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Revision 1

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 1.196

CONTROL ROOM HABITABILITY AT LIGHT-WATER NUCLEAR POWER REACTORS

A. INTRODUCTION

This guide provides guidance and criteria that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for implementing the agency's regulations in Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10, Part 50, of the *Code of Federal Regulations* (10 CFR Part 50), "Domestic Licensing of Production and Utilization Facilities," as they relate to control room habitability (CRH). Specifically, this guide outlines a process that licensees may apply to control rooms that are modified, are newly designed, or must have their conformance to the regulations reconfirmed.

In Appendix A to 10 CFR Part 50, General Design Criteria (GDC) 1, 3, 4, 5, and 19 apply to CRH, as follows:

- GDC 1, "Quality Standards and Records," requires that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions performed.
- GDC 3, "Fire Protection," requires that SSCs important to safety be designed and located to minimize the effects of fires and explosions.
- GDC 4, "Environmental and Dynamic Effects Design Bases," requires SSCs important to safety to be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs).

- GDC 5, “Sharing of Structures, Systems, and Components,” requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, the orderly shutdown and cooldown of the remaining units.
- GDC 19, “Control Room,” requires that a control room be provided from which actions can be taken to operate the nuclear reactor safely under normal conditions and to maintain the reactor in a safe condition under accident conditions, including a LOCA. Adequate radiation protection is to be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of specified values.

Since the NRC initially issued Regulatory Guide 1.196 in May 2003, the staff determined that the information presented in Appendix B to that guide did not accurately represent a viable technical specification for CRH at light-water nuclear power reactors. In particular, it referred to failure of a particular surveillance as a plant state, rather than having the results of the surveillance factor into the operability determination. In addition, it did not provide for a definite time to restore functionality to the control room envelope, whereas all improved standard technical specifications (iSTS) contain such provisions. Moreover, Appendix B was included as a “strawman,” to be deleted when details had been more carefully worked out with industry participation, and those technical specifications placed in the iSTS with all other acceptable technical specifications.

As of the publication date of this Revision 1 of Regulatory Guide 1.196, no utility has been granted the technical specification changes represented by Appendix B to the original version of this guide. Consequently, the NRC staff elected to remove Appendix B (and all related references) from this revision. Removal of Appendix B from this revised guide does not require any stakeholder to take any action and does not reduce safety in any way. Moreover, the owners’ group Technical Specification Task Force has provided ample opportunity for public comment regarding this revision. Therefore, the staff views the removal of Appendix B as a neutral action, for which further public comments are unnecessary. For that reason, the staff chose not to issue this revision as a draft guide for public comment before publishing this Revision 1 of Regulatory Guide 1.196.

This regulatory guide contains information collections that are covered by the requirements of 10 CFR Part 50, and that the Office of Management and Budget (OMB) approved under OMB control number 3150-0011. The NRC may neither conduct nor sponsor, and a person is not required to respond to, an information collection request or requirement unless the requesting document displays a currently valid OMB control number.

B. DISCUSSION

A licensee may use this guide for assessing CRH following changes to control room habitability systems (CRHSs) or the sources that would lead to consequences to the operator. Changes that may impact the existing CRH assessments and may result in re-analysis of the licensee's CRH include the following examples:

- changes in procedures, operation, performance,[1] alignment, or function of the CRHSs
- new hazardous chemicals or radioactive sources introduced onsite or in the vicinity of the plant
- increases in hazardous chemical or radioactive source quantities, concentrations, locations, or shipments

The primary design function of CRHSs is to provide a safe environment in which the operator can keep the nuclear reactor and auxiliary systems under control during normal operations and can safely shut down those systems during abnormal situations to protect the health and safety of the public and plant workers. If the control room is not habitable or the response of the operator is impaired during an accident, there could be increased consequences to public health and safety. It is important for the operators to be confident of their safety in the control room to minimize errors of omission and commission. The regulatory positions in Section C provide methods that the NRC staff considers acceptable for ensuring that the public and the control room operator are protected.

When possible, this guide incorporates guidance contained in NEI 99-03, "Control Room Habitability Assessment Guidance" (Ref. 1). The staff has reviewed this document and concluded that portions of NEI 99-03 can serve as a valuable resource on CRH. Only the sections of NEI 99-03 that are specifically stated in the regulatory positions should be considered to be endorsed by the NRC. The staff's endorsement of these sections should not be considered an endorsement of the remainder of NEI 99-03 or of any other document referenced in NEI 99-03. Appendix A to this guide summarizes the staff's endorsement.

Definitions of key terms used within the context of this regulatory guide are given below. However, in most cases a facility's licensing basis[2] and associated documents will define the terms for a particular facility.

Control Room: The plant area, defined in the facility's licensing basis, in which actions can be taken to operate the plant safely under normal conditions and to maintain the reactor in a safe condition during accident situations. It encompasses the instrumentation and controls necessary for safe shutdown of the plant and typically includes the critical document reference file, computer room (if used as an integral part of the emergency response plan), shift supervisor's office, operator wash room and kitchen, and other critical areas to which frequent personnel access or continuous occupancy may be necessary in the event of an accident.

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[1] An example of a changed performance parameter that may require re-analysis is an increase in control room envelope (CRE) inleakage beyond that assumed in previous CRH assessments.

