26] Item III.D.2.1 consists of three parts that are combined and evaluated in NUREG-0933[2] together. The following three parts of this item were given a LOW-priority ranking in the main report of NUREG-0933[2] published in 1983.

- Item III.D.2.1(1): Evaluate the Feasibility and Perform a Value-Impact Analysis of Modifying Effluent-Monitoring Design Criteria.
- Item III.D.2.1(2): Study the Feasibility of Requiring the Development of Effective Means for Monitoring and Sampling Noble Gases and Radioiodine Released To the Atmosphere.
- Item III.D.2.1(3): Revise Regulatory Guides.

This TMI Action Plan[1] item required development and implementation of acceptance criteria for monitors used to evaluate effluent releases under accident and post-accident conditions. Criteria were to be developed for pathways to be monitored (stack, plant vent, steam dump vents) as well as for monitoring instrumentation. This was seen to encompass the requirements in NUREG-0578,[27] Recommendation 2.1.8-b, and Appendix 2 to NUREG-0654.[28]

Liquid effluents were not envisioned as posing a major release pathway because licensees typically had installed, or were installing, adequate storage capacity to prevent discharges. Consequently, existing liquid effluent monitoring systems were considered adequate.

The overall objective of Items III.D.2.1(1), III.D.2.1(2), and III.D.2.1(3) was "to provide assurance that all possible accident effluent-release pathways are monitored and that monitors will perform properly under accident conditions."[29] More specifically, under Item III.D.2.1(1), the staff would evaluate "the feasibility and perform a value-impact analysis of modifying effluent-monitoring design criteria." A number of factors were introduced in NUREG-0660[29] for evaluation. Under Item III.D.2.1(2), staff would study the feasibility of requiring the development of effective means for monitoring and sampling noble gases and radioiodine released to the atmosphere during a pressurized-water reactor (PWR) steam dump.

This issue had no impact on core-melt accident frequency.

2.2.2 Possible Solution

As explained in NUREG-0933,[2] the envisioned monitoring system would provide automatic online analysis of airborne effluents including isotopic analyses of particulate, radioiodine, and gas samples. To prevent saturation of detectors, an automatic sample cartridge changeout feature would be included. The system would include microprocessor control and real-time readouts and would be located in a low post-accident background area. The sampling system would be designed to provide a representative sample under anticipated accident release conditions.

A PWR steam-dump sampling and monitoring system would be provided for PWR safety relief and vent valves. Such a system might consist of a noble gas monitor and a radioiodine sampling and monitoring system. The features of such a system would be similar to the above-described airborne monitor with two notable differences: (1) the system would be required to function in a very high humidity (steam-air mixture) environment and (2) operation would only be required during actual steam venting. Because such venting is usually of a short-term or intermittent duration, the monitoring system activation could be keyed to the opening of the vents.

2.2.3 Priority Determination

It was assumed in the priority determination presented in NUREG-0933[2] that improved radiological monitoring of airborne effluent would result in a reduction of public risk. In this section, a summary of the prioritization analysis performed in NUREG-0933[2] is presented.

2.1.3.1 Frequency/Consequence Estimate

The magnitude of public risk reduction attributable to improved radiological monitoring of airborne effluents was not certain, but it was estimated to range from 0 to 1 percent.[30]

By implementation of radiological monitoring requirements at the time of prioritization analysis, execution of sample collection and analysis procedures during design basis conditions was estimated to require between 2 to 3 hours. During this time, radioiodine and particulate releases would be estimated based on computer-modeled interpretation of noble gas monitor readings or on previous post-accident containment atmosphere analysis results, if such results were available. Public protective action recommendations would be made based on modeled estimates rather than actual effluent data. It was assumed that these recommendations would err on the conservative side (e.g., evacuate when not really required) due to the conservatism built into the modeled source terms for radioiodine and particulate releases.

Requiring licensees to have more sophisticated airborne effluent monitors would reduce the time required for obtaining actual radioiodine and particulate release data to 15 minutes and essentially eliminate reliance on conservative theoretical release models extrapolated from noble gas monitor readings. As projected by the possible solution, real-time isotopic monitoring would save nearly 2 hours in arriving at realistic protective action recommendations based on actual releases.

Under these circumstances, the public risk reduction would be directly attributed to the decrease in public radiation exposure that would result from a more rapid assessment of the radioactive releases (about a 2-hour savings in analysis time). In addition, public risk may be reduced as a result of nonevacuation. The need for evacuation (presumed to exist if release knowledge was based only on noble gas monitor data) could be eliminated as a result of better knowledge of the isotopic releases. Nonevacuation would result in less evacuation-related risks (e.g., traffic accidents), the avoidance of which may outweigh the radiation exposure received. However, it was assumed that the public risk reduction would result primarily from the first effect (decrease in exposure due to more rapid assessment).

As the staff concluded in NUREG-0933,[2] “Based on the risk reduction potential and value/impact score, the issue was given a LOW priority ranking (see Appendix C) in November 1983. In NUREG/CR-5382,[31] it was concluded that consideration of a 20-year license renewal period could change the ranking of the issue to medium priority. Further prioritization, using the conversion factor of $2,000/man-rem approved[32] by the Commission in September 1985, resulted in an impact/value ratio (R) of $24,390/man-rem which did not change the priority ranking.”

2.3 NRC Regulations and Policies

2.3.1 Radiological Effluent Control Program

The following regulations and design criteria establish the regulatory basis for the radiological effluent control program. Collectively, these regulations require that an environmental monitoring program be established and implemented to obtain data on measurable levels of radiation and radioactive materials. The Annual Radiological Environmental Operating Report provides summaries of the data, interpretations, and analyses of trends of the results.

1. 10 CFR 20.1501, “Surveys,”[33] requires surveys that may be necessary and are reasonable to evaluate the magnitude and extent of potential radiological hazards. In 10 CFR Part 20, “Standards for Protection against Radiation,” “survey” is defined as an evaluation of the radiological conditions and potential hazards related to radioactive material or other sources of radiation including (1) a physical survey of the location of radioactive material and (2) measurements or calculations of levels of radiation or concentrations or quantities of radioactive material present. The design objectives set out in 10 CFR Part 50, Appendix I, provide numerical guidance on limiting conditions for operation for light-water-cooled nuclear power reactors to meet the requirement that radioactive materials in effluents discharged to unrestricted areas be kept as low as is reasonably achievable (ALARA).

2. 10 CFR 50.36a, “Technical Specifications on Effluents from Nuclear Power Reactors,” requires establishing technical specifications with procedures and controls over effluents including reporting (1) the quantity of each of the principal radionuclides discharged to unrestricted areas in liquid and gaseous effluents and (2) other information used to estimate the maximum potential annual radiation doses to the public from radioactive effluents.