[2] As used in this guide, the licensing basis is the documentation that describes how the plant meets applicable regulations. Design bases are defined in 10 CFR 50.2. Regulatory Guide 1.186, "Guidance and Examples for Identifying 10 CFR 50.2 Design Bases" (Ref. 2), provides additional guidance. The design bases are a subset of the licensing bases. Thus, this guide uses "licensing bases" to refer to both.

Control Room Envelope (CRE): The plant area, defined in the facility's licensing basis, that in the event of an emergency, can be isolated from the plant areas and the environment external to the CRE. This area is served by an emergency ventilation system, with the intent of maintaining the habitability of the control room. This area encompasses the control room, and may encompass other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident.

Control Room Habitability Systems (CRHSs): The systems, defined in the facility's licensing basis, that typically provide the functions of shielding, isolation, pressurization, heating, ventilation, air conditioning and filtration, monitoring, and sustenance and sanitation necessary to ensure that operators can remain in the control room, take actions to operate the plant under normal conditions, and maintain it in a safe condition during accident situations. The CRHSs include the CRE.

C. REGULATORY POSITION

1. An Overview of the Process of Demonstrating and Maintaining CRH

In demonstrating that a facility's control room conforms to the GDCs, the following CRH aspects are typically assessed:

- radiological doses
- protection from the effects of hazardous materials
- control of the reactor from either the control room or the alternate shutdown panel

The process of demonstrating the above aspects includes the following actions:

(a) identification of the licensing bases for the (i) CRHSs, (ii) areas adjacent to the CRE, and (iii) ventilation systems that serve or traverse the CRE and those located in areas adjacent to the CRE

(b) determinations of whether the design, configuration, and operation of the systems and areas identified in action (a) are consistent with the licensing bases

(c) determination of the performance characteristics for operating modes associated with radiological and hazardous chemical accidents

(d) calculation of the radiological dose consequences to control room operators

(e) evaluation of the habitability of the control room during a postulated hazardous chemical release

(f) assessment of whether a radiological, hazardous chemical, or smoke challenge could result in the inability of the control room operators to control the reactor from either the control room or, in the event of smoke, from the alternate shutdown panel

(g) maintenance and monitoring of the CRHSs

2. Demonstrating and Maintaining CRH

Regulatory Positions 2.1 through 2.7 provide guidance on the process of demonstrating and maintaining CRH.

2.1 <u>Identification of the Licensing Bases for the CRHSs</u>

2.1.1 *Identification of the Control Room and the CRE*

Confirmation of a facility's ability to meet CRH requirements begins with the identification of the control room and the CRE. A description of the control room and CRE may be contained in a number of plant documents. These documents might include the updated final safety analysis report (UFSAR), the original final safety analysis report (FSAR), the safety evaluation for the operating license (OL), system descriptions, plant drawings, operating procedures, plant amendment requests, NRC safety evaluations, Three Mile Island (TMI) Action Plan Item III.D.3.4 submittals, and responses to staff questions at the construction permit and OL stages.

2.1.2 *Determination of the Licensing Bases*

In demonstrating the habitability of a facility's control room, it is essential that the licensee know the facility's licensing bases for its CRHSs. The sources of the licensing bases of the CRHSs should be identified. Licensees should consider the documents identified in Section 4.3 of NEI 99-03 (Ref. 1) as potential sources that define the licensing bases for their CRHSs. Focusing on the events that may have established or changed these bases may help narrow this search.

Over the facility's lifetime, the licensing bases change. The staff may have reviewed and approved the licensing bases of facilities licensed before the issuance of this guide. The original licensing bases may have been submitted as part of the construction permit application. Licensees may have modified them in response to NRC questions. In addition, the licensing bases were part of the application for the OL (FSAR). Depending on the plant vintage, licensees may have modified their licensing bases in response to TMI Action Plan Item III.D.3.4. Amendments to the OL may have resulted in changes to the licensing bases of the CRHSs. Licensees should review the applicable plant changes to their licensing bases to determine the current bases.

A group of reactors received their construction permits or OLs before the GDCs were promulgated. During that time, proposed GDCs (sometimes called "Principal Design Criteria") were published in the *Federal Register* for comment. These proposed GDCs addressed CRH. Although facilities may have been licensed before the promulgation of the GDCs, licensees may have committed to the form of the GDCs as they existed at the time of licensing. A review of the record associated with the construction permit and OL proceedings should confirm whether licensees made such a commitment. Therefore, licensees that received their construction permits or OLs before the GDCs were promulgated should review their commitments to the draft form of the GDC to understand their CRH licensing bases.

For facilities licensed following the issuance of this regulatory guide, the sources for the description of the licensing bases will be the documents filed in support of the licensing application (under 10 CFR Parts 50 and 52).

2.2 Determination of Whether the CRHSs Are Consistent with the Licensing Bases

2.2.1 *Comparison of System Design, Configuration, and Operation with the Licensing Bases*

Licensees should compare the design, configuration, and operation of their CRHSs and the systems that are in adjacent areas and could interact with the CRE to their licensing bases to ensure consistency. The review of the configuration of the CRHSs should include the construction and the alignment of the systems and structures that make up the CRHSs. For new reactors and existing CRHSs undergoing redesign, this comparison should be made upon completion of construction. Section 5 of NEI 99-03 (Ref. 1) provides a method of comparing the plant's configuration and operation of ventilation systems with the licensing bases that is acceptable to the NRC staff with one clarification. Licensees should also establish the performance characteristics discussed in Regulatory Position 2.3.1 to ensure consistency between the operation of the control room ventilation systems and the licensing bases. Licensees should employ methods similar to those provided in Section 5 of NEI 99-03 when they perform these comparisons for other CRHSs.

2.2.2 *Interactions Between the CRHSs and Adjacent Areas*

The conditions that exist in the areas adjacent to the CRE influence the performance of the CRHSs. Although these systems might not be expected to operate during an emergency, during a loss of offsite power (LOOP), or with a single failure, inleakage may be increased if they do operate. Potential interactions between the CRHSs and adjacent areas that may increase the transfer of contaminants into the CRE should be identified. These interactions may be caused by ventilation systems that supply or exhaust air from areas adjacent to the CRE, are located in areas adjacent to the CRE, or have ductwork that traverses the CRE or areas adjacent to the CRE.

2.3 Determination of Performance Characteristics

2.3.1 *Performance of CRHSs*

The licensee should determine the performance characteristics of the CRE, its ventilation systems, and systems that serve or traverse areas within or located adjacent to the CRE. These parameters include, but are not limited to, differential pressures, system flow measurements (i.e., makeup and recirculation flow rates), duct static pressures, and filter differential pressures. Performance characteristics are needed to achieve the following objectives:

- Establish the operating parameters for incorporation into the licensing bases (for new reactors or those that have modified their CRE or associated ventilation systems).
- Determine the impact on systems caused by changes in the operation, design, alignment, or procedures.
- Define the limiting condition for the applicable design basis events.
- Determine new limiting conditions or perform new analyses.

Technical specifications require licensees to periodically perform measurements of several parameters important to maintaining CRH. These parameters may include system flow rates, carbon filter efficiencies, actuation signals, and CRE integrity tests. Engineered-safety-feature atmospheric cleanup systems in light-water-cooled nuclear power plants should be tested and evaluated in accordance with Regulatory Guide 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants" (Ref. 3).

In CRE integrity tests performed in 1991 – 2001 by approximately 30 percent of the licensed facilities, all but one facility measured greater inleakage than that assumed in the design analyses. In some cases, the measured inleakage exceeded the amount assumed in the design analyses by several orders of magnitude. Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors" (Ref. 4), provides guidance on this issue and an approach that the NRC staff considers acceptable to determine CRE integrity. As discussed in Regulatory Position 2.2.2, systems outside the CRE may impact CRE integrity. Testing may also be needed to understand the influence of these systems on CRH. Consistent with Regulatory Position 2.2.1, licensees should ensure that their assumed control room inleakage input value used in any accident calculations or evaluations (Regulatory Positions 2.4 and 2.5) are validated by the test methods provided in Regulatory Guide 1.197.

2.3.2 *Identification of the Limiting Condition*

The limiting condition for CRH is the configuration that results in the maximum consequences to the control room operators. Sometimes the limiting condition will arise from the configuration that produces the greatest inleakage and sometimes it will not. The latter situation can occur because the configuration that results in the largest inleakage may have mitigative features that result in smaller consequences to the control room operators. As an example, CRE inleakage may be greatest for a radiological accident that does not have a LOOP. However, the absence of a LOOP could provide mitigative features that reduce the overall consequences to the control room operators.

In determining the limiting condition for potential radiological accidents, it should not be presumed that the LOCA is the limiting accident because it has the largest initial source of activity. Other accidents (e.g., fuel handling accidents) may produce larger control room operator doses because the manner in which the CRHSs respond may provide less protection to the operators. Therefore, licensees should perform an analysis of the consequences of each applicable radiological accident as discussed in Regulatory Position 2.4 to ensure that they have identified the limiting accident.

Unless a facility relies on a common control room isolation process for all types of radiological accidents, the limiting accident may not be obvious for the following reasons:

- The inleakage characteristics of the envelope may vary with the CRHS's response to an accident.
- The mitigative equipment used to reduce the radioactivity released to the environment may vary with the accident.
- The location of the release points for the various accidents relative to the control room intakes may result in less favorable atmospheric dispersion and higher magnitude intake concentrations.

Licensees should factor all the potential differences in accidents and the CRHS's performance in order to determine the limiting condition.

A few plants are within the exposure range for an accidental release from a nearby nuclear plant or have separate control rooms for multiple units on the same site. An accident in an adjacent unit should not prevent the safe shutdown of an operating unit. Regulatory Position 2.6, "Reactor Control," describes criteria used for determining a safe shutdown of the reactor. The release point, atmospheric dispersion, and postulated source term from the adjacent unit should be reviewed to assess the impact on the operating unit's control room.

For hazardous chemicals, a logic process similar to that employed for radiological accidents should be used to determine the limiting condition.

2.4 Radiological Consequences

Licensees should calculate control room operator doses for the methodology and accidents identified in Regulatory Guide 1.195 (Ref. 5) or Regulatory Guide 1.183 (Ref. 6). For CREs under construction, the control room operators' doses should be based on expected CRHS performance values. When the envelope and associated ventilation systems are operational, the inleakage value should be determined using Regulatory Guide 1.197 (Ref. 4).

2.5 Hazardous Chemicals

Licensees should evaluate the impact of hazardous chemicals on control room operators using the methodology of Regulatory Guide 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release" (Ref. 7). Regulatory Guide 1.78 encourages licensees to conduct periodic surveys of stationary and mobile sources of hazardous chemicals in the vicinity of their plant sites. The periodicity should be based on the number, size, and type of industrial and transportation activities in the vicinity of the plant and regional and local changes in uses of land. The staff recommends conducting a survey of the location, types, and quantities of the mobile and stationary hazardous chemical sources at least once every 3 years, or more frequently as applicable. The staff also recommends annual performance of an onsite survey of hazardous chemical sources.