3. 10 CFR 20.1302, "Compliance with Dose Limits for Individual Members of the Public," establishes requirements for surveys in the unrestricted and controlled areas and for radioactive materials in effluents discharged to unrestricted and controlled areas. The purpose of these surveys is to demonstrate compliance with the dose limits of 10 CFR 20.1301, "Dose Limits for Individual Members of the Public." Although 10 CFR 20.1302(b)(2) provides a second method of demonstrating compliance with dose limits for individual members of the public, nuclear power plant technical specifications essentially require use of 10 CFR 20.1302(b)(1) to determine the total effective dose equivalent to the individual likely to receive the highest dose. This requirement is based on actual, realistic exposure pathways to a real individual. (See also RG 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Demonstrating Compliance with 10 CFR Part 50, Appendix I"[34] and Attachment 6 to SECY-03-0069, "Results of the License Termination Rule Analysis,"[35] dated May 2, 2003).

4. 10 CFR 72.44(d), "License Conditions,"[36] establishes environmental monitoring requirements for each facility holding a specific license under Part 72 authorizing receipt, handling, and storage of spent fuel, high-level radioactive waste, and/or reactor-related greater than class "C" waste.

5. Section IV.B of Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low As Is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

6. General Design Criterion 60, "Control of releases of radioactive materials to the environment," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," specifies nuclear power units shall control liquid and gaseous effluents and handle solid waste for both normal and anticipated operational occurrences.

7. General Design Criterion 64, "Monitoring radioactivity releases," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," specifies that a means shall be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released during both normal and anticipated operational occurrences.

Six basic documents contain the regulatory guidance for implementing the 10 CFR Part 20 and 10 CFR Part 50 regulatory requirements and plant technical specifications related to monitoring and reporting of radioactive material in effluents and environmental media, solid radioactive waste disposal, and the public dose resulting from licensed operation of a nuclear power plant:

1. RG 1.21 (Rev. 2, 06/2009), "Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste,"[37] addresses the measuring, evaluating, and reporting of effluent releases, solid radioactive waste, and public dose from nuclear power plants. The guide describes the important concepts in planning and implementing an effluent and solid radioactive waste program. Concepts covered include meteorology, release points, monitoring

methods, identification of principal radionuclides, unrestricted area boundaries, continuous and batch release methods, representative sampling, composite sampling, radioactivity measurements, decay corrections, quality assurance, solid radioactive waste shipments, and public dose assessments.

2. RG 4.1 (Rev. 2, 06/2009), "Radiological Environmental Monitoring for Nuclear Power Plants,"[38] addresses the environmental monitoring program. The guide discusses principles and concepts important to environmental monitoring at nuclear power plants. The regulatory guide addresses the need for preoperational and background characterization of radioactivity. It also addresses environmental monitoring (both onsite and offsite), including the exposure pathways. The guide defines the exposure pathways, the program scope of sampling media and sampling frequency, and the methods of comparing environmental measurements to effluent releases in the Annual Radiological Environmental Operating Report.

3. RG 4.15 (Rev. 2, 07/2007), "Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination)—Effluent Streams and the Environment,"[39] provides the basic principles of QA in all types of radiological monitoring programs for effluent streams and the environment. The guide addresses all types of licenses including nuclear power plants. The guide provides the principles for structuring organizational lines of communication and responsibility using qualified personnel, implementing standard operating procedures, defining data quality objectives (DQOs), performing quality control checking for sampling and analysis, auditing the process, and taking corrective actions.

4. NUREG-1301, "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Pressurized Water Reactors."[40]

5. NUREG-1302, "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Boiling Water Reactors." [41]

   NUREG-1301[40] and NUREG-1302[41] provide the detailed implementation guidance by describing effluent and environmental monitoring programs. The NUREGs specify effluent monitoring and environmental sampling requirements, surveillance requirements for effluent monitors, types of monitors and samplers, sampling and analysis frequencies, types of analysis and radionuclides analyzed, lower limits of detection (LLDs), specific environmental media to be sampled, and reporting and program evaluation and revision.

6. RG 1.109 (Rev. 1, 10/1977), "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Demonstrating Compliance with 10 CFR Part 50, Appendix I,"[34] provides the detailed implementation guidance for demonstrating that radioactive effluents conform to the ALARA design objectives of 10 CFR 50, Appendix I. The regulatory guide describes calculational models and parameters for estimating dose from effluent releases including the dispersion of the effluent in the atmosphere and different water bodies.

These six documents, when used in an integrated manner, provide the basic guidance and implementation details for developing and maintaining effluent and environmental monitoring programs at nuclear power plants. The four regulatory guides specify the guidance for radiological monitoring and the assessment of dose, and the two NUREGs provide the specific implementation details for effluent and environmental monitoring programs.

Section 11.5 of the SRP,[3] "Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems," outlines the acceptance criteria acceptable to meet the relevant requirements of NRC's regulations described above.

2.3.2 Accident Monitoring Instrumentation

Regulations and design criteria for accident monitoring instrumentation in power plants are outlined by Criterion 13, Criterion 19, Criterion 64 of Appendix A to 10 CFR Part 50, and Subsection (2)(xix) of 10 CFR 50.34(f).

Criterion 13, "Instrumentation and Control," requires operating reactor licensees to provide instrumentation to monitor variables and systems over their anticipated ranges for accident conditions as appropriate to ensure adequate safety.

Criterion 19, "Control Room," requires operating reactor licensees to provide a control room from which actions can be taken to maintain the nuclear power unit in a safe condition under accident conditions including loss-of-coolant accidents (LOCAs). In addition, operating reactor licensees must provide equipment (including the necessary instrumentation) at appropriate locations outside the control room with a design capability for prompt hot shutdown of the reactor.

In addition, Subsection (2)(xix) of 10 CFR 50.34(f), "Additional TMI-Related Requirements," requires operating reactor licensees to provide adequate instrumentation for use in monitoring plant conditions following an accident that includes core damage.

RG 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants"[42] (Rev. 4, June 2006), describes a method that the NRC staff considers acceptable for use in complying with the agency's regulations with respect to satisfying criteria for accident monitoring instrumentation in nuclear power plants. RG 1.97[42] (Rev.4) endorses (with certain clarifying regulatory positions specified in Section C of the guide) the "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations" that the Institute of Electrical and Electronics Engineers (IEEE) promulgated as IEEE Std. 497-2002.[43] IEEE Std. 497-2002[43] specifies some requirements for instruments that are used for monitoring the magnitude of releases of radioactive materials through identified pathways.