For CREs under construction, the hazardous chemical analysis should be based on the expected performance values. When the envelope and associated ventilation systems are operational, the calculation should be based on an inleakage value determined according to Regulatory Guide 1.197 (Ref. 4).

2.6 Reactor Control

This Regulatory Position provides guidance for assessing whether a radiological, hazardous chemical, or smoke challenge could result in the inability of the control room operators to control the reactor from the control room, or in the event of a smoke challenge, from the control room or the alternate shutdown panel. This regulatory position does not address the performance of the reactor controls and instrumentation systems that are affected by environmental conditions caused by a radiological, hazardous chemical, or smoke event, nor does it address human engineering (i.e., temperature, vibration, sound, or lighting).

Demonstrating the facility's ability to maintain a habitable control room includes ensuring that an accident arising from a radiological event, hazardous chemicals, or a smoke challenge will not prevent the control room operators from controlling the reactor. Facilities should demonstrate that they meet the reactor control aspects of their licensing bases (typically GDC-19). The specific acceptance criteria for radiological events are summarized in Regulatory Position 4.5 of Regulatory Guide 1.195 (Ref. 5) for plants employing TID-14844 source term methodology and Regulatory Position 4.4 of Regulatory Guide 1.183 (Ref. 6) for plants employing alternative source term methodology. The specific acceptance criterion for chemical events is given in Regulatory Position 3.1 of Regulatory Guide 1.78 (Ref. 7).

Smoke may be a CRH concern if there is significant inleakage from outside the envelope or if a fire develops in the control room. In these situations, smoke could challenge the operator's ability to shut down the reactor from within the control room or remotely. No regulatory limit exists on the amount of smoke allowed in the control room. Therefore, the plant's ability to manage smoke infiltration is assessed qualitatively. Licensees should perform a qualitative assessment to ensure that the plant can be safely shut down from either the control room or the alternate shutdown panel during an internal or external smoke event. The staff endorses Appendix E of NEI 99-03 (Ref. 1) as an acceptable method for performing this qualitative assessment with the following exceptions. The second sentence of Section 1 should read, "The guidance ensures that the operator maintains an ability to safely shut down the plant during a smoke event originating inside or outside the control room." Replace the words "fire/smoke event" in the first sentence of Section 2 with "smoke event originating from either inside or outside the control room." The title of Section 3 should be "Contingency Logic Evaluation," and the third bullet should be deleted. The last bullet in Section 3 should be the last bullet in Section 2.

2.7 Maintaining and Monitoring CRHSs

CRH is maintained and monitored during the operating life of the plant by a CRHS program. A CRHS program includes periodic evaluations, maintenance, configuration control, and training. This Regulatory Position covers CRHS programs, and it provides methods to mitigate degraded and nonconforming conditions when the plant does not meet the specific acceptance criteria given in Regulatory Position 2.6 or is outside its licensing basis. The following methods of maintaining and monitoring CRHSs should be used.

2.7.1 *Periodic Evaluations and Maintenance*

Periodic evaluations of CRH demonstrate that the CRHSs meet their functional criteria. These evaluations include periodic assessments and tests.

Periodic assessments of the CRHS's material condition, configuration controls, safety analyses, and operating and maintenance procedures should be performed. CRHS programs should assess the system and material conditions as described in Section 9.3.1, "System Material Condition," of NEI 99-03 (Ref. 1).

Licensees should perform testing to ensure they maintain CRH. Routine performance measurements are described in Regulatory Position 2.3.1. The complexity of testing following modifications should depend on the effect of the modification on CRH. Regulatory Guide 1.197 (Ref. 4) provides a testing method for verification of CRE integrity. A frequency for CRE integrity testing is provided in Regulatory Guide 1.197. Regulatory Position 2.5 above provides a method and a suggested frequency for evaluating the impact of hazardous chemicals on control room operators.

A maintenance program should be established for the CRHSs. Table H-1 of NEI 99-03 (Ref. 1) should be used to identify systems and components to be included in a maintenance program. Guidance on air filtration and adsorption units of post-accident engineered-safety-feature atmosphere cleanup and normal atmospheric cleanup system maintenance is provided in Regulatory Position 5 of Regulatory Guides 1.52 (Ref. 3) and 1.140, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants" (Ref. 13), respectively. ASHRAE Guideline 1-1996 (Ref. 14) may be used to establish a maintenance program for systems that handle hazardous chemical and smoke challenges.

2.7.2 *Configuration Control and Training*

Configuration control and training are effective tools that can minimize the impact that changes to CRHSs can have on CRH. Section 9.4 of NEI 99-03 (Ref. 1) provides configuration controls that include CRE boundary and breach control, procedure control, toxic gas control, design change, and safety analysis controls.

The staff endorses the controls discussed in Sections 9.4.1 through 9.4.5 of NEI 99-03 (Ref. 1) with two exceptions. The staff does not endorse Appendix K, "Control Room Envelope Boundary Control Program," referenced in Section 9.4.1, "CRE Boundary/Breach Control." Instead of endorsing the method of equating a breach size to an inleakage flow rate, the staff endorses the method of breach control contained in the STSs (Refs. 8, 9, 10, 11, 12), which allows the control room boundary to be opened intermittently under administrative controls.

For entry and exit through doors, the administrative control of the opening is performed by the persons entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for control room isolation is indicated. Regulatory Guides 1.183 or 1.195 should be used instead of Appendix C (referenced in Section 9.4.5); Regulatory Guide 1.194 should be used instead of Appendix D (referenced in Section 9.4.5); Regulatory Guide 1.78 should be used instead of Appendix G (referenced in Section 9.4.5) (Refs. 6, 5, 15, and 7).

Furthermore, the staff endorses Section 9.5, "Training," of NEI 99-03 with one exception. Section 9.5 recommends training using NEI 99-03. Instead, the NRC staff endorses training using only the sections of NEI 99-03 that the staff has endorsed.

2.7.3 *Degraded and Nonconforming Conditions*

Methods available to address short term degraded or nonconforming conditions are provided in Section 8.4, "Methods Available to Address Degraded or Nonconforming Conditions" of NEI 99-03 (Ref. 1). Section 8.4 includes guidance on compensatory measures such as self-contained breathing apparatus (SCBA) and potassium iodide (KI) tablets. These methods are acceptable with the following exceptions. Appendices C and D are not endorsed; instead, Regulatory Guides 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants" (Ref. 15), and 1.195 (Ref. 5), or Regulatory Guide 1.183 (Ref. 6) should be used. The staff endorses the use of the guidance in Appendix F of NEI 99-03 while corrective actions are being taken to resolve CRHSs that do not meet their licensing bases, subject to the following:

(1) Section 2.2 of Appendix F addresses the training and qualification of control room operators for SCBA. If SCBA units will be used as an interim compensatory measure for greater than 90 days while the plant is in Operating Condition or Mode 1, simulator crew training accident scenarios in which operators wear SCBAs should be performed. These scenarios should last about 2 hours and include a simulated watch turnover.

(2) Section 2.6 of Appendix F addresses the availability of adequate methods to refill depleted SCBA cylinders. The impact of a LOOP or airborne contamination at the refill compressor stations should be considered.

(3) Section 2 of Appendix F addresses additional guidance for evaluating the habitability of a control room during a chemical release. Replace the sentence beginning with "Additional guidance" with "Additional guidance is provided in Regulatory Guide 1.78, 'Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release'" (Ref. 7).

(4) Section 2.5 and 2.6 of Appendix F should be replaced with the following to correct an editorial error: "2.5 Persons Using Tight Fitting (Facepiece) Respirators Should Not Have Any Facial Hair That Interferes With the Sealing Surfaces of the Respirator. The required minimum staffing of control room operators qualified in SCBA use should be clean shaven.
2.6 Adequate Method(s) to Refill SCBA Air Cylinders Should Be Available."

Some licensees were allowed to leave TMI Action Item III.D.3.4 actions open until the alternative source term rulemaking and regulatory guidance were published. These actions were completed with the issuance of 10 CFR 50.67 and Regulatory Guide 1.183 (Ref. 6). The regulatory positions in this regulatory guide on control room habitability provide methods that the NRC staff considers acceptable for closing open TMI Action Plan Item III.D.3.4 actions.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for using this regulatory guide. No backfitting is intended or approved in connection with the issuance of this guide.

Except in those cases in which an applicant or licensee proposes or has previously established an acceptable alternative method for complying with specified portions of the NRC's regulations, the NRC staff will use the methods described in this guide to evaluate CRH for nuclear power plants for which the construction permit or license application is docketed after the issue date of this guide and plants for which the licensees voluntarily commit to the provisions of this guide.

REGULATORY ANALYSIS

The regulatory analysis for this regulatory guide is available in Draft Regulatory Guide DG-1114, "Control Room Habitability at Light-Water Nuclear Power Reactors."[3] The NRC issued DG-1114 in March 2002 to solicit public comment on the initial draft of Regulatory Guide 1.196.

BACKFIT ANALYSIS

The regulatory guide does not require a backfit analysis as described in 10 CFR 50.109(c) because it does not impose a new or amended provision in the NRC's rules or a regulatory staff position interpreting the NRC's rules that is either new or different from a previous applicable staff position. In addition, this regulatory guide does not require the modification or addition to systems, structures, components, or design of a facility or the procedures or organization required to design, construct, or operate a facility. Rather, a licensee or applicant may select a preferred method for achieving compliance with a license or the rules or orders of the Commission as described in 10 CFR 50.109(a)(7). This regulatory guide provides an opportunity to use part of an industry-developed standard.

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3 Draft Regulatory Guide DG-1114 is available electronically under Accession #ML020790125 in the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html. Copies are also available for inspection or copying for a fee from the NRC's Public Document Room (PDR), which is located at 11555 Rockville Pike, Rockville, Maryland; the PDR's mailing address is USNRC PDR, Washington, DC 20555-0001. The PDR can also be reached by telephone at (301) 415-4737 or (800) 397-4205, by fax at (301) 415-3548, and by email to PDR@nrc.gov.