2.4 Assessment and Conclusion

2.4.1 Item III.D.2.1(1): Evaluate the Feasibility and Perform a Value-Impact Analysis of Modifying Effluent-Monitoring Design Criteria

The overall objective of this issue, which "is to provide assurance that all possible accident effluent-release pathways are monitored and that monitors will perform properly under accident conditions," is covered by GDC 64, "Monitoring radioactivity releases." GDC 64 states that "Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, *effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents*." Moreover, Subsection (2)(xvii)(E) 10 CFR 50.34(f) establishes the requirement for monitoring noble gas effluents and continuous sampling of radioactive iodines and particulates in gaseous effluents. According to this part of the regulation, "each applicant for a light-water-reactor construction permit or manufacturing license whose application was pending as of February 16, 1982" in addition to "each applicant for a design certification, design approval, combined license, or manufacturing license under part 52" of 10 CFR needs to "provide instrumentation to measure, record and readout in the control room: … (E) noble gas effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples." Finally, Subsections (2)(xxvii) and (2)(xxviii) of 10 CFR 50.34(f) establish requirements for monitoring of inplant radiation and airborne radioactivity for a broad range of routine and accident conditions and for evaluating potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions.

In addition to regulations stated above, Section 11.5 of the SRP,[3] "Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems," states that "Provisions should be made for the installation of instrumentation and monitoring equipment and/or sampling and analyses of all normal and potential effluent pathways for release of radioactive materials to the environment, including nonradioactive systems that could become radioactive through interfaces with radioactive systems." Table 1 of Section 11.5 of the SRP[3] specifies the gaseous streams or effluent release points that should be monitored and sampled. In addition, for monitoring the effluents during a postulated event, Section 11.5 of the SRP[3] states that "Provisions should be made for monitoring instrumentation, sampling, and sample analyses for all identified gaseous effluent release paths in the event of a postulated accident."

As explained earlier, implementation of the proposed solutions has no impact on the core-melt accident frequency. Moreover, "while protective actions can be recommended based on effluent releases in progress, the probability for a core-melt scenario was such that actions would be recommended based on anticipated releases prior to the actual release themselves. Under this assumption, monitoring effluent releases would have little or no impact on public risk and would be mainly for confirmation and quantification."

Specific requirements related to some of the factors in the proposed design criteria mentioned in NUREG-0660 have not been established; however, based on the review of NRC's regulations presented above, staff concludes that the overall objectives of Item III.D.2.1 (1) are met by the existing regulations. Moreover, the low safety significance of the issue does not warrant further actions to evaluate and implement the proposed solutions. Therefore, the staff recommends changing the status of Item III.D.2.1 (1) and dropping this issue from further pursuit.

2.4.2 Item III.D.2.1(2): Study the Feasibility of Requiring the Development of Effective Means for Monitoring and Sampling Noble Gases and Radioiodine Released To the Atmosphere

In addition to Criterion 64, "Monitoring radioactivity releases," of the GDC, Subsection (2)(xvii)(E) of 10 CFR 50.34(f) establishes the requirement for monitoring noble gas effluents and continuous sampling of radioactive iodines and particulates in gaseous effluents. According to this part of the regulation, "each applicant for a light-water-reactor construction permit or manufacturing license whose application was pending as of February 16, 1982" in addition to "each applicant for a design certification, design approval, combined license, or manufacturing license under part 52" of 10 CFR needs to "Provide instrumentation to measure, record and readout in the control room: … (E) noble gas effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples."

Based on the review of NRC's regulations related to this issue presented above and the low safety significance of this issue, the staff concludes that Item III.D.2.1 (2) is adequately addressed by the existing regulations. Therefore, the staff recommends changing the status of Item III.D.2.1 (2) and dropping this issue from further pursuit.

2.4.3 Item III.D.2.1(3): Revise Regulatory Guides

NUREG-0660[29] called for this issue to "revise 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, Standard Review Plan Section 11.5, Process and Effluent Radiological Monitoring and Sampling Systems, and further revise Regulatory Guide 1.97, as necessary." All of these documents have been updated since the issuance of NUREG-0660.[29] Some specific factors of the design criteria mentioned in NUREG-0660[29] have not been included in these updates. However, the overall objective of the issue has been thoroughly addressed in these updates. As of April 2010, the latest revision of each document is available as follows: RG 1.21,[37] Rev. 2 (June 2009); SRP[3] Section 11.5 (March 2007); and RG 1.97,[42] Rev. 4 (June 2006).

Because of the revisions made on RG 1.21,[37] SRP[3] Section 11.5 and RG 1.97,[42] staff recommends changing the status of Item III.D.2.1 (3) and dropping this issue from further pursuit.

3. GENERIC ISSUE 81: IMPACT OF LOCKED DOORS AND BARRIERS ON PLANT AND PERSONNEL SAFETY

3.1 Overview

Generic issue 81, “Impact of Locked Doors and Barriers on Plant and Personnel Safety,” was proposed to address the risk of possible locked doors failure that may be required for fire protection, radiation protection, flood protection, and administrative controls during abnormal or accident situations when emergency conditions may require prompt and unlimited access. This issue was initially placed in the DROP category in 1984 and was given a LOW-priority ranking later in 1992. Staff conducted a review of this issue in 2010 to determine whether any new information would necessitate reassessment of original prioritization evaluation. Staff determined that the operating experience has not indicated a change in the significance of this issue. In addition, staff verified that the regulations related to this issue establish requirements that provide prompt access to affected areas and equipment during emergencies. These regulations include Subsections (e)(9)(i), (g)(5)(i), (g)(5)(ii), (e)(8)(iii), (e)(9)(ii) of 10 CFR 73.55, Appendix R to 10 CFR Part 50, “Fire Protection Program for Nuclear Power Facilities,” and RG 5.65, “Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls.” Because the existing regulations and guidance adequately address this issue and the operating experience has not indicated a change in the significance of this issue, staff recommends changing the status of generic issue 81 and dropping this issue from further pursuit.

Section 3.2 describes a historical background of the identification and prioritization of generic issue 81. Section 3.3 presents an overview of the NRC regulatory framework and publication related to protection of vital equipment at nuclear power reactors. Finally, in Section 3.4 a discussion is provided to demonstrate the application of the NRC regulatory framework to this issue and to support its disposition.

3.2 Historical Background

3.2.1 Description

The possible failure of locked doors and barriers that may be required for fire protection, radiation protection, flood protection, and administrative controls is of special concern during abnormal or accident situations when emergency conditions may require prompt and unlimited access of the plant operators to safety equipment to assure proper plant shutdown.

In October 1982, the Executive Director for Operations appointed the Committee to Review Safety Requirements at Power Reactors (CRSRPR) to review NRC security requirements at nuclear power plants with a view toward evaluating the impact of these requirements on operational safety. Overall, the CRSRPR did not identify any clear operational safety problems associated with implementation of NRC's security requirements. However, the Committee found that there was the potential for security measures at a site to adversely affect safety and issued its recommendations in a report[44] on February 28, 1983. In view of one of the findings in this report, a memorandum[45] was issued on May 31, 1983, identifying this issue and suggesting that a multidisciplinary group be convened to perform an integrated assessment of the potential safety problem associated with locked doors and barriers. Based on the responses to the memorandum, a consensus supported the creation of the multidisciplinary group to gather the necessary information and to prepare a scope of the issue for appropriate consideration.[46] This approach was approved,[47] and action on this matter was formally initiated. [48]

The multidisciplinary group held its first meeting on February 28, 1984, and issued a report on June 8, 1984.[49] Because a proposed rule (SECY-83-311)[50] specifically designed to address the security barrier issue had been prepared independently and IE Information Notice No. 83-36[51] also had been issued, the work of the group was limited to nonsecurity barriers.

The proposed rule[52] was eventually adopted and stated that "NRC is amending its regulations to provide a more safety conscious safeguards system while maintaining current levels of protection." Regulatory changes included: (1) permitting suspension of security based upon 10 CFR 50.54(x) and (y); (2) requiring the access authorization system to be designed to accommodate the potential need for rapid ingress and egress of individuals during emergency conditions or situations that could lead to emergency conditions; and (3) ensuring prompt access to vital equipment by periodically reviewing physical security plans for potential impact on plant and personnel safety. The rule was implemented with RG 5.65, "Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls,"[53] and Generic Letter 87-08,[54] which addressed the issuance of vital area keys to operations personnel. At the time of evaluation of this issue in 1995, the Reactor Safeguards Branch in the Office of Nuclear Reactor Regulation (NRR) indicated that almost all licensees were in compliance with RG 5.65[53] and Generic Letter 87-08[54] and had implemented mechanical key overrides for electronically controlled access doors. The rulemaking resulted in security plan amendments that increased the focus on plant and personnel safety.

Subsequent to the above work, a main feedwater pipe rupture event at the Surry plant (see Generic Issue 139, "Thinning of Carbon Steel Piping in LWRs"[2]) caused the failure of a security card-reader that was located about 50 feet from the break point. This failure was caused by intrusion of water and steam that saturated the card-reader. As a result, key cards could not be used to open plant doors. The control room doors were opened to provide access to the control room, and security personnel were assigned to the control room to provide access security. One operator was temporarily trapped in a stairway because of the card-reader failure. Electric override switches were later installed to remedy this problem. Because of the failure of the security card-reader during the Surry event, the staff determined that Issue 81 should be expanded to include potential electric door lock failures and reevaluated to determine whether the previous priority ranking (DROP) should be changed.[55]

3.2.2 Possible Solution

Staff proposed in NUREG-0933[2] that "[a]n evaluation of each plant's locked doors and barriers might be required and appropriate procedural and hardware changes may have to be made to establish that operator access is unimpeded during emergency, abnormal, or accident conditions, and that prompt operator action, as required, is possible."

3.2.3 Priority Determination

In the event of an accident, failure of the electronic card-reader access control system (ACS) could result in an impediment to operator actions outside of the control room that are required for recovery. Some examples of possible operator actions are: (1) locally overriding a failed component, (2) replacing or repairing a failed component, or (3) realigning valves to bypass a failed pump or clogged pipe. If the card-reader ACS fails, the operator will be impeded in his access through the door.

Even if the ACS fails, there is a large probability that the plant will have a mechanical key override or that the locks will fail open. The study conducted by the CRSRPR estimated that a majority of plants did not have problems with ACS computer failure either because the doors fail open, mechanical key overrides are available, or the number of controlled areas is small.[56] An NRR review of plant safeguards revealed that only one plant that did not have a mechanical key override on ACS-controlled doors had locks that failed open. Based on these data, a probability of 0.01 was assumed to account for the occurrence of no key override due to lost or misplaced keys, mechanical failure of the override, or failure of an electronic ACS to fail open if so designed.[2]

The estimated frequency of card-reader ACS failure and its impact on plant safety indicated that improvements in this area were not a cost-effective way to increase overall plant safety.[2] Moreover, the multidisciplinary task group concluded that the locks and barriers associated with these areas could easily be defeated or bypassed in an emergency situation, if necessary, provided enough time was available to take the necessary steps. In addition, implementation of the regulatory guidance associated with rulemaking[52] resulted in better coordination between plant security and operations personnel. Thus, this issue was given a LOW-priority ranking. Consideration of a 20-year license renewal period did not change the priority of the issue.[56]

3.3 NRC Regulations and Policies

NRC's principal requirements with respect to the protection of items of vital equipment at nuclear power reactors are contained in 10 CFR Part 73, "Physical Protection of Plants and Materials." These requirements are aimed at safeguarding against sabotage that could cause a radiological release. 10 CFR Part 73 “prescribes requirements for the establishment and maintenance of a physical protection system that will have capabilities for the protection of special nuclear material at fixed sites and in transit and of plants in which special nuclear material is used.” The physical security plan provides high assurance against the design basis threat outlined in 10 CFR Part 73.1(a) to ensure activities involving special nuclear material are not inimical to common defense and security and do not constitute an unreasonable risk to the public health and safety.

10 CFR 73.55 establishes the detailed requirements for development and implementation of a physical security plan. The physical security plan defines the administrative, physical, and operational measures that provide protection of the facility and any associated special nuclear material from both internal and external threats. Compliance with 10 CFR 73.55 provides high assurance that a facility is protected against theft or diversion of nuclear material or radiological sabotage. 10 CFR 73.55(e), “Physical Barriers;” 10 CFR 73.55(g), “Access Controls;” and 10 CFR 73.55(g), “Testing

and Maintenance," contain rules that are related to generic issue 89 that will be discussed in the next section

RG 5.65, "Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls,"[53] (September 1986) describes measures the NRC staff considers acceptable to implement regulatory requirements on access controls. The purpose of these measures, in part, is to ensure adequate access for safety purposes while providing necessary physical security.

Finally, guidance for review of combined license applications is located in SRP[3] 13.6.1, "Physical Security - Combined License Review Responsibilities." The same guidance applies to licensing actions under Part 50. Guidance for the review of design certification applications is located in SRP[3] 13.6.2, "Physical Security - Design Certification." Lastly, guidance for the review of early site permit applications is located in SRP[3] 13.6.3, "Physical Security - Early Site Permit."