REFERENCES

1. NEI 99-03, "Control Room Habitability Assessment Guidance," Revision 0, Nuclear Energy Institute, Washington, DC, June 2001.[1]

2. Regulatory Guide 1.186, "Guidance and Examples for Identifying 10 CFR 50.2 Design Bases," U.S. Nuclear Regulatory Commission, Washington, DC, December 2000.[2]

3. Regulatory Guide 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," Revision 3, U.S. Nuclear Regulatory Commission, Washington, DC, June 2001.[2]

4. Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," U.S. Nuclear Regulatory Commission, Washington, DC, May 2003.[2]

5. Regulatory Guide 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design-Basis Accidents at Light-Water Nuclear Power Reactors," U.S. Nuclear Regulatory Commission, Washington, DC, May 2003.[2]

6. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," U.S. Nuclear Regulatory Commission, Washington, DC, July 2000.[2]

7. Regulatory Guide 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," Revision 1, U.S. Nuclear Regulatory Commission, Washington, DC, December 2001.[2]

8. NUREG-1431, "Standard Technical Specifications Westinghouse Plants: Specifications," Volume 1, Revision 2, U.S. Nuclear Regulatory Commission, Washington, DC, June 2001.[3] (ADAMS Accession Number ML011840223)

9. NUREG-1430, "Standard Technical Specifications Babcock and Wilcox Plants," Volume 1, Revision 2, U.S. Nuclear Regulatory Commission, Washington, DC, June 2001.[3] (ADAMS Accession Number ML011770186)

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[1] Copies are also available for inspection or copying for a fee from the NRC's Public Document Room (PDR), which is located at 11555 Rockville Pike, Rockville, Maryland; the PDR's mailing address is USNRC PDR, Washington, DC 20555-0001. The PDR can also be reached by telephone at (301) 415-4737 or (800) 397-4205, by fax at (301) 415-3548, and by email to PDR@nrc.gov.

[2] Requests for single copies of draft or active regulatory guides (which may be reproduced) or for placement on an automatic distribution list for single copies of future draft guides in specific divisions should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Reproduction and Distribution Services Section, or by fax to (301)415-2289; email DISTRIBUTION@nrc.gov. Copies are also available for inspection or copying for a fee from the NRC's Public Document Room (PDR), which is located at 11555 Rockville Pike, Rockville, Maryland; the PDR's mailing address is USNRC PDR, Washington, DC 20555-0001. The PDR can also be reached by telephone at (301) 415-4737 or (800) 397-4205, by fax at (301) 415-3548, and by email to PDR@nrc.gov.

[3] Available through NRC's Public Web site, http://www.nrc.gov, in the Electronic Reading Room through ADAMS (using the accession number) and online at http://www.nrc.gov/reactors/operating/licensing/techspecs.html.

10. NUREG-1432, "Standard Technical Specifications Combustion Engineering Plants," Volume 1, Revision 2, U.S. Nuclear Regulatory Commission, Washington, DC, June 2001.[3] (ADAMS Accession Number ML011930335)

11. NUREG-1433, "Standard Technical Specifications General Electric Plants, BWR/4," Volume 1, Revision 2, U.S. Nuclear Regulatory Commission, Washington, DC, June 2001.[3] (ADAMS Accession Number ML011780639)

12. NUREG-1434, "Standard Technical Specifications General Electric Plants, BWR/6," Volume 1, Revision 2, U.S. Nuclear Regulatory Commission, Washington, DC, June 2001.[3] (ADAMS Accession Number ML011780537)

13. Regulatory Guide 1.140, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," Revision 2, U.S. Nuclear Regulatory Commission, Washington, DC, June 2001.[2]

14. ASHRAE Guideline 1-1996, "The HVAC Commissioning Process," American Society of Heating, Refrigerating and Air Conditioning Engineers, Atlanta, Georgia, June 1996.

15. Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC, May 2003.[2]

APPENDIX A

REGULATORY GUIDE ENDORSEMENT
OF NEI 99-03, REV. 0, JUNE 2001, BY SECTION

NEI 99-03 Section	NEI 99-03 Section Title	Endorsement Status	Regulatory Guide Section	Exceptions/Remarks
1	Introduction	Not addressed	N/A	N/A
1.1	Purpose And Scope	Not addressed	N/A	N/A
1.2	History	Not addressed	N/A	N/A
1.3	Document Organization	Not addressed	N/A	N/A
2	Regulatory Requirements and Guidance	Not addressed	N/A	N/A
2.1	Purpose And Scope	Not addressed	N/A	N/A
2.2	Regulatory Requirement: General Design Criterion 19	Not addressed	N/A	N/A
2.3	Regulatory Guidance	Not addressed	N/A	N/A
2.3.1	Regulatory Guides	Not addressed	N/A	N/A
2.3.2	NUREGs	Not addressed	N/A	N/A
2.3.3	Information Notices	Not addressed	N/A	N/A
2.4	Generic Issues	Not addressed	N/A	N/A

NEI 99-03 Section	NEI 99-03 Section Title	Endorsement Status	Regulatory Guide Section	Exceptions/Remarks
3	Industry Issues Associated with Control Room Habitability	Not addressed	N/A	N/A
3.1	Purpose And Scope	Not addressed	N/A	N/A
3.2	Licensing Basis Different From As-Built Plant	Not addressed	N/A	N/A
3.3	Analyses Different From As-built or As-Operated Plant	Not addressed	N/A	N/A
3.4	DBA Analyzed Not Most Limiting	Not addressed	N/A	N/A
3.4.1	Adjacent Unit Accident (a Special Case)	Not addressed	N/A	N/A
3.5	Smoke Infiltration	Not addressed	N/A	N/A
3.6	Toxic Gas Evaluation	Not addressed	N/A	N/A
3.7	Control Room Air In-Leakage Greater Than Assumed	Not addressed	N/A	N/A
3.7.1	Radiological Considerations	Not addressed	N/A	N/A
3.7.2	Toxic Gas Considerations	Not addressed	N/A	N/A