3.4 Assessment and Conclusion

According to Subsection (9)(i) of 10 CFR 73.55(e), "Vital equipment must be located only within vital areas, which must be located within a protected area so that access to vital equipment requires passage through at least two physical barriers, except as otherwise approved by the Commission and identified in the security plans." During emergencies or abnormal conditions, it may be necessary for certain licensee personnel to gain quick access to vital equipment to mitigate or terminate some adverse plant condition. Paragraph 73.55(g)(5)(i) requires that "The licensee shall design the access control system to accommodate the potential need for rapid ingress or egress of authorized individuals during emergency conditions or situations that could lead to emergency conditions." Moreover, paragraph 73.55(g)(5)(ii) states that "To satisfy the design criteria of paragraph (g)(5)(i) of this section during emergency conditions, the licensee shall implement security procedures to ensure that authorized emergency personnel are provided prompt access to affected areas and equipment."

In addition, requirements have been established to ensure that personnel can quickly evacuate vital areas if the emergency condition results in high radiation or other dangerous conditions within the vital area. Paragraphs 73.55(e)(8)(iii) and 73.55(e)(9)(ii) state, in part, this requirement for protected area and vital area, respectively. 73.55(e)(8)(iii) states that "All emergency exits in the protected area must be alarmed and secured by locking devices that allow prompt egress during an emergency and satisfy the requirements of this section for access control into the protected area." In addition, for 73.55(e)(9)(ii) states that "The licensee shall protect all vital area access portals and vital area emergency exits with intrusion detection equipment and locking devices that allow rapid egress during an emergency and satisfy the vital area entry control requirements of this section."

Finally, Appendix R to 10 CFR Part 50, "Fire Protection Program for Nuclear Power Facilities," states that administrative controls shall establish procedures to define the strategies for fighting fires in all safety-related areas and areas presenting a hazard to safety-related equipment. Under these strategies, in part, "All access and egress routes that involve locked doors should be specifically identified in the procedure with the appropriate precautions and methods for access specified."

In addition to regulations stated above, for emergencies or abnormal conditions, RG 5.65[53] states that “Licensees can provide for rapid ingress/egress during such conditions by providing backup keys to vital areas and methods of opening locked doors in the case of computer or power failure.” Moreover, RG 5.65[53] describes acceptable procedures for providing for safe ingress/egress during a power or computer outage.

Based on the review of NRC’s regulations related to this issue presented above, the staff concludes that the existing regulations adequately establish requirements that provide prompt access to affected areas and equipment during emergencies. Therefore, the staff recommends changing the status of generic issue 81 and dropping this issue from further pursuit.

4. GENERIC ISSUE 127: MAINTENANCE AND TESTING OF MANUAL VALVES IN SAFETY-RELATED SYSTEMS

4.1 Overview

Generic issue 127, "Maintenance and Testing of Manual Valves in Safety-Related Systems," was proposed in 1986 to assess the adequacy of the maintenance program for manual valves. This issue was given a LOW priority ranking in 1987, as reported in Supplement 7 to NUREG-0933. Staff conducted a review of this issue in 2010 to determine whether any new information would necessitate reassessment of original prioritization evaluation. Based on the review of NRC's regulations, staff determined that this issue is addressed by Subsections (b)(2)(i) and (b)(2)(ii) of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," and Chapter 17.6, "Maintenance Rule," of the SRP (revised in 2007). In addition, the operating experience has not indicated a change in the safety significance of this issue. Because the existing regulations and guidance adequately address this issue and the operating experience has not indicated a change in the significance of this issue, staff recommends changing the status of generic issue 127 and dropping this issue from further pursuit.

Section 4.2 describes a historical background of the identification and prioritization of this issue. Section 4.3 presents an overview of the NRC regulatory framework and publication related to the maintenance rule. Finally, in Section 4.4 a discussion is provided to demonstrate the application of the NRC regulatory framework to this issue and to support its disposition.

4.2 Background

4.2.1 Description

This issue was identified in the NRC Incident Investigation Team (IIT) report on the loss of integrated control system (ICS) power event at Rancho Seco on December 26, 1985 (NUREG-1195).[57] Following the event, it was requested that the adequacy of the maintenance program for manual valves be prioritized as a generic issue.[59] In addition, the staff drafted an Information Notice[58] that was later issued as IE Information No. 86-61, "Failure of Auxiliary Feedwater Manual Isolation Valve," [60] on July 28, 1986.

In the Rancho Seco event, power was lost to the ICS and the plant responded as designed—the Auxiliary Feedwater (AFW) ICS flow-control valves and other valves went to the 50-percent open position. However, AFW flow was excessive and an unsuccessful attempt was made to manually close the flow control valve to the "A" Once-Through Steam Generator (OTSG). The operator then attempted to close the manual isolation valve and failed to do so as the valve was "frozen" in the open position and could not be moved even when a valve wrench was used. Consequently, this inability to reduce AFW flow resulted in an overcooling event. The IIT found that the failure of the AFW manual isolation valve was the result of a lack of preventive maintenance (including lubrication) on this valve during the entire operational life of the plant (about 10 to 12 years).

The manual isolation valve is a locked-open valve located in the AFW discharge header to the "A" OTSG. During the IIT investigation, a Sacramento Municipal Utility District

(SMUD) representative stated that the entire AFW system, which would include this manual isolation valve, is safety related. However, from other discussions with SMUD personnel, it appeared that this valve was only intended to be used to isolate the AFW (ICS) flow control valve for maintenance. The valve is categorized as an ASME Category E valve (i.e., it is normally locked open to fulfill its function). ASME Section XI (1974 edition) requires no regular testing of Category E valves. The position of the valves is merely recorded to verify that each valve is locked or sealed in its correct position. The current edition of ASME Section XI no longer includes a Category E for valves.

Following the incident, it was found that licensees did not have a regular maintenance program that applied to every manual valve. NRC did not have a requirement for maintenance and testing of convenience valves such as the locked-open manual valve involved in the Rancho Seco incident. ASME Section XI specifies inservice inspection, testing, repair, and replacement of valves that are components in systems classified as ASME Classes 1, 2, and 3 and are required to perform a specific function in shutting down a reactor to a cold shutdown condition or in mitigating the consequences of an accident. Manual valves in safety-related systems that are classified Quality Group A, B, or C in conformance with RG 1.26[15] are constructed to ASME Section III, Classes 1, 2, or 3 or to earlier codes and standards, as appropriate. These manual valves may be fill, vent, drain, or convenience valves and are constructed to the same code class as the system, or part of a system, of which they are a part. Such valves were not included in the inservice testing (IST) program for valves that were in conformance with ASME Section XI as noted above because they were not required to change position to perform a safety function. In the event a manual valve was required to change position to perform a safety function; it was included in the ASME Section XI IST program and classified as a safety-related valve.