NEI 99-03 Section	NEI 99-03 Section Title	Endorsement Status	Regulatory Guide Section	Exceptions/Remarks
4	Determining CRH Licensing Basis	Not addressed	N/A	N/A
4.1	Purpose And Scope	Not addressed	N/A	N/A
4.2	Understanding the Concept of Licensing Basis	Not addressed	N/A	N/A
4.2.1	Design Basis	Not addressed	N/A	N/A
4.2.2	Supporting Design Information	Not addressed	N/A	N/A
4.2.3	Licensing Basis	Not addressed	N/A	N/A
4.3	Licensing Basis Sources	Full endorsement	Regulatory Guide 1.196, Regulatory Position (RP) 2.1.2	N/A
4.4	Performing the Licensing Basis Review	Not addressed	N/A	N/A
4.5	Assembling the CRH Analysis	Not addressed	N/A	N/A
4.6	Documentation of the Existing Plant CRH Licensing and Design Basis	Not addressed	N/A	N/A

NEI 99-03 Section	NEI 99-03 Section Title	Endorsement Status	Regulatory Guide Section	Exceptions/Remarks
5	Comparing Existing Plant Configuration and Operations with Licensing Bases for CRH	Full endorsement	RG 1.196, RP 2.2.1	This section provides a method of comparing the plant's configuration and operation of ventilation systems with the licensing bases that is acceptable to the NRC staff with one clarification. Licensees should also establish the performance characteristics discussed in Regulatory Position 2.3.1 to ensure consistency between the operation of the control room ventilation systems and the licensing bases.
5.1	Purpose and Scope	Full endorsement	RG 1.196, RP 2.2.1	N/A
5.2	Review the As-Built Control Room Envelope and Control Room Ventilation Systems	Full endorsement	RG 1.196, RP 2.2.1	N/A
5.3	Review the Normal and Emergency Operating Procedures Affecting the Control Room Ventilation Systems	Full endorsement	RG 1.196, RP 2.2.1	N/A
5.4	Review the Testing Procedures Affecting Control Room Ventilation Systems and the Associated Envelope	Full endorsement	RG 1.196, RP 2.2.1	N/A
5.5	Review the Maintenance Practices and Procedures for Effect on CRH Requirements	Full endorsement	RG 1.196, RP 2.2.1	N/A
5.6	Review the Plant Modification Procedures for Consideration of the CRH Requirements	Full endorsement	RG 1.196, RP 2.2.1	N/A
5.7	Review the CRH Analyses	Full endorsement	RG 1.196, RP 2.2.1	N/A
5.8	Identified Inconsistencies	Full endorsement	RG 1.196, RP 2.2.1	N/A

NEI 99-03 Section	NEI 99-03 Section Title	Endorsement Status	Regulatory Guide Section	Exceptions/Remarks
6	Assessing Industry Issue Applicability	Not addressed	N/A	N/A
6.1	Purpose and Scope	Not addressed	N/A	N/A
6.2	Limiting DBA	Not addressed	N/A	N/A
6.2.1	Recommended Actions to Evaluate Limiting DBA	Not addressed	N/A	N/A
6.2.2	Adjacent Unit Accidents	Not addressed	N/A	N/A
6.3	Smoke Infiltration	Not endorsed	N/A	N/A
6.3.1	Recommended Licensee Action to Address Smoke Infiltration	Not endorsed	N/A	N/A
6.4	Toxic Gas Evaluation	Not addressed	N/A	N/A
6.4.1	Recommended Licensee Action to Address Toxic Gas Evaluation	Not Addressed	N/A	N/A
7	Measuring Air In-Leakage (Baseline Test)	Not addressed	N/A	N/A
7.1	Purpose and Scope	Not addressed	N/A	N/A
7.2	Preparation for Testing	Not addressed	N/A	N/A
7.3	Test Performance	Not addressed	N/A	N/A
7.4	Resolution of Identified Issues	Not addressed	N/A	N/A

NEI 99-03 Section	NEI 99-03 Section Title	Endorsement Status	Regulatory Guide Section	Exceptions/Remarks
8	Dispositioning and Managing Discrepancies	Not addressed	N/A	N/A
8.1	Purpose and Scope	Not addressed	N/A	N/A
8.2	Generic Letter 91-18	Not addressed	N/A	N/A
8.3	Determining Operability and Reportabilty	Not addressed	N/A	N/A
8.4	Methods Available to Address Degraded or Nonconforming Conditions	Partial endorsement	RG 1.196, RP 2.7.3	Appendices C and D are not endorsed. Appendix F exceptions related to: 1. Training and qualification of control room operators for SCBA 2. Availability of adequate methods to refill SCBA 3. Two typographical errors as described in the text of this guide.
8.4.1	Compensatory Measures	Partial endorsement	RG 1.196, RP 2.7.3	N/A
8.4.2	Dose Analysis Revision Option	Partial endorsement	RG 1.196, RP 2.7.3	N/A
8.4.3	Repairing or Modifying the Plant	Partial endorsement	RG 1.196, RP 2.7.3	N/A
8.4.4	Technical Specification Changes	Partial endorsement	RG 1.196, RP 2.7.3	N/A