At the time, the NRC requirements for valve testing were contained in 10 CFR 50.55 (a)(g) that incorporates ASME Section XI. Therefore, regulatory requirements for valve testing extended only to valves that were within the IST program. The Quality Group (Safety Class) and construction code of each valve was verified, and the valve category was also verified for conformance with Section XI, IWV-2000. In addition, the NRR staff performed a completeness review to assure that all appropriate valves within the scope of ASME Section XI were included in the IST program. The licensees are responsible for performing the testing, repair, and maintenance on the valves that are within their IST and maintenance programs.

4.2.2 Possible Solutions

The staff proposed in NUREG-0933[2] to (1) develop or revise regulatory requirements relating to the inspection, testing, and maintenance of those fill, vent, drain, and convenience valves in safety-related systems that do not change position for the systems to perform their safety function or (2) identify this as an item for which NRC has concern, notify the licensees by an information notice, and let them determine the maintenance practices they wish to implement.

4.2.3 Priority Determination

In December 1987, staff made a priority determination in NUREG-0933[2] assigning a low priority to this issue "…due to the minimal estimated reduction in public risk resulting from the resolution of this issue." In arriving at this determination, the staff concluded in NUREG-0933[2] that the risk from this issue was very low and "Due to the low costs associated with maintaining the manual isolation valves, it would appear to be cost effective for plant operators to maintain them as a good practice and not require a regulatory requirement. The power replacement cost for one day of plant outage which may result from the inability to isolate would pay the plant life costs for isolation valve maintenance. In view of this cost saving potential, the release of the Information Notice may resolve this issue."

4.3 Regulatory Framework

4.3.1 Regulatory Background

Operating experience from the 1980s, including the Rancho Seco event, led the staff to implement a combination of the possible solutions noted above. Specifically, on July 10, 1991, NRC published 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," also known as the maintenance rule. The associated RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,"[61] provides insight into the bases for establishing the new rule. In the introduction, RG 1.160 states:

> "…The NRC's determination that a maintenance rule was needed arose from the conclusion that proper maintenance is essential to plant safety. As discussed in the regulatory analysis for this rule, there is a clear link between effective maintenance and safety as it relates to such factors as the number of transients and challenges to safety systems and the associated need for operability, availability, and reliability of safety equipment. In addition, good maintenance is also important in providing assurance that failures of other than safety-related structures, systems, and components (SSCs) that could initiate or adversely affect a transient or accident are minimized."

The latter reference to failures of other than safety-related SSCs speaks directly to this generic issue and the Ranch Seco event because the malfunction of a manual isolation valve adversely affected the transient, resulting in an overcooling event.

4.3.2 Publications

Below is a list of RGs and Regulatory Issue Summaries (RIS) NRC publications related to the maintenance rule and nonsafety-related SSCs. This list represents the depth and breadth of the applicability of the maintenance rule to nonsafety-related SSCs. In addition to this list, the maintenance rule also has been incorporated in Section 17.6 of the SRP, "Maintenance Rule."

1. RG 1.54 (Rev.1, 07/2000), "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants,"[62] is applicable to this generic issue because licensees that commit to this RG should meet the quality assurance provisions and guidance

contained in the standards in this RG and must also meet the commitments and provisions contained in their Quality Assurance Program including service level III protective coatings. These coatings are used in areas outside the reactor containment where failure could adversely affect the safety function of a safety-related SSC.

2. RG 1.160 (Rev.2, 03/1997), "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,"[61] is applicable to this generic issue because it covers the general provisions and guidance for complying with the Maintenance Rule including the regulatory position on nonsafety-related SSCs that are relied upon to mitigate accidents or transients.

3. RG 1.182 (05/2000), "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants,"[63] is a companion guide to RG 1.160 and provides guidance on methods acceptable to the NRC staff for assessing and managing the increase in risk that may result from maintenance activities and for implementing the optional reduction in scope of SSCs considered in the assessments. These maintenance activities also would include maintenance on nonsafety-related equipment such that failures will not occur that prevent the fulfillment of safety-related functions.

4. RIS 01-003 (01/23/2001), "Changes, Tests, and Experiments,"[64] among other purposes, clarifies the regulatory position on the requirements for performing 10 CFR 50.59 evaluations or 10 CFR 50.65(a)(4) maintenance risk assessments. The maintenance risk assessments also would include maintenance activities on nonsafety-related SSCs relied upon to mitigate accidents or transients.

5. RIS 06-007 (06/12/2006), "Changes to the Safety System Unavailability Performance Indicators,"[65] informs licensees that in April 2006 the agency replaced the Safety System Unavailability (SSU) Performance Indicators (PI) with the Mitigating Systems Performance Index (MSPI). Among other issues, the MSPI addressed the inconsistency of reporting unavailability data between the SSU PI and the maintenance rule. As such, the MSPI accounts for unavailability and unreliability contributions, some of which will be derived from activities associated with maintaining nonsafety-related SSCs relied upon to mitigate accidents or transients.

4.4 Assessment and Conclusion

The evaluation of this issue resulted in a LOW-priority rating as reported in NUREG-0933[2] published in December 1987. As published in 1991, sections (a) and (b) of the maintenance rule 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," state that

> "(a)(1) Each holder of an operating license for a nuclear power plant under this part and each holder of a combined license under part 52 of this chapter after the Commission makes the finding under § 52.103(g) of this chapter, shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that these structures, systems, and components, as defined in paragraph (b) of this section, are capable of fulfilling their intended functions. These goals shall be established

commensurate with safety and, where practical, take into account industrywide operating experience. When the performance or condition of a structure, system, or component does not meet established goals, appropriate corrective action shall be taken. For a nuclear power plant for which the licensee has submitted the certifications specified in § 50.82(a)(1) or 52.110(a)(1) of this chapter, as applicable, this section shall only apply to the extent that the licensee shall monitor the performance or condition of all structures, systems, or components associated with the storage, control, and maintenance of spent fuel in a safe condition, in a manner sufficient to provide reasonable assurance that these structures, systems, and components are capable of fulfilling their intended functions.

(b) The scope of the monitoring program specified in paragraph (a)(1) of this section shall include safety related and nonsafety related structures, systems, and components, as follows:

(1) Safety-related structures, systems and components that are relied upon to remain functional during and following design basis events to ensure the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition, or the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the guidelines in Sec. 50.34(a)(1), Sec. 50.67(b)(2), or Sec. 100.11 of this chapter, as applicable.

(2) Nonsafety related structures, systems, or components:

(i) That are relied upon to mitigate accidents or transients or are used in plant emergency operating procedures (EOPs); or

(ii) Whose failure could prevent safety-related structures, systems, and components from fulfilling their safety-related function; or

(iii) Whose failure could cause a reactor scram or actuation of a safety-related system."

Sections (b)(2)(i) and (b)(2)(ii) address the event presented in this generic issue and, as demonstrated above with applicable operating experience, has addressed similar subsequent events. Moreover, the SRP was revised in 2007 to include Chapter 17.6, "Maintenance Rule," which outlines the criteria for evaluating licensee applications for the scope, monitoring, evaluation, and risk assessment and management of implementing 10 CFR 50.65 including Section III, 1.B, which outlines the criteria for including nonsafety-related SSCs in accordance with 50.65(b)(2). Criterion iii of this section applies directly to this generic issue, stating that the description of the maintenance rule scoping process should address

"SSCs whose failure could prevent safety-related SSCs from fulfilling their safety-related functions in accordance with 50.65(b)(2)(ii). The applicant should describe how the process considers system interdependencies,

> including failure modes and effects of nonsafety-related SSCs (*e.g.*, support systems) that could directly affect safety-related functions."

Based on the review of NRC's regulations and guidance related to this issue, the staff concludes that existing regulations and guidance adequately address this issue. Therefore, the staff recommends changing the status of generic issue 127 and dropping this issue from further pursuit.

5. GENERIC ISSUE 167: HYDROGEN STORAGE FACILITY SEPARATION

5.1 Overview

Generic issue 167, “Hydrogen Storage Facility,” was proposed in 1993 to address the potential risk from large H_2 storage facilities outside the reactor, auxiliary, and turbine buildings. This issue was given a LOW-priority ranking in June 1995. Staff conducted a review of this issue in 2010 to determine whether any new information would necessitate reassessment of original prioritization evaluation. Based on the review of NRC’s regulations, staff determined that this issue is addressed by Inspection Procedures 71111.05AQ and 71111.05T. In addition, the operating experience has not indicated a change in the significance of this issue. Because the existing regulations and guidance adequately address this issue and the operating experience has not indicated a change in the significance of this issue, staff recommends changing the status of generic issue 167 and dropping this issue from further pursuit.

Section 5.2 describes a historical background of the identification and prioritization of this issue. Section 5.3 presents an overview of the NRC regulatory framework and publication related to this issue. Finally, in Section 5.4 a discussion is provided to demonstrate the application of the NRC regulatory framework to this issue and to support its disposition.

5.2 Background

5.2.1 Description

Issue 106, “Piping and the Use of Highly Combustible Gases in Vital Areas (Rev. 2),” was resolved with the issuance of Generic Letter 93-06[66] that included evaluation of the risk from (1) the storage and distribution of H_2 for the volume control tank in PWRs and the main electric generator in boiling-water reactors (BWRs) and PWRs; (2) other sources of H_2 such as battery rooms, the waste gas system in PWRs, and the offgas system in BWRs; and (3) small, portable bottles of combustible gases used in maintenance, testing, and calibration. However, the potential risk from large H_2 storage facilities outside the reactor, auxiliary, and turbine buildings was not addressed. Studies performed during and subsequent to the resolution of Issue 106 raised concerns about the magnitude of the excluded risk.[67] Thus, in December 1993, this issue was identified[68] to address this excluded risk.

NRC Information Notice No. 89-44, "Hydrogen Storage on the Roof of the Control Room," [69] was issued in May 1989, and each NRC Regional Office was expected to determine whether the plants in its region had similar safety-related concerns. The information compiled by these offices was reviewed and issued in the preliminary report SCIE-EGG-103-89.[70] The storage of gaseous or liquid H_2 at 119 power plants was then investigated, and possible accident scenarios resulting from a fireball, explosion, or presence of unburned H_2 gas in ventilation air intakes were examined. Explosion was identified as the scenario posing the greatest risk potential. The analysis in SCIE-EGG-103-89[70] focused on explosion with all quantification performed relative to this accident only.

The safety concern was whether or not adequate physical separation exists between H_2 storage facilities and buildings or structures housing systems important to safety at nuclear power plants. As reported in SCIE-EGG-103-89,[70] "[a]t the Trojan Nuclear Plant, April 17, 1989, [NRC] inspectors identified a potential safety problem concerning the storage of 32,000 standard cubic feet (scf) of hydrogen gas on the control room roof. The 32,000 scf was made up of four 8,000 scf tanks. This discovery raised concerns about possible similar hazards in the storage of hydrogen at other nuclear facilities."

5.2.2 Possible Solutions

Staff stated in NUREG-0933[2] that “possible solutions included relocation (or placement in pits) of storage facilities, buildings, and equipment, and the construction of blast shields, or a combination of these.” The resolution for this issue was assumed to be the construction of concrete walls enclosing the H_2 storage facility. This structure would serve as a blast shield in the event of an explosion, essentially eliminating the risk.

5.2.3 Priority Determination

Based on the impact/value ratio and the potential reduction in Core Damage Frequency and public risk described In NUREG-0933,[2] staff assigned a LOW-priority ranking to this issue in June 1995.

5.3 Regulatory Framework

5.3.1 Regulatory Background

In lieu of the staff-proposed solutions as cited above, NRC has addressed this issue through a combination of regulatory vehicles including the issuance and implementation of temporary instructions, rulemaking, and the continued, periodic inspection of fire protection programs and plant modifications (changes, tests, and experiments) at licensee facilities.

The foundation for these regulatory vehicles is Criterion 3 of 10 CFR 50, Appendix A which states

> “…Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.”

To satisfy this criterion, in November 1980, NRC added 10 CFR 50.48 and 10 CFR 50, Appendix R. Together, these regulations seek to establish safety margins through minimizing the potential for fires and explosions; rapidly detecting, controlling, and extinguishing fires that do occur; and ensuring post-fire survival of the systems needed to shut down the reactor. The inappropriate separation of hydrogen storage facilities can challenge a licensee's ability to meet all of these objectives.

In 2000, NRC implemented the Reactor Oversight Process (ROP) that includes quarterly, annual, and triennial fire protection inspections via Inspection Procedures 71111.05AQ[71] and 71111.05T[72]. The ROP also includes the Significance Determination Process specifically for fire protection found in Inspection Manual Chapter 0609F, "Fire Protection Significance Determination Process."[73] In April 2002, NRC issued Temporary Instruction (TI) 2515/146, Revision 1, "Hydrogen Storage Locations." NRC issued this TI to verify licensee compliance with applicable codes and commitments regarding the location of hydrogen storage at operating nuclear power plants. NRC's supplemental oversight of this issue was in response to a hydrogen fire that occurred in the hydrogen storage facility at James A. Fitzpatrick nuclear power plant on January 14, 1999. Following this event, in May 1999, NRC conducted a survey of all operating plants to update information about hydrogen storage facilities. As a result of the less–than-complete survey responses from 30 licensees, NRC issued and implemented TI 2515/146,[74] Revision 1.