NEI 99-03 Section	NEI 99-03 Section Title	Endorsement Status	Regulatory Guide Section	Exceptions/Remarks
9	Long-term CRH Integrity Program	Not addressed	N/A	N/A
9.1	Purpose and Scope	Not addressed	N/A	N/A
9.2	CRH Integrity Program	Not addressed	N/A	N/A
9.3	Periodic Evaluations	Not addressed	N/A	N/A
9.3.1	System Material Condition	Full endorsement	RG 1.196, RP 2.7.1	N/A
9.3.2	Post-maintenance Activities	Not addressed	N/A	N/A
9.3.3	In-leakage Assessments	Not addressed	N/A	N/A
9.3.3.1	Assessment Scope	Not addressed	N/A	N/A
9.3.3.2	Assessment Frequency	Not addressed	N/A	N/A
9.3.3.3	Determine Need to Test	Not addressed	N/A	N/A
9.3.4	Toxic Gas Evaluation	Not addressed	N/A	N/A
9.4	Configuration Control	Partial Endorsement	RG 1.196, RP 2.7.2	The NRC staff references the configuration controls in Section 9.4. These include CRE boundary and breach control, procedure control, toxic gas control, design change, and safety analysis controls.
9.4.1	CRE Boundary / Breach Control	Partial endorsement	RG 1.196, RP 2.7.2	The NRC staff does not endorse Appendix K.
9.4.2	Procedure Control	Full endorsement	RG 1.196, RP 2.7.2	N/A
9.4.3	Toxic Chemical Control	Full endorsement	RG 1.196, RP 2.7.2	N/A
9.4.4	Design Change Control	Full endorsement	RG 1.196, RP 2.7.2	N/A

NEI 99-03 Section	NEI 99-03 Section Title	Endorsement Status	Regulatory Guide Section	Exceptions/Remarks
9.4.5	Safety Analyses Control	Partial endorsement	RG 1.196, RP 2.7.2	Rather than endorse Appendices C, D, and G referenced in Section 9.4.5, Regulatory Guides 1.183, 1.194, 1.195, and 1.78 should be used to provide analysis assumptions used in safety analyses.
9.5	Training	Partial endorsement	RG 1.196, RP 2.7.2	The NRC staff endorses training using only the sections of NEI 99-03 that the staff has endorsed.
9.6	Testing	Not addressed	N/A	N/A
10	References	Not addressed	N/A	N/A
Appendix A	Licensing Basis History	Not addressed	N/A	N/A
Appendix B	Control Room Habitability Regulatory Information	Not addressed	N/A	N/A
Appendix C	CRH Dose Analysis: Regulatory Enhancements	Not endorsed	RG 1.196, RP 2.7.3	Regulatory Guide 1.195 should be used.
Appendix D	Atmospheric Dispersion	Not endorsed	RG 1.196, RP 2.7.3	Regulatory Guide 1.194 should be used.
Appendix E	Smoke Infiltration Impact on Safe Shutdown	Partial endorsement	RG 1.196, RP 2.6	The NRC staff endorses Appendix E as an acceptable method for performing this qualitative assessment with exceptions stated in Regulatory Position 2.6. However, the reference to Section 6 is not endorsed. Remove the words “as described in Section 6” in the first sentence.

NEI 99-03 Section	NEI 99-03 Section Title	Endorsement Status	Regulatory Guide Section	Exceptions/Remarks
Appendix F	Compensatory Measures Allowable on an Interim Basis	Partial endorsement	RG 1.196, RP 2.7.3	Appendix F exceptions relate to: 1. Training and qualification of control room operators for SCBA 2. The impact of a loss of offsite power or airborne contamination at the refill compressor stations. 3. Two typographical errors as described in the text of this guide.
Appendix G	Toxic Gas Assessments	Not Addressed	N/A	Regulatory Guide 1.78 should be used.
Appendix H	System Assessment	Partial endorsement	RG 1.196, RP 2.7.1	The NRC staff endorses Table H-1 as guidance for developing a maintenance program.
Appendix I	Testing Program	Not Addressed	N/A	N/A
Appendix J	Control Room Envelope Sealing Program	Not addressed	N/A	N/A
Appendix K	Control Room Envelope Boundary Control Program	Not endorsed	RG 1.196, RP 2.7.2	The staff does not endorse Appendix K. Instead of endorsing the method of equating a breach size to an inleakage flow rate, the staff endorses the method of breach control contained in the STSs (NUREG-1431, NUREG 1430, NUREG-1432, NUREG-1433, and NUREG-1434) (Refs. 8–12), which allows the control room boundary to be opened intermittently under administrative controls.
Appendix L	Glossary of Terms	Not addressed	N/A	N/A

APPENDIX B

ACRONYMS

ASHRAE	American Society for Heating, Refrigeration and Air-Conditioning Engineers
ASME	American Society of Mechanical Engineers
B&W	Babcock and Wilcox
BWR	Boiling Water Reactor
CE	Combustion Engineering
CRE	Control Room Envelope
CRH	Control Room Habitability
CRHS	Control Room Habitability System
ASTM	American Society for Testing and Materials
ESF	Engineered Safety Feature
FSAR	Final Safety Analysis Report
GE	General Electric
GDC	General Design Criteria
LOCA	Loss-of-Coolant Accident
LOOP	Loss of Offsite Power
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
OL	Operating License
OMB	Office of Management and Budget
SCBA	Self-Contained Breathing Apparatus
SRP	Standard Review Plan
SSC	Structures, Systems, and Components
STS	Standard Technical Specification
TMI	Three Mile Island
TSTF	Technical Specification Task Force
TSC	Technical Support Center
UFSAR	Updated Final Safety Analysis Report