In July 2004, NRC approved the risk-informed and performance-based alternative regulation, 10 CFR 50.48(c), allowing licensees to focus their fire protection activities on the areas of greatest risk.

5.3.2 Publications

Since 1995, NRC has issued very few generic communications related specifically to this generic issue. Of note is NRC Information Notice (IN) 2001-12, "Hydrogen Fire at Nuclear Power Station,"[75] dated July 13, 2001. This IN alerts licensees to the potential hazards associated with hydrogen storage facilities including their separation, maintenance, and monitoring. In addition, Section 9.5.1, "Fire Protection Program" of the SRP[3] provides the staff with the review criteria for evaluating licensee applications with respect to comprehensive identification and analysis of fire and explosion hazards, among other elements. Included in this section is the reference to RG 1.189, "Fire Protection for Nuclear Power Plants,"[76] that includes guidance on the use of National Fire Protection Association codes for the separation of gaseous and liquefied hydrogen systems. Moreover, over 100 generic communications and regulatory guides exist that cover various aspects of the fire protection program and its requirements, many of which have a general reference to performing appropriate analyses for explosive hazards and separating hydrogen systems.

5.4 Assessment and Conclusion

This evaluation of this issue resulted in a LOW-priority rating as reported in NUREG-0933[2] published in June 1995. Between the publication of this generic issue in 1995 and the year 2000, very little followup was performed regarding this specific issue. During that time, most licensees committed to National Fire Protection Association code NFPA 50A, "Gaseous Hydrogen Systems at Consumer Sites," and NFPA 50B, "Liquefied Hydrogen System at Consumer Sites," as part of their licensing basis.[77] These codes provided separation distances for gaseous and liquefied hydrogen providing a basis for inspection and potential enforcement, further supporting the LOW-priority rating of this generic issue.

In 2000, with the implementation of the ROP, Inspection Procedures 71111.05AQ[71] and 71111.05T[72] were issued. The objectives of these inspection procedures are to

- Evaluate the adequacy and implementation of the licensees' fire protection programs.

- Review the procedures to incorporate and implement changes to the respective fire protection programs.

- Determine the adequacy of the licensees's systems for taking corrective action when warranted by QA programs, generic deficiencies, or events.

With respect to this generic issue, these inspection procedures verify that a licensee's fire protection program included the control of combustible material, including the appropriate storage of bulk flammable gases and liquids like hydrogen. To that end, inspection procedures also verify that the licensee's fire protection program consists of a fire hazard analysis, which includes analyses for postulated hydrogen explosions. The fire protection program also includes the facility's technical specifications, which includes the appropriate limiting condition for operations to prevent the postulated fire conditions.

In December 2002, NRC reported the results of the inspections under TI 2515/146[74]. The report highlighted findings related to the adequate separation of hydrogen storage facilities from risk significant tanks or SSCs and from ventilation intakes. The licensees of these plants committed to taking appropriate corrective actions.

With respect to recent enforcement, in December 2008, inspectors identified a Severity Level IV non-cited violation of 10 CFR 50.59, "Changes, Tests, and Experiments," for the licensee's failure to perform a safety evaluation associated with installation of a bulk hydrogen storage facility located directly above buried Circulating Water System return lines.[78]

Based on the review of NRC's regulations and guidance related to this issue, the staff concludes that existing regulations and guidance adequately address this issue. Therefore, the staff recommends changing the status of generic issue 167 and dropping this issue from further pursuit.

6. REFERENCES

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2. NUREG-0933, "Resolution of Generic Safety Issues (formerly entitled, "A Prioritization of Generic Safety Issues") (NUREG-0933, Main Report with Supplements 1–32)," August 2008.

3. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission.

4 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

5. American National Standards Institute, "Quality Assurance Program Requirements for Nuclear Power Plants," ANSI N45.2. Washington, DC.

6. American Society of Mechanical Engineers, "Quality Assurance Program Requirements for Nuclear Facility Applications," ANSI/ASME Standard NQA -1, Washington, DC.

7. Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)," U.S. Nuclear Regulatory Commission, Washington, DC, Rev. 3, August 1985.

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12. Regulatory Guide 1.176, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Graded Quality Assurance," U.S. Nuclear Regulatory Commission, Washington, DC, August 1998.

13. SECY-98-300, Options for Risk-Informed Revisions to 10 CFR Part 50 - "Domestic Licensing of Production and Utilization Facilities," December 23, 1998.

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15. Regulatory Guide 1.26, “Quality Group Classification, and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, Washington, DC, Rev. 4, March 2007.

16. Regulatory Guide 1.29, “Seismic Design Classification,” U.S. Nuclear Regulatory Commission, Washington, DC, Rev. 4, March 2007.

17. Regulatory Guide 1.37, “Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, Washington, DC, Rev. 1, March 2007.

18. Nuclear Information and Records Management Association, “Authentication of Records and Media,” NIRMA TG 11-1998, Windham, NH.

19. Nuclear Information and Records Management Association, “Management of Electronic Records.” NIRMA TG 15-1998, Windham, NH.

20. NIRMA TG 16-1998, “Software Configuration Management and Quality Assurance.”

21. NIRMA TG 21-1998, “Electronic Records Protection and Restoration.”

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23. Regulatory Guide 1.8, “Qualification and Training of Personnel for Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, Washington, DC, Rev. 3, May 2000.

24. ANSI/ANS-3.1-1993, "Selection, Qualification, and Training of Personnel for Nuclear Power Plants,"

25. SECY-03-0117, “Approaches for Adopting More Widely Accepted International Quality Standards,” July 9, 2003.

26. Task III.D.2: “Public Radiation Protection Improvement (Rev. 3) (NUREG-0933, Main Report with Supplements 1–32), available at: http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0933/sec1/3-d2r3.html.

27. NUREG-0578, “TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations,” U.S. Nuclear Regulatory Commission, July 1979.

28. NUREG-0654, “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, February 1980, (Rev. 1) November 1980.

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30. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983, (Supplement 1) May 1983, (Supplement 2) December 1983, (Supplement 3) September 1985, (Supplement 4) July 1986, (Supplement 5) July 1996.

31. NUREG/CR-5382, "Screening of Generic Safety Issues for License Renewal Considerations," U.S. Nuclear Regulatory Commission, December 1991.

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